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AUXILIARY SYSTEMS

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NOTE:

(a) This figure corresponds to a controlled engineering drawing that is incorporated by reference into the FSAR Update. See Table 1.6-1 for the correlation between the FSAR Update figure number and the corresponding controlled engineering drawing number.

Chapter 9

APPENDICES

Appendix	Title

Deleted in Revision 24

Chapter 9

AUXILIARY SYSTEMS

This chapter discusses auxiliary systems installed in Unit 1 and Unit 2 at the Diablo Canyon Power Plant (DCPP) site. Fuel storage and handling systems; water systems; process auxiliaries; and air conditioning, heating, cooling, and ventilation systems are described as well as other auxiliary systems. The design classifications for these various systems and their associated structures and components are discussed in Section 3.2.

9.1 FUEL STORAGE AND HANDLING

The fuel storage and handling systems provide safe and effective means of storing, transporting, and handling new and irradiated nuclear fuel. These systems are located mainly in the fuel handling building (FHB), adjacent to the east walls of the containment structures. Separate facilities are provided for each unit.

The fuel storage and handling systems comply with the criticality accident requirements of 10 CFR 50.68(b), "Criticality Accident Requirements," in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24, "Criticality Accident Requirements." In accordance with 10 CFR 50.68(b)(6), radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

9.1.1 NEW FUEL STORAGE

New fuel is stored in new fuel storage racks (NFSRs), which are in storage vaults in the FHB, or in the spent fuel pool (SFP) storage racks, as discussed in Section 9.1.2. The NFSRs are designed to store, protect, and prevent criticality of new fuel assemblies until used within the reactor.

9.1.1.1 Design Bases

Refer to Section 9.1.1.3.6 for a discussion of k_{eff} limits including allowance for uncertainties.

9.1.1.1.1 General Design Criterion 2, 1967 – Performance Standards

New fuel storage is located in the FHB, which is designed to withstand the effects of, or is protected against, natural phenomena, such as earthquakes, tornadoes, flooding, winds, and tsunamis, and other local site effects.

9.1.1.1.2 General Design Criterion 3, 1971 – Fire Protection

The NFSRs are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.1.1.1.3 General Design Criterion 11, 1967 – Control Room

The NFSRs are designed to or contain instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room.

9.1.1.1.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the NFSR variables within prescribed operating ranges. NFSR area radiation monitoring equipment is provided in the control room.

9.1.1.1.5 General Design Criterion 18, 1967 – Monitoring Fuel and Waste Storage

The NFSRs are provided with monitoring and alarm instrumentation for conditions that might contribute to radiation exposure.

9.1.1.1.6 General Design Criterion 66, 1967 – Prevention of Fuel Storage Criticality

The NFSRs are designed to prevent new fuel criticality through physical systems or processes such as geometrically safe configurations.

9.1.1.1.7 New Fuel Storage System Safety Function Requirements

(1) Protection from Missiles

The NFSRs are designed to be protected against the effects of missiles that might result from plant equipment failures and from events and conditions outside the plant.

(2) Protection Against High Energy Pipe Rupture Effects

The NFSRs are designed to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to prevent a criticality accident.

9.1.1.1.8 10 CFR 50.68(b) – Criticality Accident Requirements

The NFSRs are designed to support compliance with the applicable requirements of 10 CFR 50.68(b) to prevent inadvertent criticality.

9.1.1.2 System Description

There are two NFSRs for each unit. Each NFSR is approximately 9 feet 6 inches wide, 13 feet long, and 13 feet 6 inches high (excluding centering cones). It is built from Type 304 stainless steel.

The storage cells in the NFSRs are in seven rows, five deep, and are spaced to have a nominal center-to-center distance of 22 inches. They are of Type 304 stainless steel and have a cone shaped top entrance to facilitate loading of fuel elements. They are shaped in a 9-inch square (cross-section) hollow beam configuration, standing upright.

The NFSRs are designed in accordance with the American Institute of Steel Construction (AISC) Standard AISC 360-1969, Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, published on February 12, 1969. The ASME BPVC Section III-1983 is used to determine allowable limits for materials not addressed by the AISC specification.

The NFSRs are designed to withstand a vertical (uplift) force of 4000 pounds in the unlikely event that an assembly would bind in the NFSR while being lifted by the SFP bridge crane.

The NFSRs are located in the FHB at elevation 125 feet. Assembly access is from elevation 140 feet.

For each unit, new fuel assemblies with possible inserts (e.g., rod cluster control assemblies [RCCAs]) are stored to facilitate the unloading of new fuel assemblies from trucks. The storage vaults are designed to hold new fuel assemblies in NFSRs and are utilized primarily for the temporary storage of the replacement fuel every cycle. The storage vault for each unit consists of two NFSRs with 35 storage cells per NFSR (70 cells per unit). The cells are arranged in a 5x7 array for each NFSR.

9.1.1.3 Safety Evaluation

9.1.1.3.1 General Design Criterion 2, 1967 – Performance Standards

The NFSRs are located in the FHB which is a PG&E Design Class I structure. This building is designed to withstand the effects of winds (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), and earthquakes (refer to Section 3.7), and to protect PG&E Design Class I structures, systems and components (SSCs) to ensure their safety functions are maintained.

The NFSRs and the anchorage of the NFSRs to the floor are designed for the Design Earthquake (DE) and Double Design Earthquake (DDE) loading conditions and evaluated for a Hosgri seismic event. In addition, the NFSRs must be capable of maintaining the horizontal center-to-center spacing of the fuel assemblies, and of supporting assemblies vertically under postulated seismic events.

Each NFSR is seismically qualified to store only six assemblies; one at each corner, in addition to one assembly in the central cell at the East and West ends. Each fuel assembly is assumed to also contain an insert component (e.g. RCCA). The assumed combined weight of each fuel assembly and insert is 1570 pounds. Also, together with these six assemblies and their potential inserts, a maximum of two additional insert components may be stored in each NFSR in cells face-adjacent to the corner cells in diagonally opposite corners. The seismic qualification is based on an assumed weight of each of the two additional inserts of 170 pounds.

9.1.1.3.2 General Design Criterion 3, 1971 – Fire Protection

The NFSRs are designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.1.1.3.3 General Design Criterion 11, 1967 – Control Room

The area radiation monitoring equipment for the new fuel storage area is annunciated in the control room.

9.1.1.3.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided to monitor area radiation variables within specified operating ranges.

Radiation levels are monitored locally in the FHB. Signal input is provided to local alarms and main control room annunciators, and to the fuel handling building ventilation system (FHBVS) to automatically initiate iodine removal mode operation (refer to Section 9.4.4.2).

9.1.1.3.5 General Design Criterion 18, 1967 – Monitoring Fuel and Waste Storage

Radiation monitors are provided in storage areas and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions. New fuel storage is provided with monitoring and alarm instrumentation for conditions that might contribute to radiation exposure (refer to Section 11.4.2.1.3 and Table 11.4-1).

9.1.1.3.6 General Design Criterion 66, 1967 – Prevention of Fuel Storage Criticality

The fuel storage criticality analysis assumes the NFSRs are completely filled with 5.0 weight percent U-235 fuel with no credit taken for any burnable absorber that may be present in the fuel assemblies (e.g., integral fuel burnable absorber [IFBA]). Although the storage vault containing the NFSRs is normally dry, two accident scenarios were considered as part of the NFSRs' design bases: (1) when fully flooded with unborated water, a $k_{eff} \leq 0.95$ must be maintained after allowing for calculational uncertainties, and (2) when flooded with aqueous foam, a $k_{eff} \leq 0.98$ must be maintained after allowing for calculational uncertainties.

For each case, calculations were made for both the Westinghouse standard and optimized fuel assembly (OFA) designs, which have different fuel rod diameters. The standard fuel gave the higher reactivity for the aqueous foam case whereas the OFA fuel gave the higher reactivity under the fully flooded accident condition. For the fully flooded case, the calculated k_{eff} was 0.9380 ± 0.0069 (95% probability, 95% confidence level). For the aqueous foam case, the calculated k_{eff} was 0.8949 ± 0.0053 (95% probability, 95% confidence level). Thus, allowing for all uncertainties, the maximum k_{eff} was 0.9449 for the flooded case, and 0.9002 for the aqueous foam case. These maximum values are within their respective 0.95 and 0.98 10 CFR 50.68(b)(2) and 10 CFR 50.68(b)(3) limits.

Center-to-center assembly spacing is held to a tolerance of $\pm$1/16 inch to ensure a K_{eff} of less than 0.95, even when the storage vault is flooded with unborated water. After the NFSRs were installed, a dummy fuel element was inserted in each location and critical measurements taken to ensure proper arrangement and support. A metal cap covers the top of the NFSR. If a fuel assembly is accidentally dropped, it will only be able to drop into a holder and could not drop into the space between fuel assemblies. Refer to Section 15.4.5 for a discussion of fuel handling accidents (FHAs).

9.1.1.3.7 New Fuel Storage System Safety Function Requirements

(1) Protection from Missiles

There are no credible missiles outside of containment resulting from plant equipment failure that would prevent the PG&E Design Class I NFSRs from performing their design functions (refer to Section 3.5.3.3). Protection of the NFSRs from the effects of missiles and protection of PG&E Design Class I SSCs from damage that may result from these events is discussed in Section 3.5.

(2) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect the PG&E Design Class I NFSRs from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Section 3.6.

9.1.1.3.8 10 CFR 50.68(b) – Criticality Accident Requirements

The new fuel storage and handling systems comply with the criticality accident requirements of 10 CFR 50.68(b), “Criticality Accident Requirements,” in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24, “Criticality Accident Requirements.”

In accordance with 10 CFR 50.68(b)(1), plant procedures prohibit handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical.

Refer to Section 9.1.1.3.6 for a discussion of implementation of 10 CFR 50.68(b)(2) and 10 CFR 50.68(b)(3), with regard to k_{eff} limits for new fuel storage in the NFSRs assuming other-than-dry conditions.

The requirements in 10 CFR 50.68(b)(4) and 10 CFR 50.68(b)(5) regard spent fuel and special nuclear material other than nuclear fuel, respectively, and are therefore not applicable to the NFSRs.

In accordance with 10 CFR 50.68(b)(6), radiation monitors are provided in storage and associated handling areas when new fuel is present to detect excessive radiation levels and to initiate appropriate safety actions (refer to Sections 9.1.1.3.4 and 9.1.1.3.5).

Refer to Section 9.1.1.3.6 for information on implementation of 10 CFR 50.68(b)(7) regarding maximum nominal U-235 enrichment of new fuel assemblies.

Implementation of 10 CFR 50.68(b)(8) was completed with the submittal of Revision 15 of the UFSAR.

9.1.1.4 Tests and Inspections

New fuel is stored in the NFSRs prior to removal for receipt inspection and placement in the SFP. Tests and inspections are conducted in accordance with plant procedures.

9.1.1.5 Instrumentation Applications

The NFSR radiation monitor provides a signal to the ventilation control logic to realign the FHBVS exhaust flow through the charcoal filter banks (Iodine Removal Mode) upon sensing a high radiation condition (refer to Section 9.1.1.3.4).

9.1.2 SPENT FUEL POOL STORAGE SYSTEM

The SFP, shown in Figure 9.1-2, is the storage space in the DCPP 10 CFR Part 50 facilities for irradiated spent fuel from the reactor. New fuel may also be stored in the SFP. The figure shows the arrangement of the SFP storage racks. The SFP is not required for any plant operating mode safety-related function. As described in Section 3.2, the SFP concrete structure is PG&E Design Class I. Two pools are provided, one for each unit. The Unit 2 SFP is a mirror image of the Unit 1 SFP, reflected around column line 12[9].

The SFP for each unit comprises high density fuel storage racks, the SFP liner, the SFP liner leakage detection system, and key instruments to monitor SFP level, temperature and rate of temperature change, and local radiation levels. The SFP interfaces with the SFP cooling and cleanup system (refer to Section 9.1.3) and the fuel handling system (FHS) (refer to Section 9.1.4).

Refer to Section 9.1.4.2.6 for information on the Diablo Canyon independent spent fuel storage installation (ISFSI).

9.1.2.1 Design Bases

9.1.2.1.1 General Design Criterion 2, 1967 – Performance Standards

The SFP is designed to withstand the effects of, or is protected against, natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.1.2.1.2 General Design Criterion 3, 1971 – Fire Protection

The SFP is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.1.2.1.3 General Design Criterion 11, 1967 – Control Room

The SFP is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room.

9.1.2.1.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the SFP variables within prescribed operating ranges.

9.1.2.1.5 General Design Criterion 18, 1967 – Monitoring Fuel and Waste Storage

The SFP is provided with monitoring and alarm instrumentation for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

9.1.2.1.6 General Design Criterion 66, 1967 – Prevention of Fuel Storage Criticality

The SFP is designed to prevent criticality in SFP-stored new and spent fuel through physical systems or processes such as geometrically safe configurations.

9.1.2.1.7 General Design Criterion 68, 1967 – Fuel and Waste Storage Radiation Shielding

The SFP is designed to provide shielding for radiation protection to meet the requirements of 10 CFR Part 20.

9.1.2.1.8 General Design Criterion 69, 1967 – Protection Against Radioactivity Release from Spent Fuel and Waste Storage

The SFP is designed to provide containment of radioactive releases to the public environs as a result of an accident.

9.1.2.1.9 Spent Fuel Pool Storage System Safety Function Requirements

(1) Protection from Missiles

The SFP is designed to be protected against the effects of missiles that might result from plant equipment failures and from events and conditions outside the plant.

(2) <u>Protection Against High Energy Pipe Rupture Effects</u>

The SFP is designed to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary for the PG&E Design Class I SFP to perform its design functions.

9.1.2.1.10 10 CFR 50.68(b) – Criticality Accident Requirements

The SFP is designed to support compliance with the applicable requirements of 10 CFR 50.68(b) to prevent inadvertent criticality, with a noted exemption.

9.1.2.1.11 Safety Guide 13, March 1971 – Fuel Storage Facility Design Basis

<u>Regulatory Position 1</u>:

The SFP is designed to Category I seismic requirements.

<u>Regulatory Position 2</u>:

The PG&E Design Class I FHB that houses the SFP is designed to prevent tornadoes and tornado-borne missiles from causing significant loss of watertight integrity of the SFP and to prevent tornado-borne missiles from contacting fuel within the SFP.

Regulatory Position 5.b:

The SFP is designed to withstand, without leakage which could uncover the fuel, the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted.

Regulatory Position 7:

The SFP design provides for reliable monitoring equipment that will alarm both locally and in the control room if the SFP water level falls below a predetermined level or if high local radiation levels are experienced.

9.1.2.2 System Description

The SFP for each unit is a reinforced concrete structure with seam-welded stainless steel plate liner. The floor elevation of the SFP is nominally at elevation 99 feet, and the normal water surface is at elevation 137 feet 8 inches. The SFP storage racks rest on bridge plates supported on the floor and have a total height of 14 feet 11 inches. Thus normal water depth over the SFP storage racks is 23 feet 9 inches. DCPP Technical Specifications require that the SFP level be greater than or equal to 23 feet over the top of irradiated fuel assemblies seated in the storage racks during movement of irradiated fuel.

The pool is filled with borated water at a concentration greater than or equal to 2000 ppm boron as discussed in the Technical Specifications (Reference 2). Additional borated water to maintain this concentration is supplied from the refueling water storage tank (RWST) via the refueling water purification (RWP) system, or from other sources within the chemical and volume control system (CVCS).

Each SFP is designed to hold 1324 assemblies in the SFP storage racks, allowing for the concurrent storage of a full core of irradiated fuel assemblies and a managed quantity of spent fuel assemblies from reactor refueling operations. RCCAs and burnable poison rods requiring removal from the reactor are normally stored in the spent fuel assemblies.

The high density SFP storage racks for each fuel pool consist of a total of 16 stainless steel racks of various sizes, with a total of 1324 fuel assembly storage cells plus 10 miscellaneous storage locations. Individual storage cells have an 8.85-inch (nominal) square cross-section, and each is sized to contain and protect a single Westinghouse-type pressurized water reactor (PWR) 17x17 fuel assembly. The cells are arranged with a nominal 11-inch center-to-center spacing in the 16 rack modules.

The SFP storage racks are freestanding, with no connection to the SFP floor, walls, or adjacent SFP storage rack modules. The SFP storage rack support feet rest on bridge plates on the SFP floor. Each module is equipped with a girdle bar on the outside of each of the modules' four sides, near the top. Each girdle bar serves as a designated impact location, and each is designed to accommodate impact loads, which may occur during a seismic event. They also maintain a specified minimum gap between the cell walls of adjacent SFP storage rack modules for all loading conditions.

Adjacent to the SFP is the stainless-steel-lined fuel transfer canal, which is connected to the refueling cavity (inside the containment). The transfer canal is separated from the SFP by a gate. An inflatable gate seal provides leak protection.

Almost all components in contact with the SFP water (handling tools, new fuel elevator, etc.) are constructed of stainless steel, which has very good corrosion resistance. There are a few components made of bronze, which also has good corrosion resistance in aqueous environments. The compatibility of bronze and stainless steel maintains the integrity of the stainless steel components in the SFP.

A cooling and cleanup system for the SFP water is described in Section 9.1.3.

9.1.2.3 Safety Evaluation

Refer to Section 9.1.2.3.10 for a discussion of k_{eff} limits including allowance for uncertainties.

9.1.2.3.1 General Design Criterion 2, 1967 – Performance Standards

The FHB is PG&E Design Class I (refer to Section 3.8) and contains the PG&E Design Class I SFP and SFP storage racks. These components are designed to withstand the effects of winds and tornadoes (refer to Section 3.3); floods and tsunamis (refer to Section 3.4); external missiles (refer to Section 3.5); and earthquakes (refer to Section 3.7). Note that some metal siding on the FHB may detach during the postulated tornado.

In General Electric Topical Report APED - 5696, Tornado Protection for the Spent Fuel Storage Pool (refer to Reference 5, Section 3.3.3), a highly conservative model demonstrates that removal of more than 5 feet of water by a tornado mechanism is highly improbable. The 20 feet of water cover remaining over the fuel racks is shown to provide adequate protection against both fuel damage and liner penetration from a wide spectrum of tornado-generated missiles ranging up to a 3 inch diameter steel cylinder (7 feet long) or a 14 inch diameter wood pole (12 feet long). A potential for damage can only be shown by arbitrarily assuming long cylindrical objects hurled into the pool by winds acting on their maximum cross-sectional area and then impacting the pool with their minimum cross-sectional area. The probability of such an event is calculated to be about once per 1.4 billion reactor lifetimes. It is therefore concluded that adequate protection against tornado forces and tornado-generated missiles has been provided for the SFP (refer to Section 3.3).

The SFP storage racks are designed for the DE, DDE, and Hosgri Earthquake (HE) with the racks filled with fuel assemblies and as shown in Table 3.2-3. The SFP storage racks are designed to withstand a vertical (uplift) force of 4400 pounds in the unlikely event that an assembly would bind in the rack while being lifted by the spent fuel bridge crane. The design classification of the SFP storage racks, as well as the SFP structure, is discussed in Section 3.2.2.1.1, and the design requirements and acceptance criteria are discussed in Section 3.8.4.2.

Refer to Section 3.8.4.2.6 for a discussion of the evaluation of the SFP structure for postulated interactions of the SFP storage racks with the structure as a result of a seismic event.

9.1.2.3.2 General Design Criterion 3, 1971 – Fire Protection

The SFP storage system area is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.1.2.3.3 General Design Criterion 11, 1967 – Control Room

SFP level monitoring equipment and area radiation monitoring equipment are provided in the control room to support actions to maintain and control the safe operational status of the plant (refer to Sections 9.1.3.3.5 and 11.4.2.1.4.1, respectively).

9.1.2.3.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation is provided to give an alarm in the control room when the water level in the SFP reaches either the high- or low-level alarm setpoint (refer to Section 9.1.3.3.5).

Local instrumentation is provided to measure the temperature of the water in the SFP and provide local indication as well as annunciation in the control room when normal temperatures or rates of temperature change are exceeded (refer to Section 9.1.3.3.5).

Radiation levels are monitored locally in the SFP area (refer to Section 9.1.2.3.5). Signal input is provided to local alarms and control room annunciators, and to the FHBVS to initiate automatic iodine removal mode operation (refer to Section 9.4.4.2).

9.1.2.3.5 General Design Criterion 18, 1967 – Monitoring Fuel and Waste Storage

A controlled and monitored ventilation system removes any gaseous radioactivity from the atmosphere above the SFP and discharges it through the plant vent. This system is described in Section 9.4.4.

Radiation monitors RM-14 and RM-14R continuously monitor the gases discharging through the plant vent—including the gases exhausted from the SFP areas—and alarm when the activity level of the gases reaches a preset limit. The alarm alerts the operators to take appropriate action. Refer to Section 9.4.4.2 for a discussion of FHBVS operation, including operation following detection of high activity levels. The radiation monitors for the fuel storage areas, RM-58 or RM-59, change the FHBVS exhaust mode from normal to automatic iodine removal mode (roughing, high-efficiency particulate air [HEPA], and charcoal filters) if measured area radiation levels increase to preset alarm levels.

An SFP area radiation monitoring system is provided for personnel protection and general surveillance of the SFP area. This system is described in Section 11.4.2.1.4.1. Continuous monitoring and recording readouts and high radiation level alarms in the control room, plus local audible and visual indicators, are provided for use during the movement of irradiated fuel assemblies in the FHB. Refer to Section 9.4.4.2 for discussion concerning FHBVS modes of operation and Section 15.4 for a discussion of radiation monitoring and ventilation during a FHA. SFP radiation, temperature, and level instrumentation are discussed in Section 9.1.2.3.4.

9.1.2.3.6 General Design Criterion 66, 1967 – Prevention of Fuel Storage Criticality

Constraints for storing fuel in the high density SFP racks are identified in the DCPP Technical Specifications. Fresh and burned fuel assembly storage in the SFP is maintained such that any four cells are in one of three configurations allowable by the DCPP Technical Specifications. The three allowable storage configurations specified by the DCPP Technical Specifications are: (a) the all cell, (b) the 2x2 array, and (c) the checkerboard configuration. Spent fuel assemblies satisfying the discharge burnup requirements of Figure 9.1-2A can be stored in the all cell configuration in the SFP. One fuel assembly, with an initial enrichment less than or equal to 4.9 weight percent U-235 or with an initial enrichment less than or equal to 5.0 weight percent U-235 and an IFBA loading equivalent to 16 rods each with 1.5 milligrams B-10 per inch over 120 inches, and three fuel assemblies satisfying the discharge burnup requirements of Figure 9.1-2B can be stored in the 2x2 array configuration. Fresh and spent fuel assemblies not satisfying the initial enrichment, discharge burnup, and IFBA loading requirements for the all cell and 2x2 array configurations must be stored in the checkerboard configuration with water cells or non-fissile material. Figure 9.1-2 shows the arrangement of the rack modules for Unit 1. The Unit 2 SFP is a mirror image of the Unit 1 SFP, reflected around column line $12^{\underline{9}}$ (south end of Unit 1 SFP).

Potential SFP fuel-mishandling accidents and fuel-drop accidents, which result in a reactivity insertion, were evaluated in the SFP criticality analysis. Examples of such accidents include the misplacement of a fresh fuel assembly in place of a burned fuel assembly within a rack module, the misplacement of a fresh fuel assembly outside the rack module, accidental drop of a fresh fuel assembly outside the rack module, and the T-Bone drop of a fresh assembly. The most limiting SFP accident was determined to be the misplacement of a fresh fuel assembly with 4.9 weight percent U-235 outside the rack module such that it is adjacent to a fresh fuel assembly with 4.9 weight percent U-235 within the rack module, resulting in two fresh assemblies adjacent to each other in the SFP. For the most limiting SFP accident, an SFP soluble boron concentration of 806 ppm is required to maintain K_{eff} less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level. The DCPP Technical Specifications establish that the minimum SFP boron concentration is 2000 ppm. This boron concentration is more than sufficient concentration to maintain 5 percent subcriticality margin in the SFP during the most limiting SFP accident. Administrative procedures to ensure the presence of soluble boron in the SFP during fuel handling operations preclude the possibility of the simultaneous occurrence of two independent accident conditions such as a fuel assembly misplacement and loss of soluble boron.

9.1.2.3.7 General Design Criterion 68, 1967 – Fuel and Waste Storage Radiation Shielding

During transfer of spent fuel assemblies, the borated water level in the SFP is maintained to provide at least 9 feet of water above the top of the active portion of the spent fuel assemblies. Normally, the borated water level in the SFP is maintained to provide at least 23 feet of water over the top of the irradiated fuel assemblies seated in the storage racks, as discussed in the Technical Specifications. This ensures sufficient water depth to remove 99.5 percent of all the iodine activity that could be released from a rupture of an irradiated fuel assembly. Gaseous radioactivity above the SFP is thus maintained below 10 CFR Part 20 limits. This water barrier also serves as a radiation shield, limiting the gamma dose rate at the pool surface.

SFP concrete walls have been evaluated for effectiveness of shielding in the SFP area including adjacent corridors and stairways (refer to Section 12.1).

9.1.2.3.8 General Design Criterion 69, 1967 – Protection Against Radioactivity Release from Spent Fuel and Waste Storage

The SFP containment is provided by maintaining at least 23 feet of water over the top of the irradiated fuel assemblies (refer to Section 9.1.2.3.7). The SFP area is enclosed, and is maintained under negative pressure during normal operation. The plant Technical Specifications prescribe operability requirements for FHBVS equipment that are applicable during movement of recently irradiated fuel assemblies, and a surveillance requirement that periodically tests the ability of the FHBVS, while in the post-accident, iodine removal mode of operation, to maintain a negative pressure in the FHB. All ventilation air is passed through HEPA filters prior to being released to the plant vent. In the event of an accident, high activity would be detected by the radiation monitor and the exhaust air would be diverted through charcoal filters (refer to Sections 9.4.4.2 and 11.4.3.15).

Refer to Section 9.1.2.3.7 for a discussion of iodine release attenuation attributable to SFP water depth over the top of the irradiated fuel assemblies.

9.1.2.3.9 Spent Fuel Pool Storage System Safety Function Requirements

(1) Protection from Missiles

There are no credible missiles outside of containment resulting from plant equipment failure that would prevent the PG&E Design Class I SFP from performing its design functions (refer to Section 3.5.3.3). Protection of the SFP from the effects of missiles and protection of PG&E Design Class I SSCs from damage that may result from these events is discussed in Section 3.5.

(2) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect the PG&E Design Class I SFP from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Section 3.6.

9.1.2.3.10 10 CFR 50.68(b) – Criticality Accident Requirements

In accordance with 10 CFR 50.68(b)(1), plant procedures prohibit handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water, with the exception that the U.S. Nuclear Regulatory Commission (NRC) granted an exemption request related to the need to take credit for borated water for the loading, unloading, and handling of the components of the HI-STORM 100 dual-purpose dry cask storage system (refer to Section 9.1.4.3.8).

The requirements in 10 CFR 50.68(b)(2) and 10 CFR 50.68(b)(3) regard storage of fresh fuel in the NFSRs and are therefore not applicable to the SFP.

In accordance with 10 CFR 50.68(b)(4), the high-density SFP storage racks are designed to ensure that, with credit for soluble boron (References 8 and 9) and with fuel of the maximum fuel assembly reactivity, a K_{eff} of less than or equal to 0.95 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with borated water, and a K_{eff} of less than 1.0 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with unborated water.

The associated spent fuel criticality analysis (Reference 24) modeled a full-pool representation of the storage racks and infinite arrays of fuel using the SCALE-PC computer code, which includes the CSAS25 control module and functional modules BONAMI, NITAWL-II and KENO-Va, and employs the 44-Group Evaluated Nuclear Data File Version 5 (ENDF/B-V) neutron cross-section library. SCALE-PC has been validated against 30 critical experiments and the calculations adequately reproduced the data. The DIT computer code was used to generate a set of isotopic concentrations based on ENDF/B-VI. DIT has been benchmarked against Combustion Engineering PWR cores and against other PWR lattice codes, such as CASMO, with very good agreement.

Reference 24 considers the 2x2 array and checkerboard spent fuel configurations and associated fuel type and burnup characteristics specified in the DCPP Technical Specifications. The analysis assumed a fresh fuel assembly, which was a conservative representation of the Westinghouse OFA 17x17 fuel assembly with a nominal enrichment of 4.9 weight percent U-235 and no IFBAs. This fresh assembly conservatively envelopes the characteristics of possible fresh fuel types that may be used. The analysis assumed a burned fuel assembly, which was a conservative representation of a Westinghouse Standard 17x17 fuel assembly. This burned assembly conservatively envelopes the characteristics of burned fuel assemblies stored in the SFP. The analysis evaluated the region of the SFP that does not contain Boraflex panels since the storage requirements for this region are more restrictive and yield more conservative reactivity results than the region containing Boraflex panels. Therefore, the analysis does not credit the negative reactivity associated with the Boraflex panels.

Reference 24 considered biases and uncertainties such that the K_{eff} value was determined at a 95 percent probability, 95 percent confidence level. The biases considered included a KENO-Va computer code methodology bias and a reactivity bias to account for a range of SFP water temperature. The uncertainties considered included those due to fuel assembly manufacturing tolerances, rack fabrication tolerances, KENO-Va computer code methodology, fuel assembly reactivity, and absolute fuel assembly burnup.

Reference 24 assumed a core moderator average temperature of 579.95°F. Higher moderator temperature affects analysis results for discharged spent fuel in a non-conservative direction (i.e., reduces margin). Since moderator temperature in the core can potentially reach 582.3°F, an evaluation was performed to assess the impact of this higher temperature on the analysis (Reference 25). The evaluation concluded that adequate margin exists such that the potentially higher moderator temperature is acceptable and can be accommodated in the existing SFP criticality analysis.

Reference 24 does not consider the "all cell" storage configuration in the SFP (i.e., all cells filled and all fuel assemblies with discharge burnup in the acceptable area of Technical Specification Figure 3.7.17-2). A Holtec analysis (Reference 26) is the analysis of record (AOR) for that configuration. The core moderator average temperature value assumed for the Holtec AOR is 591.5°F. This value bounds the maximum potential core average temperature of 582.3°F. Therefore, the AOR for the "all cell" storage configuration is bounding for the maximum expected core moderator temperature.

For normal conditions (no SFP fuel mishandling or fuel drop accident), the SFP criticality analysis determined that a K_{eff} of less than 1.0 is maintained, at a 95 percent probability, 95 percent confidence level, if the SFP storage racks are flooded with unborated water.

An SFP boron dilution analysis was performed to evaluate the time and water volumes required to dilute the SFP from the DCPP Technical Specification required minimum boron concentration of 2000 ppm to approximately 800 ppm. The 800 ppm endpoint was utilized to ensure that the K_{eff} of the SFP storage racks would remain less than or equal to 0.95.

A large volume of pure water (approximately 339,000 gallons) is necessary to dilute the SFP from 2000 ppm to 800 ppm. Dilution sources available which exceed 339,000 gallons (primary water makeup system, makeup water system [MWS], and fire protection system) were evaluated against the calculated dilution volumes to determine the potential of an SFP-dilution event. The dilution from seismic events or random pipe breaks is bounded by the primary water makeup system flow. Dilution due to the drain system was not evaluated since backflow through the system is not considered credible. Also, the SFP demineralizer was not evaluated since it cannot provide sufficient dilution.

The boron dilution analysis demonstrates that adequate time is available to identify and mitigate the dilution event before the K_{eff} of the SFP storage racks would exceed 0.95. A dilution event large enough to result in a significant reduction in the SFP boron concentration would involve the transfer of a large quantity of water from a dilution source and a significant increase in SFP level, which would ultimately overflow the pool. Such a large water volume turnover, and overflow of the SFP, would be readily detected and terminated by plant personnel. In addition, because of the large quantities of water required and the low dilution flow rates available, any significant dilution of the SFP boron concentration would only occur over a long period of time (hours to days). Detection of an SFP boron dilution via SFP level alarms, visual inspection during normal operator rounds, significant changes in SFP boron concentration, or significant changes in the unborated water source volume, would be expected before a dilution event sufficient to increase K_{eff} above 0.95 could occur.

The results of the boron dilution analysis concluded that an unplanned or inadvertent event that would result in the dilution of the SFP boron concentration from 2000 ppm to 800 ppm is not a credible event. However, even if the SFP were diluted to zero ppm boron, which would take significantly more water than evaluated in the dilution analysis, the SFP would remain subcritical, and the health and safety of the public would be protected. Sampling of the SFP boron concentration is required by the DCPP Technical Specifications on a 7-day frequency, which provides adequate assurance that smaller and less readily identifiable boron concentration reductions are not taking place.

Controls over special nuclear material are maintained to prevent a criticality accident. In addition to the controls for both new (fresh) and spent (burned) fuel, as required by 10 CFR 50.68(b)(5) the quantity and forms of special nuclear material other than nuclear fuel that is stored onsite in any given area are less than the quantities necessary for a critical mass. Special nuclear materials are required to be stored in inventory control areas except when in transit.

In accordance with 10 CFR 50.68(b)(6), radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions (refer to Sections 9.1.2.3.4 and 9.1.2.3.5).

Refer to Section 9.1.1.3.6 for information on implementation of 10 CFR 50.68(b)(7) regarding maximum nominal U-235 enrichment of new fuel assemblies that may be stored in the SFP.

As discussed in Section 9.1.1.3.8, implementation of 10 CFR 50.68(b)(8) was completed with the submittal of Revision 15 of the UFSAR.

9.1.2.3.11 Safety Guide 13, March 1971 – Fuel Storage Facility Design Basis

Regulatory Position 1:

The SFP is designed to accommodate both new and spent fuel assemblies in a subcritical array such that a $k_{eff} < 1.0$ is maintained if flooded with unborated water. It is constructed of reinforced concrete as part of the auxiliary building structure. The design is described in Section 3.8.2.3. The entire structure and the SFP storage racks have been designed in accordance with PG&E Design Class I seismic requirements. Criteria set by Safety Guide 13, March 1971 (Reference 3), have been followed.

The SFP storage racks are designed in accordance with Safety Guide 13, March 1971, and the ASME BPVC Section III-1983 (Subsection NF).

Regulatory Position 2:

Refer to Section 9.1.2.3.1 for a discussion of the DCPP environmental protections, such as protection from cyclonic winds and external missiles, for the PG&E Design Class I SFP. Refer to Section 3.3 for a broad discussion of wind and tornado loadings on PG&E Design Class I structures.

Regulatory Position 5.b:

PG&E has evaluated the drop of a loaded transfer cask from highest point in the lift to the bottom of the cask recess area in the SFP. The postulated drop consists of 4.67 feet in air followed by 42.83 feet in water. Analysis demonstrates the adequacy of the affected structures during the postulated drop, demonstrating that the drop will not cause (1) loss of building structural function; (2) damage to the SFP resulting in loss of SFP water; or (3) unacceptable damage to other systems or equipment. The SFP stainless steel liner may be damaged in this drop; however, the structural integrity of the concrete forming the SFP is maintained, preventing any uncontrolled leakage.

Refer to Section 9.1.4.3.9 for a discussion of fuel handling area crane design provisions that prevent the movement of heavy loads over the area of the pool that can contain spent fuel.

In addition, protection of nuclear fuel assemblies from overhead load handling is a key element of the Control of Heavy Loads Program described in Section 9.1.4.3.10.

Regulatory Position 7:

Refer to Section 9.1.2.3.4 for a discussion of the alarms provided both locally and in the control room if the water level in the SFP falls below a predetermined level (control room alarm) or if high local radiation levels are experienced (local and control room alarm).

9.1.2.4 Tests and Inspections

After erection of the SFP storage racks, tests were conducted with a dummy fuel assembly by passing it into and out of each storage position to ensure that binding would not occur.

9.1.2.5 Instrumentation Applications

Instrumentation applications are described in Section 9.1.2.3.4.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

The piping, valves, pumps, and heat exchanger used to cool the SFP are PG&E Design Class I. Additionally, the piping and components used to transfer water to and from the RWST are PG&E Design Class I. Portions of the SFP cleanup system are PG&E Design Class I including the RWP filter, SFP resin trap filter, SFP demineralizer, and RWP pump. The piping and components used to transport water to the SFP filter (including the filter itself) are PG&E Design Class II. Refer to Table 3.2-3 for a listing of PG&E Design Class II equipment that is included in the SFP cooling and cleanup system.

The SFP skimmer system, including piping, valves, pumps, strainers, and filters, is PG&E Design Class III.

Section 9.1.3.3.9 provides discussion on PG&E Design Class I systems that are available to provide makeup water to the SFP.

Each unit has a completely independent SFP cooling and cleanup system.

The SFP cooling and cleanup system design parameters are given in Table 9.1-1.

9.1.3.1 Design Bases

9.1.3.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portions of the SFP cooling and cleanup system are designed to withstand the effects of, or are protected against, natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

Tornado-induced FHB structural damage does not impair the ability of the plant to achieve safe shutdown.

9.1.3.1.2 General Design Criterion 3, 1971 – Fire Protection

The PG&E Design Class I portions of the SFP cooling and cleanup system are designed and located to minimize, consistent with other requirements, the probability and effect of fires and explosions.

9.1.3.1.3 General Design Criterion 11, 1967 – Control Room

The PG&E Design Class I portions of the SFP cooling and cleanup system are designed to or contain instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room.

9.1.3.1.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the PG&E Design Class I portions of the SFP cooling and cleanup system variables within prescribed operating ranges.

9.1.3.1.5 General Design Criterion 18, 1967 – Monitoring Fuel and Waste Storage

The PG&E Design Class I portions of the SFP cooling and cleanup system are equipped with adequate instrumentation to identify conditions that contribute to a loss of continuity in decay heat removal and to radiation exposures.

9.1.3.1.6 General Design Criterion 67, 1967 – Fuel and Waste Storage Decay Heat

The PG&E Design Class I portions of the SFP cooling and cleanup system provide a reliable and adequate decay heat removal system to prevent damage to the fuel in the SFP that could result in radioactivity release to plant operating areas or the public environs.

9.1.3.1.7 Spent Fuel Pool Cooling and Cleanup System Safety Function Requirements

(1) <u>Missile Protection</u>

The PG&E Design Class I SFP cooling and cleanup system components are designed to be protected against internal missiles generated outside containment and dynamic effects that might result from plant equipment failure.

(2) Water Purification

The SFP cooling and cleanup system purifies and demineralizes SFP water to maintain SFP water quality to ensure access to the SFP storage racks for fuel handling and maintain optical clarity of the SFP water.

9.1.3.1.8 10 CFR 50.55a(g) – Inservice Inspection Requirements

The SFP cooling and cleanup system ASME Code components are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.1.3.1.9 Safety Guide 13, March 1971 –Fuel Storage Facility Design Basis

The SFP cooling and cleanup system is designed and constructed in accordance with Safety Guide 13, March 1971.

9.1.3.1.10 Generic Letter 96-04, June 1996 – Boraflex Degradation in Spent Fuel Pool Storage

Silica levels in the SFP are monitored using DCPP procedures for trending purposes due to the possible interaction between SFP water and reactor coolant system (RCS) water during refueling outages.

9.1.3.2 System Description

The PG&E Design Class I portions of the SFP cooling and cleanup system maintain a water inventory in the SFP sufficient to keep spent fuel immersed at all times and provide a highly reliable pumped-fluid system to transfer decay heat from the SFP to the component cooling water (CCW) system via the SFP heat exchanger.

The SFP cooling and cleanup system, shown in Figure 3.2-13, removes decay heat from fuel stored in the SFP. Spent fuel is placed in the SFP during the refueling sequence and stored there until it is shipped offsite or loaded into a fuel transfer cask and transported to the Diablo Canyon ISFSI. Refer to Reference 12 for information regarding the Diablo Canyon ISFSI. Heat is transferred through the SFP heat exchanger to the CCW system.

When the SFP cooling and cleanup system is in operation, water flows from the SFP to the SFP pump suction, is pumped through the tube side of the heat exchanger, and is returned to the SFP. The suction line, which is protected by a strainer, is located at an elevation 4 feet below the normal SFP water level, while the return line contains an antisiphon hole near the surface of the water to prevent gravity drainage of the SFP.

While the heat removal operation is in process, a portion of the SFP water may be diverted away from the heat exchanger through the RWP filter, spent fuel pit demineralizer, spent fuel pit resin trap filter, and the spent fuel pit filter to maintain water clarity and purity. A resin trap and check valve, located upstream of the demineralizer, prevent backflushing demineralizer resins to the SFP. Transfer canal water may also be circulated through the same demineralizer and filter by opening the gate between the canal and the SFP. This purification loop removes fission products and other contaminants, which could be introduced if a fuel assembly with defective cladding is transferred to the SFP.

The RWP system can be aligned to recirculate the contents of the RWST. The RWP system is placed into service by manual operation of isolation valves and manual RWP pump start. This alignment enables tank mixing and cleanup of the RWST contents via the RWP filter, demineralizer, and resin trap filter. Processing the RWST contents through the RWP system enables the removal of radiological impurities to ensure RWST activities (refer to Table 12.1-13) and radiation exposure rates (refer to Table 12.1-14) are within 10 CFR Part 20 limits and are as low as is reasonably achievable (ALARA).

The RWP system filters and demineralizes the RWST water in order to maintain water quality and clarity for fuel transfer and inspection purposes. Refueling water clarity is both a personnel and equipment safety and radiation ALARA consideration. Also, the RWP system may be used to filter the contents of the RWST prior to employing a reverse osmosis system (ROS). The ROS is a temporary system, connected directly to the RWST, which may be used to reduce silica concentrations in the RWST. Design and administrative controls ensure minimum required RWST volume and boron concentrations are maintained throughout ROS operation.

During refueling outages, connections are provided such that the refueling water may be pumped from either the RWST or the refueling cavity, through the filter, demineralizer, and resin trap and discharged to either the refueling cavity or the RWST. In addition to this flowpath, it is possible to manually align the SFP cleanup system with the RWP system to clean the refueling canal water during fuel movement. The RWP pump may also be utilized to pump down the refueling canal by pumping water to the liquid hold-up tanks (LHUTs) through the RWP filter. To further assist in maintaining SFP water clarity, the water surface is cleaned by a skimmer loop. Water is removed from the surface by the skimmers, pumped through a strainer and filter, and returned to the SFP surface at three locations remote from the skimmers.

Refer to Section 9.1.2.3.6 for discussion on the required boron concentration in the SFP. Additional borated water is supplied from the RWST via the RWP system, or from other sources within the CVCS, to maintain the required boron concentration in the SFP.

A gate is installed between the SFP and the transfer canal so that the transfer canal may be drained to allow maintenance of the fuel transfer equipment. The water in the transfer canal is first pumped, using a portable pump, into the SFP and then is transferred to a holdup tank in the CVCS by the SFP pump. When needed for refueling operations, water is returned directly to the transfer canal by the holdup tank recirculation pump.

In the event of high level in the SFP, water can be removed via the SFP pump and pumped to the LHUTs. Water can also be removed by the RWP pump delivering to the RWST. In the event of low water level, makeup water can be transferred to the SFP from either PG&E Design Class I or PG&E Design Class II sources as described in Section 9.1.3.3.9.

9.1.3.2.1 Component Description

SFP cooling and cleanup system codes and classifications are given in Table 3.2-3. System design and operating parameters are given in Table 9.1-2.

9.1.3.2.1.1 Spent Fuel Pool Pump

The SFP pumps are horizontal, centrifugal units, with all wetted surfaces being stainless steel or an equivalent corrosion-resistant material. The pumps are controlled manually from a local station. There are no Class 1E electrical loads in the SFP system; however, the SFP cooling pumps are powered from a Class 1E source. For Modes 5, 6, and no mode operation during electrical bus outages and maintenance periods, the standby/redundant pump may be temporarily aligned to an alternate Class 1E source, using installed transfer switches, until its primary Class 1E power supply is returned to service. If connection to the alternate Class 1E bus via the transfer switch is necessary in Modes 1 through 4, Engineering will be required to evaluate the acceptability of this configuration on a case by case basis prior to use of the transfer switch.

9.1.3.2.1.2 Spent Fuel Pool Skimmer Pump

The SFP skimmer pump is a horizontal centrifugal unit, with all wetted surfaces being stainless steel or an equivalent corrosion-resistant material. The pump is controlled manually from a local station.

9.1.3.2.1.3 Refueling Water Purification Pump

The RWP pump is a horizontal centrifugal unit, with all wetted surfaces being stainless steel or an equivalent corrosion-resistant material. The pump is operated manually from a local station.

9.1.3.2.1.4 Spent Fuel Pool Heat Exchanger

The spent fuel heat exchanger is of the shell and U-tube type with the tubes welded to the tubesheet. CCW circulates through the shell, and SFP water circulates through the tubes. Construction is carbon steel on the shell-side and stainless steel on the tube side.

9.1.3.2.1.5 Spent Fuel Pool Demineralizer

The SFP demineralizer is a flushable, mixed bed demineralizer. The demineralizer is designed to provide adequate SFP water purity and limit the dose rate at the surface of the SFP. Submersible demineralizer vessels may be used in the SFP or reactor cavity to perform or augment water purification.

9.1.3.2.1.6 Spent Fuel Pool Resin Trap Filter

The SFP resin trap filter is designed to prevent resin beads from entering the SFP and connected systems. It is designed to remove particles 5 microns or greater.

9.1.3.2.1.7 Spent Fuel Pool Filter

The SFP filter is designed to improve the SFP water clarity by removing particles 5 microns or greater. Underwater vacuum and/or underwater vacuums equipped with surface skimmer filter units may be used to augment SFP or reactor cavity water treatment.

9.1.3.2.1.8 Spent Fuel Pool Skimmer Filter

The SFP skimmer filter is used to remove particles that are not removed by the strainer. It is designed to remove particles 5 microns or greater.

9.1.3.2.1.9 Refueling Water Purification Filter

The RWP filter is designed to improve the clarity of the refueling water in the refueling canal or in the RWST by removing particles 5 microns or greater. The RWP filter also filters the SFP contents and functions as a prefilter to the SFP demineralizer.

9.1.3.2.1.10 Spent Fuel Pool Strainer

A strainer is located within the SFP on the pump suction enclosure for removal of relatively large particles, which might otherwise clog the SFP demineralizer or damage the SFP pump. It is a slotted screen design with stainless steel construction.

9.1.3.2.1.11 Spent Fuel Pool Skimmer Strainer

The SFP skimmer strainer is designed to remove debris from the skimmer process flow. It is an in-line basket strainer of stainless steel construction.

9.1.3.2.1.12 Spent Fuel Pool Skimmers

Two SFP skimmers are provided to remove water from the surface of the SFP. The skimmer heads are manually positioned to take water from near the SFP surface. The skimmer, pipe, and supports are of stainless steel construction.

9.1.3.2.2 Valves

Manual stop valves are used to isolate equipment, and manual throttle valves provide flow control. Valves in contact with SFP water are austenitic stainless steel or equivalent corrosion-resistant material.

9.1.3.2.3 Piping

All piping in contact with SFP water is austenitic stainless steel. The piping is welded except where flanged connections are used to facilitate maintenance.

9.1.3.3 Safety Evaluation

9.1.3.3.1 General Design Criterion 2, 1967 – Performance Standards

The FHB and auxiliary building, which contain the spent fuel cooling and cleanup systems are PG&E Design Class I structures (refer to Section 3.8). The auxiliary building is designed to withstand the effects of winds and tornadoes (refer to Section 3.3) and external missiles (refer to Section 3.5). The basic structure of the FHB is tornado resistant, with damage to non-structural building components. Damage to these components does not adversely impact the plant's ability to achieve safe shutdown (refer to Section 3.3). The auxiliary and FHBs are designed to withstand the effect of floods (refer to Section 3.4), earthquakes (refer to Section 3.7), and to protect the spent fuel cooling and cleanup system, ensuring its design function will be performed.

Protection of fuel assemblies from tornadoes and external missiles is discussed in Section 9.1.2.3.1.

A piping design analysis, including DE, DDE, and HE seismic loads, was performed to ensure that the cooling loop conforms to PG&E Design Class I piping criteria as indicated in Table 3.2-3. The failure or malfunction of any of the spent fuel cooling and cleanup system components, including failures resulting from the DE, DDE, or a HE will not cause the fuel to be uncovered.

9.1.3.3.2 General Design Criterion 3, 1971 – Fire Protection

The SFP area is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.1.3.3.3 General Design Criterion 11, 1967 – Control Room

Annunciation in the control room is provided when normal SFP temperatures are exceeded. Additionally, instrumentation is provided to give an alarm in the control room when the water level in the SFP reaches either the high or low level setpoint. These instruments support actions to observe the safe operational status of the SFP cooling and cleanup system from the control room.

9.1.3.3.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation is provided for the SFP cooling and cleanup system as follows:

- Local instrumentation is provided to give indication of the temperature of the SFP water as it leaves the SFP heat exchanger. Refer to Section 9.1.3.3.5 for temperature instrumentation provided for the water in the SFP.

- Local instrumentation is provided to measure and give indication of pressures across the SFP and RWP pumps. Instrumentation is also provided to measure pressure differential on the SFP demineralizer, SFP resin trap filter and RWP filter.

- Local instrumentation is provided to measure and give indication of flows in the outlet line of the SFP filter and the inlet line to the SFP demineralizer.

9.1.3.3.5 General Design Criterion 18, 1967 – Monitoring Fuel and Waste Storage

Local instrumentation is provided to measure the temperature of the water in the SFP and give local indication as well as annunciation in the control room when normal temperatures are exceeded.

As indicated in Section 9.1.3.3.3, instrumentation is provided to give an alarm in the control room when the water level in the SFP reaches either the high or low level setpoint.

The exposure rate at the SFP surface is routinely monitored with radiation surveys and monitoring equipment (refer to Section 12.3). The major contributor to the surface dose is the radioactivity within the SFP water and not the spent fuel assemblies stored in the SFP. The SFP demineralizer will be used as necessary to maintain radiation exposures ALARA.

9.1.3.3.6 General Design Criterion 67, 1967 – Fuel and Waste Storage Decay Heat

The SFP cooling system is designed to remove that amount of decay heat that is produced by spent fuel assemblies that are stored in the SFP following a refueling. A cask pit rack was installed in the SFP of each unit during Cycle 14, prior to the 14th refueling outage. During Cycle 15, the cask pit rack was removed from Unit 1. The cask pit rack in Unit 2 was removed during Cycle 16.

PG&E has updated its SFP thermal-hydraulic analyses as part of the cask pit rack project (Reference 15). These updated analyses were performed using more recent analytical methods that have been previously accepted by the NRC. These analytical methods and the associated full core and emergency offload scenarios discussed below will bound both the installation of the cask pit rack and future DCPP SFP fuel storage requirements once the cask pit rack has been removed. These new analyses will serve as the new licensing basis of record for future spent fuel storage requirements, including the temporary supplemental spent fuel storage capacity provided by the cask pit rack.

The cask pit thermal-hydraulic analyses are based on the evaluation of three offload scenarios that bound the past and future operating practices at DCPP. For each scenario, the transient and steady state decay heat loads were combined to provide a total decay heat load on the SFP cooling system (Reference 15).

The partial core offload scenario assumes a discharge of 96 fuel assemblies during the 15th refueling outage. All of the 96 fuel assemblies offloaded are conservatively assumed to have a burnup of 52,000 MWD/MTU.

The full core offload scenario assumes a discharge of 193 fuel assemblies during the 15th refueling outage. The 193 offloaded assemblies are separated into two distinct groups; 101 assemblies with 52,000 MWD/MTU burnup and 92 assemblies with 25,000 MWD/MTU burnup.

The emergency full core offload scenario assumes that the 15th refueling outage is completed in 30 days, leaving 104 assemblies in the SFP at restart. After 36 days of operation at 100 percent power in Cycle 16, an emergency full core offload is performed, completely filling all available storage locations. The 193 assemblies are separated into two distinct groups: 113 assemblies with 40,000 MWD/MTU burnup and 80 assemblies with 3,000 MWD/MTU burnup.

All of these scenarios have been evaluated with a base decay heat load contribution from previously discharged fuel assemblies using actual operational data for operating Cycles 1 through 11. The contribution to the base decay heat load from fuel that will be discharged in Cycles 12 through 14 is based on an assumed discharge of 104 assemblies each Cycle using bounding assumptions on fuel assembly burnup and operating power. Cycle lengths assumed for Cycles 12 through 14 are assumed to be 18 months, which conservatively minimizes the decay time and maximizes the base decay heat load. All three of these scenarios assume a core offload rate of four assemblies per hour, starting 100 hours after reactor shutdown, and other appropriately conservative fuel assembly discharge and burnup assumptions.

Conservative values for pump flow and heat exchanger performance were selected to provide bounding calculations for the peak SFP bulk temperature. The thermal performance of the heat exchangers was determined with all heat transfer surfaces assumed to be fouled to their design basis maximum levels and also included an allowance for five percent tube plugging. CCW supplied to the heat exchanger was assumed to be 75°F at a flow rate of 3400 gpm which is bounded by SFP cooling system design parameters. Plant procedures are currently in place to limit the peak

SFP temperature to within 140°F. The procedural controls currently suspend offload activities at a SFP temperature of 125°F to maintain peak SFP bulk temperatures less than 140°F. Past operating experience at DCPP has shown that peak SFP temperatures are less than 115°F during a typical full core offload.

Due to the many variables that can have an impact on peak SFP temperature, DCPP may elect to use a cycle specific offload analysis, following the requirements of Section 3.1 of Attachment 2 to Matrix 5 of Section 2.1 of RS-001, December 2003, in lieu of the operating restrictions of the bounding thermal analyses described above. Consideration will be given to the actual core power history, scheduled offload start time, offload rates, actual CCW temperature, and actual CCW and SFP cooling water flow rates to the SFP heat exchanger in the establishment of the specific control values. If DCPP elects to use a cycle specific analysis, plant procedures will require that core offload be suspended at a temperature, which would ensure that the 140°F limit is not exceeded.

Results of the thermal-hydraulic analyses are described below.

The partial core offload analysis resulted in a maximum SFP bulk temperature of 127°F. The full core offload analysis resulted in a maximum SFP bulk temperature of 157°F. The emergency core offload analysis resulted in a peak bulk temperature of less than 162°F.

The time-to-boil evaluation assumed that forced cooling was lost the moment the peak SFP bulk temperature for each case was reached. The SFP time to boil and corresponding maximum boil-off rates were then determined.

For the worst-case scenario, the emergency core offload, the calculated time-to-boil was determined to be 3.76 hours after a loss of forced cooling at the peak SFP bulk temperature. The corresponding maximum boil-off rate for this condition was approximately 87 gpm.

Given the conservatisms incorporated into the calculations, actual times-to-boil will be higher than these calculated values and actual boil-off rates will be lower than calculated. Based on the time-to-boil, plant personnel will have sufficient time to identify elevated SFP temperatures and adequate time to provide makeup to the SFP, if needed.

Local temperature analyses were also performed to determine maximum local water and fuel cladding temperatures. The worst case peak local water temperature is 188 °F and below the local saturation temperature (240 °F) at the depth of the cask pit. The results also demonstrate that the peak fuel cladding temperature of 213 °F for the hottest fuel assembly is below the local saturation temperature, and the critical heat flux for departure from nucleate boiling is not exceeded. Therefore no bulk boiling will occur in the SFP, and the local water and fuel temperatures are acceptable.

The SFP cooling and cleanup system has no emergency function during an accident. This manually controlled system may be shut down for limited periods of time for maintenance or replacement of components. Redundancy of the SFP cooling and cleanup system components is not required because of the large heat capacity of the SFP and the slow heatup rate. In the unlikely event that the SFP pump should fail, the backup pump can provide circulation of the SFP water through the SFP heat exchanger. If a failure should occur that would prevent the use of the SFP heat exchanger for cooling the SFP water (e.g., severance of the piping which constitutes the cooling recirculation path), natural surface cooling would maintain the water temperature at or below the boiling point. A PG&E Design Class I backup makeup water source is provided to ensure that the water level in the SFP can be maintained.

9.1.3.3.7 Spent Fuel Pool Cooling and Cleanup System Safety Function Requirements

(1) <u>Missile Protection</u>

There are no credible missiles outside of containment resulting from plant equipment failure that would prevent the PG&E Design Class I portions of the SFP cooling and cleanup system from performing their design functions (refer to Section 3.5.1.2). Dynamic effects as a result of plant equipment failure will not prevent the PG&E Design Class I portions of the SFP cooling and cleanup system from performing their design functions.

(2) <u>Water Purification</u>

The system's demineralizer and filters are designed to provide adequate purification of the SFP contents to permit access to the SFP storage area and maintain optical clarity of the SFP water. The optical clarity of the SFP water surface is maintained by use of the system's skimmers, strainer, and skimmer filter. Refer to Section 9.1.3.2 for a description of water purification equipment used to maintain optical clarity of the SFP water. The purification loop is capable of removing fission products and other contaminants from the SFP water, including small quantities of fission products from leaking spent fuel assemblies.

The SFP demineralizer and filter flowpath bypasses the SFP heat exchanger. The demineralizer and filter may be brought into or out of service by manual operation of isolation valves. No other operator action is required. The bypass flowpath can be through the filter only, or through the filter and demineralizer in series, as shown in Figure 3.2-13. The piping configuration allows an alternate flowpath that utilizes the RWP filter upstream of the SFP demineralizer to allow optimization of filter and demineralizer operation. The demineralizer may also be used in conjunction with the RWP pump, filter, and resin trap to clean and purify the refueling water while SFP heat removal operations proceed.

A significant reduction of radioactive effluents is achieved through filtration and ion exchange, and by recycling refueling water as opposed to disposing of and making new refueling water. In addition, by concentrating impurities and radioactive particles on filter medium and demineralizer resin, which are more easily shielded, radiation levels are maintained ALARA.

Only a very small amount of water is interchanged between the refueling canal and the SFP while fuel assemblies are transferred in the refueling process. Whenever a leaking fuel assembly is transferred from the fuel transfer canal to the SFP, a small quantity of fission products may enter the spent fuel cooling water. The purification loop removes fission products and other contaminants from the water, thereby maintaining radioactivity concentration in the SFP water ALARA.

The SFP water meets the following water quality requirements:

Boric acid, ppm as boron, minimum	2000
pH at 77°F, minimum	4.1
Chloride, ppm maximum	0.15
Fluoride, ppm maximum	0.15

Boron concentrations in the SFP water are maintained as discussed in the Technical Specifications, and the pH of SFP water is controlled to prevent separation of top nozzles from a fuel assembly as a result of intergranular stress corrosion cracking.

The inservice inspection (ISI) requirements for the SFP cooling and cleanup system are contained in the ISI Program Plan.

9.1.3.3.8 10 CFR 50.55a(g) – Inservice Inspection Requirements

The ISI requirements for the SFP cooling and cleanup system are contained in the ISI Program Plan.

9.1.3.3.9 Safety Guide 13, March 1971 – Fuel Storage Facility Design Basis

<u>Regulatory Position 1:</u>

The spent fuel storage facilities, including structures and equipment are PG&E Design Class I. Refer to Section 3.2 for a discussion of this issue.

As discussed in Section 9.1.3.3.1, the FHB and auxiliary building, which contain the spent fuel cooling and cleanup systems are PG&E Design Class I (refer to Section 3.8). A piping design analysis, including seismic loads, was performed to ensure that the SFP cooling loop conforms to PG&E Design Class I piping criteria and the SFP cooling pumps are powered from a Class 1E source.

Regulatory Position 2:

Analysis of potential water loss from the SFP shows that the water cover remaining over the fuel provides adequate protection against both fuel damage and pool liner penetration from tornado missiles (refer to Section 3.3.2.5.2.3).

Section 9.1.3.3.1 provides a discussion of the DCPP environmental protections for the SFP cooling and cleanup system.

Regulatory Position 6:

The most serious failure of this system would be complete loss of water in the SFP. To protect against this possibility, the SFP cooling suction connection enters near the normal water level so that the SFP cannot be gravity-drained. The cooling water return line contains an antisiphon hole to prevent the possibility of gravity draining the SFP.

System piping is arranged so that failure of any pipeline cannot inadvertently drain the SFP below the water level required for radiation shielding. This level is maintained by:

(1) SFP suction piping located 20 feet above the top of the fuel assemblies, and

(2) a siphon breaker on the cooling pipe's return line into the SFP.

This design ensures greater than ten feet of water exists over the top of the fuel assemblies should inadvertent drainage occur.

Normal SFP water levels are maintained a minimum of 23 feet over the top of irradiated fuel assemblies seated in the storage racks as discussed in Section 9.1.2.2.

Regulatory Position 7:

Section 9.1.3.3.3 provides a discussion of DCPP capabilities for monitoring water level in the SFP.

Radiation monitoring of the SFP area is discussed in 9.1.1.1.5 and 9.1.2.3.5.

Regulatory Position 8:

Demineralized makeup water can be added directly to the SFP by a PG&E Design Class I source. Water from the condensate storage tank (CST) is pumped to the SFP using the makeup water transfer pumps (refer to Section 9.2.6 and Table 9.2-9) and appropriate interconnecting piping and valves. This source has the capability of providing up to 200 gpm of demineralized water, if required. The above tank, pumps, piping, and valves are designed in accordance with Safety Guide 13, March 1971. The transfer tank is another PG&E Design Class I source of SFP makeup, and water can be pumped to the SFP by the makeup water transfer pumps. However, the flowpath from the transfer tank is not completely PG&E Design Class I. In addition to the above source of makeup water, the PG&E Design Class I fire water tank could provide makeup from local hose reels.

9.1.3.3.10 Generic Letter 96-04, June 1996 – Boraflex Degradation in Spent Fuel Pool Storage

Samples are taken from the SFP monthly and are tested for silica in accordance with the response to Generic Letter 96-04, June 1996 and as modified in Reference 27. This testing was originally established to trend Boraflex degradation in the SFP. The current method of crediting soluble boron to control reactivity in the SFP precludes the need to monitor Boraflex degradation. However, due to the fact that it is desirable to minimize silica content in water that could interact with RCS water, the silica levels in the RWST and SFP water continue to be monitored. As discussed in Section 9.1.3.2, a ROS may be used to reduce the silica concentration in the RWST to levels that are compatible for the RCS.

9.1.3.4 Inspection and Testing Requirements

Active components of the SFP cooling and cleanup system are either in continuous or intermittent use during normal system operation. Periodic visual inspection and preventive maintenance are conducted using normal industry practice. Refer to Section 9.1.3.3.8 for SFP cooling and cleanup system ISI requirements.

9.1.3.5 Instrumentation Applications

The instrumentation provided for the SFP cooling and cleanup system is discussed in Sections 9.1.3.3.4 and 9.1.3.3.5. Alarms and indications are provided as noted.

Local instrumentation is provided to measure and give indication of pressures across the skimmer pumps. Instrumentation is also provided to measure pressure differential on the SFP filter and SFP skimmer filter and strainer.

9.1.4 FUEL HANDLING SYSTEM

The FHS consists of equipment and structures utilized for handling new and spent fuel assemblies in a safe manner during refueling and fuel transfer and cask loading operations.

The FHS makes use of components and structures that fall within PG&E Design Class I, PG&E Design Class II, and PG&E Design Class III classifications. The FHS components are PG&E Design Class I with the following exceptions:

(1) The fuel transfer system used to move fuel assemblies between containment and the FHB uses PG&E Design Class III components, with

the exception of the fuel transfer tube and quick opening hatch, which are PG&E Design Class I.

(2) The tools used for fuel handling activities are PG&E Design Class II and PG&E Design Class III.

(3) The fuel handling area movable partition walls are PG&E Design Class II.

(4) The seal ring and assembly between the reactor vessel (RV) and the refueling cavity are PG&E Design Class II.

(5) The components used to store damaged fuel rods, debris captured from the fuel nozzles, and spent fuel storage rack poison specimens are PG&E Design Class II.

(6) The containment structure polar crane and portions of the fuel handling area crane are PG&E Design Class I. Note that the cables and hooks for the containment structure polar crane are PG&E Design Class II.

(7) All other cranes and hoists used in the FHS are PG&E Design Class II and PG&E Design Class III.

9.1.4.1 Design Bases

9.1.4.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the FHS is designed to withstand the effects of, or is protected against, natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.1.4.1.2 General Design Criterion 4, 1967 – Sharing of Systems

The FHS and components are not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.1.4.1.3 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided to monitor and maintain FHS variables within prescribed operating ranges.

9.1.4.1.4 General Design Criterion 49, 1967 – Containment Design Basis

The FHS is designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident (LOCA), including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems (ECCSs).

9.1.4.1.5 General Design Criterion 66, 1967 – Prevention of Fuel Storage Criticality

Equipment used to store spent fuel assemblies is designed to maintain the spent fuel assemblies in a subcritical state, even under postulated seismic conditions.

9.1.4.1.6 General Design Criterion 68, 1967 – Fuel and Waste Storage Radiation Shielding

Shielding for radiation protection is provided in the design of the spent fuel facilities as required to meet the requirements of 10 CFR Part 20.

9.1.4.1.7 10 CFR 50.55a(g) – Inservice Inspection Requirements

ASME code components within the PG&E Design Class I portion of the FHS are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.1.4.1.8 10 CFR 50.68(b) – Criticality Accident Requirements

FHS components that store and transport fuel assemblies are designed in accordance with 10 CFR 50.68, with a noted exemption.

9.1.4.1.9 Safety Guide 13, March 1971 – Fuel Storage Facility Design Basis

<u>Regulatory Position 1:</u>

The FHS is designed to Category I seismic requirements.

<u>Regulatory Position 2:</u>

The PG&E Design Class I portion of the FHS is designed to prevent tornadoes and tornado-borne missiles from causing significant loss of watertight integrity of the SFP and to prevent tornado-borne missiles from contacting fuel within the SFP.

<u>Regulatory Position 3:</u>

Interlocks are provided to prevent cranes from passing over the SFP when fuel handling is not in progress.

Regulatory Position 5(b):

The SFP is designed to withstand, without leakage which could uncover the fuel, the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted.

9.1.4.1.10 NUREG-0612, July 1980 – Control of Heavy Loads at Nuclear Power Plants

The FHS is designed and constructed to meet the requirements of NUREG-0612, July 1980, minimizing the probability of dropping a heavy load on spent fuel or equipment necessary for safe shutdown of the plant.

9.1.4.2 System Description

The FHS equipment needed for the refueling of the reactor core consists of cranes, lifting and handling devices including tools, and a fuel transfer system.

The reactor is refueled with fuel handling equipment designed to handle the spent fuel under water from the time it leaves the RV until it is placed in the SFP racks. Underwater transfer of spent fuel provides an effective, economic, and transparent radiation shield as well as a reliable cooling medium for removal of decay heat. Boric acid is added to the water to ensure subcritical conditions.

The associated fuel handling structures may be generally divided into three areas: (a) the refueling cavity and refueling canal which are flooded only during plant shutdown for refueling, (b) the SFP which is kept full of water and is always accessible to operating personnel, and (c) the new fuel storage area which is separate and protected for dry storage. The refueling canal and the SFP are connected by the fuel transfer tube. This tube is fitted with a quick opening hatch on the canal end and a gate valve on the SFP end. The quick opening hatch is in place, except during refueling, to ensure containment integrity.

The new fuel containers are unloaded from the shipping vehicle and placed on the 115 foot elevation using the fuel handling area crane. New fuel assemblies are removed one at a time from the shipping containers using the new fuel handling tool and the spent fuel bridge hoist such that only one fuel assembly is moved or suspended at a time in a specific area. Plant administrative controls do not permit more than one fuel assembly to be out of storage or in transit between its associated shipping cask and dry storage rack at one time. The assemblies are stored either in the NFSRs in the fuel storage area or in the SFP. Each assembly is inspected for possible shipping damage prior to insertion into the reactor core.

New fuel is delivered to the reactor by first transferring the fuel with the spent fuel bridge hoist to the new fuel elevator. The fuel is lowered into the SFP where the spent fuel handling tool is interchanged with the new fuel tool. The assembly is then stored in the SFP or transferred to the upender for movement to the reactor.

The upender at either end of the fuel transfer tube is used to pivot a fuel assembly to the horizontal position for passage through the transfer tube. Fuel is carried through the tube on a transfer car. After the transfer car transports the fuel assembly through the transfer tube, the upender at that end of the tube pivots the assembly to a vertical position so that it can be lifted out of the fuel container. Fuel is moved between locations in the RV and the transfer mechanism by the manipulator crane.

In the SFP, fuel assemblies are moved about by the SFP bridge hoist. A long-handled tool is used with the bridge hoist to prevent the lifting of fuel assemblies any closer than 8 feet from the SFP surface; this ensures that sufficient radiation shielding is maintained. A shorter tool is used to handle new fuel, but the new fuel elevator must be used to lower the assembly to a depth at which the hoist and the long-handled tool can be used to place the new assembly into the upender or into a SFP cell.

When fuel repair or post-irradiation examinations are necessary, the fuel assemblies may be reconstituted in the new fuel elevator. The new fuel elevator is modified temporarily by the insertion of hard stops and resetting the upper electrical limit switch to prevent raising the irradiated fuel assemblies to within eight feet of the SFP surface. The SFP bridge hoist is utilized to transfer fuel rods within the SFP. The tooling on the hoist is configured to maintain nine feet of water shielding over the active fuel.
Decay heat, generated by the spent fuel assemblies in the SFP, is removed by the SFP cooling and cleanup system. Refer to Section 9.1.3.3.6 for discussion on decay heat removal.

The decontamination area has a stainless-steel-lined base, and a curb is provided around it to prevent the water and solvents used during decontamination from spreading over the building floor. Drains in the floor of the area remove the decontaminants to the waste disposal system for processing.

9.1.4.2.1 Component Description

The following sections describe major components of the FHS as they relate to refueling and fuel transfer operations.

In the event of a power failure, no hazardous condition would exist during refueling or spent fuel handling. Electric motors associated with the manipulator crane, SFP bridge crane and hoist, fuel handling area crane, fuel transfer system, and new fuel elevator have solenoid-actuated brakes capable of holding the rated loads during a power failure. The Unit 2 SFP bridge crane trolley has an electromagnetic brake. When current is interrupted to the motor, the Unit 1 solenoid or Unit 2 electromagnetic brake is spring set. During a power failure, a fuel assembly being handled would remain in the position held at the time of failure.

9.1.4.2.1.1 Manipulator Crane

The manipulator crane shown in Figure 9.1-8 is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the refueling cavity and runs on rails set into the edge of the refueling cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly. A long tube with a pneumatic gripper on the end is lowered down out of the mast to grip the fuel assembly. The gripper tube is long enough so that the upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. While inside the mast tube, the fuel is transported to its new position.

All controls for the manipulator crane are mounted on a console on the trolley. The bridge is positioned on a coordinate system laid out on one rail. A video indexing system with a camera mounted to the bridge, over an indicating scale, and a monitor on the console indicates the position of the bridge. The trolley is positioned with the aid of a scale on the bridge structure. The scale is read directly by the operator at the console. The drives for the bridge, trolley, and winch are variable speed and include a separate inching control on the winch. Electrical interlocks and limit switches on the bridge and trolley drives prevent damage to the fuel assemblies. The winch is provided with a limit switch and a backup programmable limit switch to prevent a fuel assembly from being raised above a safe shielding depth. In an emergency, the bridge, trolley, and winch can be operated manually by means of handwheels on the motor shafts.

The main and auxiliary hoists are equipped with two independent braking systems. A solenoid release-spring set electric brake is mounted on the motor shaft. This brake operates in the normal manner to release upon application of current to the motor and set when current is interrupted. The second brake is a mechanically actuated load brake internal to the hoist gear box that sets if the load starts to overhaul the hoist. It is necessary to apply torque from the motor to raise or lower the load.

In raising, the motor cams release the brake open; in lowering, the motor slips the brake allowing the load to lower. This brake actuates upon loss of torque from the motor for any reason and is not dependent on any electrical circuits. On the main hoist the motor brake is rated at 350 percent of operating load and the mechanical brake at 300 percent.

The manipulator crane structure is designed for Class C, Moderate Service, as defined by the Overhead Electric Crane Institute Specification No. 61. The electrical interlocks that ensure safe operation are designed to meet the single failure criteria of IEEE 279-1971 (Reference 4). The electrical wiring meets the applicable requirements of the National Fire Code, Electrical, Volume 5, Article 610. The design of the crane meets the applicable requirements of Section 1910.179 of subpart N of the OSHA Code. The bridge, trolley, and hoist drive motors are NEMA Class D induction motors with Class H insulation. The hoist rope is stainless steel, and the load rating is sufficient to support five times the design load. The crane design class is provided in Table 3.2-3.

The manipulator crane design load is the dead weight plus 4500 lb (three times the fuel assembly weight). The crane was erected in the shop and given a complete functional test including a load test at 110 percent of the design load. The maximum operating load of fuel assembly plus gripper is approximately 2500 lb. The gripper itself has four fingers gripping the fuel, any two of which will support the fuel assembly weight. Test loads during the life of the facility are in accordance with requirements established by the State of California Division of Occupational Safety and Health as part of its responsibilities for implementing OSHA in the state.

9.1.4.2.1.2 Spent Fuel Pool Bridge

The Unit 2 spent fuel pool (SFP) bridge, shown in Figure 9.1-9A, is a wheel-mounted walkway spanning the SFP and carrying two monorail hoists on an overhead structure. Both hoists have a maximum lift capability of 43 feet. One hoist is used for maintenance of the fuel transfer system and for removing new fuel assemblies from their shipping containers.

The Unit 1 SFP bridge, shown in Figure 9.1-9B, is a wheel-mounted walkway spanning the SFP and carrying two monorail hoists on an overhead structure. One hoist has a maximum lift capability of 21 feet. The second hoist has a lift capability of 61 feet. This hoist is used for maintenance of the fuel transfer system and for removing new fuel assemblies from their shipping containers. Fuel assemblies are moved within the SFP by means of a long-handled tool suspended from either hoist. The bridge, trolley, and hoists are all electrically driven. The maximum lift of either hoist, combined with the long-handled tool length, is designed to maintain a safe shielding depth above a spent fuel assembly within the pool.

For fuel repairs and post-irradiation examinations, the fuel rod handling tool configuration on the hoist is required to be physically verified to maintain nine feet of water shielding in the SFP prior to handling irradiated fuel rods. The fuel rod handling tool configuration on the hoist is controlled by an approved administrative control.

The design class of the bridge is given in Table 3.2-3.

9.1.4.2.1.3 Fuel Handling Area Crane

The fuel handling area crane is an overhead bridge crane located in the fuel handling area at elevation 170 feet. This crane, shown in Figure 3.8-59, has a 125 ton capacity main hook for handling spent fuel casks and a 15 ton capacity auxiliary hook for handling new fuel shipping containers. The crane structures and components responsible for lifting of the maximum critical load (MCL) and distribution of the load to the FHB superstructure have been upgraded to meet single-failure-proof criteria through the replacement of the original trolley and reanalysis of existing structures retained in the new design.

The crane was originally designed and fabricated to the Specification for Electrical Overhead Traveling Cranes for Steel Mill Service, Association of Iron and Steel Engineers Standard No. 6 (tentative) dated May 1, 1969. All members not covered by that standard are designed and fabricated in accordance with the Specification for the Design, Fabrication and Erection of Structural Steel for Buildings by the AISC, dated February 12, 1969.

The electrical installation and all electrical equipment is in accordance with the National Electrical Code dated 1968, and the National Electrical Manufacturers Association.

Design, fabrication, and erection of the crane rail and crane support structure is in accordance with the AISC specifications. The fuel handling area crane complies with the requirements of OSHA Subpart N, Materials Handling and Storage, of 29 CFR Part 1910, Section 1910.179.

The integrity of the crane load transfer path is ensured by the following measures:

(1) The main hoist sister hook is designed such that the loads imposed by the MCL at both the pin location and the hooks result in factors of safety of 3.45 against yield strength of the hook material. The application of the MCL is considered to be fully carried by each of the attachment points. Stresses in the shank and nut are designed to remain below levels resulting in a factor of safety of 10 against the ultimate strength of the hook and nut material.

(2) The main hook and nut is shop tested at 2 times the MCL.

(3) Following the shop test, the main hook receives a volumetric and liquid penetrant inspection.

(4) The main hoist hooks and auxiliary hoist hook are field tested for 10 minutes at 1-1/4 times their rated loads.

(5) The main hoist ropes are designed with greater than 10:1 safety factor against breaking strength. They are part of a four part double reeving system that is designed to withstand the loss of any one of the four ropes without the loss of function and without significant vertical motion of the MCL.

(6) All components in the load transference path that must retain their structural integrity are considered to be Critical Items as defined by ASME NOG-1-2004. These Critical Items have material traceability and are subject to appropriate non-destructive examination to provide assurance against material failure.

(7) The crane is equipped with redundant main hoist drive trains, brakes and reeving. The failure of any one component in the reeving or drive train will not result in a load drop or excessive vertical motion of the load.

(8) The electrical power supply to the crane contains a means for automatic disconnection should a seismic event occur. The loss of power results in the brakes setting to retain any load that may be currently lifted and assures that un-commanded motion due to relay chatter or other interaction will not occur. An emergency lowering procedure is available should this be necessary.

(9) The crane is equipped with redundant protections against overload. A trip signal is provided from the weight monitoring system, which provide direct indication, as well as the main hoist variable frequency drive control system, which provides load sensing by monitoring the motor torque.

(10) The crane is equipped with redundant protections against two-blocking. The first line of protection is provided by a limit switch that provides a trip signal if the bottom block approaches a two-block condition. The second level of protection is provided by the weight monitoring system, which will provide a trip signal if sensed load exceeds a preset value. A third level of protection is provided by the upper block hydraulic support system, which is designed to mitigate the loadings created during an actual two-block event. The upper block pivot arm limit switch functions to assure hoist motion is stopped prior to the end of hydraulic cylinder travel. The two-block protection system is fully shop tested at each protection level to verify the effectiveness of the system.

(11) The crane has been subjected to a Cold Proof Test as allowed in NUREG-0554, May 1979, since verification of the Nil Ductility Transition characteristics of the existing bridge components is not available. The Cold Proof Test provides assurance that the crane will not fail due to brittle fracture. The operation of the crane is administratively controlled such that crane operations are not allowed when the structure temperature is below the temperature at which the Cold Proof Test has been performed.

Conservative fleet angles, drum diameter and sheave diameter are utilized in the design in accordance with ASME NOG-1-2004 and NUREG-0554, May 1979. These conservative design parameters assure wear and fatigue of the ropes is minimized.

The hoisting ropes are 1-1/2 inch Python Power 9 V EEIPS with a 172.8 ton breaking strength. The hoisting rope for the main hook is stainless steel. The hoisting rope for the auxiliary hook (which is configured in four part double reeving) is 3/8 inch Python stainless steel with a breaking strength of 11.2 tons.

Each hoist drive has two brakes, one with time delay application. These brakes are automatically, mechanically applied in the absence of motion command or crane power. A regenerative type brake is also supplied to provide controlled lowering of a load should the main hoist motor fail to operate. Should the regenerative brake fail, emergency lowering procedures utilizing controlled manual release of the redundant mechanical brakes are available. All hoist brakes are rated at 150 percent of maximum torque that can be developed by its respective system. Trolley and bridge brakes are rated at 100 percent of motor full load torque. The bridge and trolley braking systems are manually adjusted to allow motion in the N-S and E-W directions during a seismic event. The brake adjustment is administratively controlled and is required to ensure the qualification of the FHB superstructure. A vertical stop is provided along the entire length of the runway to prevent bridge uplift and derailing. Vertical restraints are also provided on the trolley to prevent uplift and derailing.

The fuel handling area crane is equipped with a skeleton cab containing a control console chair. The crane may also be controlled by a radio remote control, which may be harness mounted and worn by the operator. Transfer of operation of the crane to the remote control is provided by a control chair mounted transfer switch.

The fuel handling area crane is also used to relocate the movable partition walls, which function to segregate the Unit 1 and Unit 2 SFP areas from the hot shop as described in Section 9.1.4.2.1.4. The crane control chair and radio remote control are equipped with switches to latch and unlatch the mechanism connecting the partition walls to the crane bridge structure. Indication of a latched condition is provided in two locations on the underside of the bridge girders.

Following erection, the hooks, lifting mechanisms, cables, brakes, trolleys, and structural members of the fuel handling area crane are tested at 1/2, 3/4, 1, and 1-1/4 times the rated load. The test loads are maintained for a minimum of 10 minutes prior to changing to a different load.

During fabrication of the upgraded trolley assembly, each hoist was functionally load tested at 1 times the rated load and proof load tested at 1-1/4 times the rated load. The emergency manual load lowering function of the main hoist was tested at 1 times the rated load. Following installation atop the existing bridge crane girders, each hoist was proof load tested at 1-1/4 times the rated load. During the site proof load test of each hoist, the trolley and bridge motions were also tested to the maximum extent practicable.

The State of California assumed responsibility for the implementation of OSHA on January 1, 1974. The crane test loads to be used throughout the life of the facility will meet specific regulations for such loads, as established by the California Division of Industrial Safety.

The conservative design stress limits and redundant design features afforded by the single-failure-proof trolley and reanalyzed crane structures provide the added reliability to justify considering the crane as single-failure-proof. Nonetheless, the travel of both hooks over the SFP is restricted as described in Section 9.1.2.

9.1.4.2.1.4 Fuel Handling Area Movable Partition Walls

Movable partition walls allow the fuel handling area crane access to the fuel handling areas for both units and to the hot shop area, while maintaining proper operation of the FHBVS. The FHBVS is described in Section 9.4.4. Additionally, the movable partition wall panel in Unit 1 and the movable partition wall panel in Unit 2 (panel 1 and panel 4 as shown in Figure 9.1-19) are equipped with a monorail.

The fuel handling area crane may be used in the fuel handling area for Unit 1, the fuel handling area for Unit 2, or the hot shop area. Two movable partition walls are provided for the plant. These movable partition walls are repositioned when it is necessary to move the fuel handling area crane from one area to another. This repositioning is accomplished prior to the start of any fuel handling operations. The location of the movable partition walls is shown in Figure 9.1-19. Figure 9.1-20 shows a movable partition wall in place along either column line 15^{7} or column line 20^{3} where the wall serves to isolate the FHBVS from the hot shop area.

A monorail is attached to the side of the movable partition wall panels, each with a capacity of 4000 pounds. Travel is limited to the eastern end of these monorails to prevent the lifting of heavy loads over spent fuel. Any lifting of items in the western portions are administratively controlled by the Control of Heavy Loads Program (refer to Section 9.1.4.3.10).

The movable partition walls travel on wheels on the same rail track used by the fuel handling area crane. A movable partition wall is moved by securely attaching it to the fuel handling area crane and moving both the crane and the movable partition wall along the track as an integrated unit.

All members are designed as described in Section 3.8.2.3. The applicable code used in the design and fabrication is the Specification for the Design Fabrication and Erection of Structural Steel for Buildings, AISC, February 12, 1969. The movable partition wall panels and monorails have been evaluated for a HE concurrent with a maximum lifted load attached.

Wheels for the movable partition walls are double flanged. Additional assurance that derailing cannot take place is provided by a vertical stop running the entire length of each track. Details showing the movable partition wall, track, and vertical stop are shown in Figure 9.1-21.

9.1.4.2.1.5 Containment Structure Polar Crane

The containment structure polar crane (one for each unit) is an overhead gantry crane located on top of the circular crane wall at elevation 140 feet. This polar crane has a main hook capacity of 200 tons and an auxiliary hook capacity of 35 tons. In addition to the main and auxiliary hooks, the polar crane is equipped with a dome service crane.

The dome service crane has a manbasket rated at 750 pounds to lift tools and personnel for inspection and maintenance of the spray ring headers and upper containment structure. It also has an auxiliary hoist rated at 1950 pounds. The arrangement for the polar crane is shown in Figure 3.8-23.

Structural design is in accordance with the Specification for Electrical Overhead Traveling Cranes for Steel Mill Service, Association of Iron and Steel Engineers Standard No. 6 (tentative) dated May 1, 1969. All members not covered by that standard are designed according to the Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, AISC, dated February 12, 1969.

The electrical installation and all electrical equipment are in accordance with the National Electrical Code, dated 1968, and the National Electrical Manufacturers Association. The containment structure polar crane complies with the requirement of OSHA Subpart N, Materials and Handling Storage, of 29 CFR Part 1910, Section 1910.179. Fabrication and erection of the crane support rail is in accordance with the AISC specifications. Design, fabrication, and erection of the concrete supporting structure are described in Section 3.8.2.1.

The integrity of the crane hooks is ensured by the following measures:

(1) Stresses in the hooks and all other mechanical parts are limited to 80 percent of yield strength for the effect of maximum torque of the motors, braking, or collision of the trolley against the rail stops.

(2) Each hook is shop tested at 1-1/2 times its rated load.

(3) Following the shop test, each hook is magnetic particle inspected.

(4) Both hooks are field tested for 10 minutes at 1/2, 3/4, and 1-1/4 times the rated loads.

The pitch diameter of running sheaves is required to be not less than 24 times the nominal rope diameter. Likewise, the drum diameter is required to be not less than 24 times the nominal rope diameter.

The hoisting ropes are 6 x 37, uncoated, extra flexible, preformed, improved plow steel rope with hemp core or independent wire rope core. The maximum calculated stress in the ropes considering the efficiency of the reeving and weight of the blocks in addition to the crane rated load, is limited to 1/6 of the manufacturer's specified breaking strength.

The entire reeving system is designed so that minimum and commonly accepted fleet angles are maintained, and the rope is guarded against leaving the drum grooves or sheaves.

Each hoist drive on the Unit 1 and Unit 2 polar crane has two sets of brakes. The primary set is an alternating current, quick acting brake and the secondary set is a direct current, inherently slower acting brake. All these brakes are automatic and mechanically applied when the current to the motors is cut off. All hoist brakes are rated at 150 percent of maximum torque that can be developed by its respective system. Trolley and gantry brakes are rated at 100 percent of motor full load torque.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.

After installation, the gantry legs and beam were tested as a structure (without running gear) with a load of 414 tons for 1 hour. With a 250 ton load, two 10-minute lifts plus gantry and trolley travel tests of the assembled crane were conducted. The maximum operating load for the containment polar crane is 185 tons.

The conservative design stress limits, the dual braking system, the preoperational tests and the test loads used throughout the life of the facility all combine to provide assurance against the modes of failure that otherwise might be assigned to a gantry crane. Nonetheless, the travel of both hooks over the opened RV is restricted and controlled by administrative controls.

During plant operation, the polar crane is parked unlocked and provided with guides to prevent derailment due to a seismic event. The guides resist horizontal shear normal to the crane rails. The crane rail is anchored continuously to the concrete by special clamps, capable of resisting forces due to an earthquake.

9.1.4.2.1.6 New Fuel Elevator

The new fuel elevator shown in Figure 9.1-10 consists of a box-shaped elevator assembly with its top end open and sized to house one fuel assembly. Depth of the structure is slightly less than the overall length of the fuel assembly, which rests on the bottom plate. The design class of the new fuel elevator is listed in Table 3.2-3. The new fuel elevator has been evaluated for the DDE and the HE concurrent with a maximum lifted load attached.

The new fuel elevator is used to lower a new fuel assembly to the bottom of the SFP where it is transported to the fuel transfer system by the SFP bridge hoist. It is also used to raise the dummy fuel assembly out of the SFP for transfer between units and occasionally for raising a newly received assembly to the surface for additional inspection.

The new fuel elevator can also be used for handling a spent fuel assembly, for example, during fuel assembly repair or post-irradiation examinations. The restriction imposed by the minimum water shielding requires that:

(1) Administrative controls are imposed to maintain an adequate submergence.

(2) The upper limit switch is adjusted to trip the hoist at a lower elevation to ensure adequate submergence.

(3) Mechanical stops are installed prior to insertion of a spent fuel assembly in the new fuel elevator to physically limit the raising of the elevator to ensure minimum submergence.

(4) No other fuel assembly movement is allowed in the SFP while an assembly is in the new fuel elevator.

9.1.4.2.1.7 Fuel Transfer System

The fuel transfer system (refer to Figure 9.1-11) includes a transfer car that runs on tracks extending from the refueling canal through the transfer tube into the SFP and an upender lifting frame at each end of the transfer tube. The upender in the refueling canal receives a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is then pivoted to a horizontal position for passage through the transfer tube and pivoted to a vertical position by the upender in the SFP. The SFP bridge hoist takes the fuel assembly to a position in the spent fuel storage racks.

A quick opening hatch is used to close the refueling canal end of the transfer tube to seal the reactor containment. The terminus of the tube outside the containment is closed by a gate valve. The design class of the fuel transfer system is given in Table 3.2-3.

9.1.4.2.1.8 Rod Cluster Control Changing

RCCAs inside containment are transferred from one fuel assembly to another by the RCC changing fixture shown in Figure 9.1-12. The five major subassemblies of the changing fixture are: frame and track structure, carriage, guide tube, gripper, and drive mechanism. The carriage is a movable container supported by the frame and track structure. The tracks provide a guide for the four flanged carriage wheels and allow horizontal movement of the carriage during changing operations. Positioning stops on both the carriage and frame locate each of the three carriage compartments directly below the guide tube. Two of these compartments hold individual fuel assemblies while the third supports a single RCCA. The guide tube, situated above the carriage and mounted on the refueling canal wall, provides for the guidance and proper orientation of the gripper and RCCA as they are being raised or lowered. The pneumatically actuated gripper engages the RCCA. Two flexure fingers can be inserted into the top of the RCCA when air pressure is applied to the gripper piston. Normally, the fingers are locked in a radially extended position. Mounted on the operating deck, the drive mechanism assembly includes the manual carriage drive mechanism, revolving stop operating handle, pneumatic selector valve for actuating the gripper piston, and electric hoist for elevation control of the gripper. The fixture is located in the containment refueling canal. The design class of the RCC changing fixture is PG&E Design Class III.

RCCAs in the SFP are transferred from one fuel assembly to another by the RCC changing tool shown in Figure 9.1-12a. The RCC changing tool is portable and functions in a manner similar to the RCC changing fixture described above. This tool is suspended from the SFP bridge crane hoist and is operated from the bridge crane walkway. The tool is lowered by the bridge hoist until it rests upon the nozzle of the desired fuel assembly seated in the spent fuel storage rack. The gripper actuator is then lowered and latched onto the RCC spider which allows the entire RCC to be drawn up inside the guide tube of the tool. Once this operation is completed, the tool may be repositioned over another fuel assembly. The above process is then reversed for reinsertion of the RCC. The RCC changing tool is PG&E Design Class II and is stored on the wall of the fuel transfer canal or SFP as needed. The tool consists of three basic assemblies: the guide tube, the support tube, and the drive mechanism.

The guide tube is the square cross-sectioned tube at the bottom of the tool. Guide plates are provided over the entire length of the tube to prevent damaging the rod clusters and to properly align the gripper. The gripper actuator is also contained within the guide tube. The electro-mechanical RCC changing tool uses a pneumatic gripper, and is operated using controls on the top of the tool. Two limit switches provide upper and lower limits for the motion of the unit. The bottom of the RCC changing tool is equipped with guide pins to insure alignment of the tool with the fuel assembly.

Above the guide tube is the support tube, which gives the proper length to the tool, provides support for the gripper actuator, and supplies protection for the lift cable. Also enclosed within the support tube are the air hose for the gripper and the electrical cable for the limit switches. To prevent tangling of the hose and cable, the cable has been placed inside the coiled air hose with seals at each end to allow separation of the two.

The drive mechanism, at the top of the tool, consists of a winch powered by an ac electric motor, the operator's panel, and four limit switches. One of the limit switches provides overload protection in the event of an RCC hang-up. The other three are geared limit switches; two providing upper and lower limits and the third controls the pneumatic system.

9.1.4.2.1.9 Spent Fuel Handling Tool

The manually actuated spent fuel handling tool, shown in Figure 9.1-13, is used to handle new and spent fuel in the SFP. It is mounted on the end of a long pole suspended from the SFP bridge hoist. An operator on the SFP bridge guides and operates the tool. The tool is stored on the wall of the SFP.

9.1.4.2.1.10 New Fuel Assembly Handling Fixture

The short-handled new fuel assembly handling fixture, shown in Figure 9.1-14, is used to handle new fuel on the operating deck of the fuel storage area, to remove the new fuel from the shipping container, and to facilitate inspection and storage of the new fuel and loading of fuel into the new fuel elevator.

9.1.4.2.1.11 Reactor Vessel Head Lifting Device

The RV head lifting device, shown in Figure 9.1-15, consists of a welded and bolted structural steel frame with suitable rigging to enable the crane operator to lift the head and store it during refueling operations. The lifting lugs are permanently attached to the RV head. The RV head lifting device is a part of the integrated head assembly (IHA), as described in Section 5.4.1.4. During evolutions that require the removal of the reactor pressure vessel head, the entire IHA is lifted at once. The design class of this device is given in Table 3.2-3.

9.1.4.2.1.12 Reactor Internals Lifting Device

The reactor internals lifting device, shown in Figure 9.1-16, is a structural frame suspended from the overhead crane. The frame is lowered onto the guide tube support plate of the internals, and is manually bolted to the support plate by three bolts. Bushings on the frame engage guide studs in the vessel flange to provide guidance during removal and replacement of the internals package. The device is stored on a PG&E Design Class II stand in the containment refueling canal.

9.1.4.2.1.13 Reactor Vessel Stud Tensioner

Stud tensioners, shown in Figure 9.1-17, are employed to secure the head closure joint at every refueling. The stud tensioner is a hydraulically operated device that permits preloading and unloading of the RV closure studs at cold shutdown conditions. A hydraulic pumping unit operates the tensioners, which are hydraulically connected in series.

9.1.4.2.2 Tool Storage Locations and Supports

The storage locations of miscellaneous fuel handling tools are shown in Figure 9.1-18. The design classifications for these locations (containment structure, FHB, and support racks) are given in Table 3.2-3. Many of the major and miscellaneous tool supports are designed to meet PG&E Design Class I requirements. If not, the tools are stored in an area such that their failure or failure of their supports would not endanger plant operation or prevent safe plant shutdown during a seismic event.

9.1.4.2.3 Refueling Procedure

The refueling operation follows a detailed procedure that provides a safe, efficient refueling operation. The following significant points are ensured by the refueling procedure:

(1) The refueling water and the reactor coolant contain a minimum of approximately 2000 ppm boron (refer to Section 9.1.3.3.7).

(2) The water level in the refueling cavity is high enough to keep the radiation levels within acceptable limits when the fuel assemblies are being removed from the core. This water also provides adequate cooling for the fuel assemblies during transfer operations. Refer to Section 9.1.4.3.6 for discussion regarding methods for maintaining adequate radiation shielding during the fuel handling process.

(3) The potential for grid strap damage during refueling operations is minimized by exercising care during the handling operations. Such care includes proper training of operators, ensuring adequate water clarity and lighting, confirmation of proper functioning and alignment of the fuel handling and transfer equipment, and implementation of appropriate fuel handling precautions.

The refueling operation is divided into four major phases: (a) preparation, (b) reactor disassembly, (c) fuel handling, and (d) reactor assembly. A general description of a typical refueling operation through the four phases is given below:

(1) Phase I - Preparation

The reactor is shut down and cooled to cold shutdown conditions with a minimum boron concentration of 2000 ppm in the reactor coolant and a Keff $\leq$ 0.95. Radiological assessments are performed in accordance with the Radiation protection program prior to general entry into containment. The coolant level in the RV is lowered to a point slightly below the vessel flange. The fuel transfer equipment and manipulator crane are checked for proper operation.

(2) Phase II - Reactor Disassembly

All cables are disconnected at the 140' elevation, seismic braces are unpinned, RV level instrumentation system tubing is disconnected, reactor head flange insulation is removed, head studs are detensioned, and the head studs and nuts are removed. The refueling cavity is then prepared for flooding by sealing off the reactor cavity; checking the underwater lights, tools, and fuel transfer system; closing the refueling canal drain holes; and removing the quick opening hatch from the fuel transfer tube.

With the refueling cavity prepared for flooding, the vessel head is unseated, raised, and placed on the head storage stand. Water from the RWST is pumped into the RCS causing the water to overflow into the refueling cavity. The control rod drive shafts are disconnected and, with the upper internals, removed from the vessel. The fuel assemblies and RCCAs are now free from obstructions, and the core is ready for refueling.

Prior to moving the upper internals over fuel, each containment penetration will be in the following status:

(a) Equipment hatch capable of being closed and held in place by a minimum of four bolts

(b) One door in each air lock capable of being closed

(c) Each penetration providing direct access from the containment atmosphere to the outside atmosphere is capable of being closed by manual or automatic isolation valve, blind flange, or equivalent; or be capable of being closed by an operable automatic containment purge and exhaust valve.

Otherwise, all operations involving movement of the upper internals over fuel are suspended.

(3) Phase III - Fuel Handling

The refueling sequence consists of either a full core off-load or a partial core off-load and incore shuffle.

The full core off-load consists of removing all of the fuel assemblies from the core, storing them in the SFP, and then returning the partially spent assemblies, as well as the new assemblies that replace the fully spent assemblies, to the core according to the final reload configuration.

An incore shuffle consists of removing the fully spent fuel assemblies from the core to the SFP, rearranging the remaining partially spent assemblies in the core, and adding new assemblies to replace the removed fully spent assemblies.

The general fuel handling sequence is:

(a) The manipulator crane is positioned over a fuel assembly in the core.

(b) The fuel assembly is lifted by the manipulator crane to a predetermined height to clear the RV and still leave sufficient water coverage to eliminate any radiation hazard to the operating personnel.

(c) The fuel transfer car is moved into the upender inside containment.

(d) The fuel assembly container is pivoted to the vertical position by the upender.

(e) The manipulator crane is moved to line up the fuel assembly with the fuel transfer system.

(f) The manipulator crane loads a fuel assembly into the fuel assembly container of the fuel transfer car.

(g) The container is pivoted to the horizontal position by the upender.

(h) The fuel container is moved through the fuel transfer tube to the SFP by the transfer car.

(i) The fuel assembly container is pivoted to the vertical position. The fuel assembly is unloaded by the spent fuel handling tool attached to the SFP bridge hoist.

Crane operations with loads over the SFP are suspended with less than 23 feet of water over the top of irradiated fuel assemblies seated in the storage racks. The water level is specified in the Technical Specifications.

Crane operations with loads containing recently irradiated fuel over the SFP are suspended with no FHBVS trains operable. Recently irradiated fuel is defined as fuel that has been part of a critical reactor within the last 100 hours.

Crane operations with loads containing recently irradiated fuel over the SFP may proceed with one FHBVS train inoperable provided the operable train is capable of being powered from an operable emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal absorbers.

(j) The fuel assembly is placed in the spent fuel storage rack.

(k) The procedure for off-load is continued until all assemblies identified by the off-load procedure are removed. The core may be either partially or fully off-loaded.

(l) On core reload, assemblies are moved from the SFP storage location (or the new fuel storage vault via the new fuel elevator) to the fuel assembly container according to the reload procedure, and the fuel assembly container is pivoted to the horizontal position and the transfer car is moved back into the refueling canal.

(m) For an incore shuffle, partially spent fuel assemblies are relocated in the reactor core, and new fuel assemblies are added to the core.

(n) Any new assembly or transferred fuel assembly that is placed in a control position will receive an RCCA in the SFP using the RCC change tool.

(o) This procedure is continued until refueling is completed.

(4) Phase IV - Reactor Assembly

Reactor assembly, following refueling, is essentially achieved by reversing the operations given in Phase II - Reactor Disassembly.

9.1.4.2.4 Reactor Vessel Closure Head Lifting Administrative Controls

DCPP administrative controls are used to control the lift and replacement of the reactor vessel closure head (RVCH). They establish limits on load height, load weight, and medium present under the load. The administrative controls: (1) use the guidance and acceptance criteria in NEI 08-05, Industry Initiative on Control of Heavy Loads (Reference 21), particularly in regards to Section 2 of the initiative, which addresses criteria for RVCH load drop and consequences analysis; and (2) provide additional assurance that the core will remain covered and cooled in the event of a postulated RVCH drop.

9.1.4.2.5 Reactor Vessel Closure Head Load Drop Analyses

RVCH drop analyses have been performed for DCPP Unit 1 and Unit 2 in accordance with NEI 08-05, which was endorsed by the NRC, with exceptions, in the NRC safety evaluation of NEI 08-05 (Reference 22). The DCPP analyses were performed in accordance with the NRC exceptions, as follows:

(1) “The staff considers the ASME Code, Section III, Appendix F acceptance criteria for limiting events (i.e., Service Level D) acceptable for the analytical methods proposed in the guidance.” Therefore, the NEI 08-05 stress based criteria using ASME Section III, Appendix F, were used in the DCPP head drop analysis evaluation for coolant retaining components.

(2) “For energy balance evaluations using the large-displacement finite element methods described in the guidance, the staff finds the criteria

applied to pipe whip restraint evaluations (i.e., one-half of ultimate strain) acceptable for the analytical methods proposed in the draft guidance." The Standard Review Plan 3.6.2, Revision 1, criteria applicable to pipe whip restraint evaluations is 0.5 of the ultimate uniform strain limit for pure tension members. Therefore, the uniform tensile strain in tensile plus bending support members was limited to 0.5 of the ultimate strain.

The purpose of the analyses was to evaluate the consequences of a postulated heavy load drop of the RVCH in the reactor cavity, while raising or lowering the RVCH during outages at DCPP. The heads weigh 366 Kip, however for analysis purposes, a conservative head weight of 380 Kip dropping a distance of 38 feet in air was assumed. The RVCH centerline is considered concentric with the RV centerline. The 38-foot height limit is removed once the RVCH centerline is outside the RV flange outside diameter. The dynamic head drop analysis methodology used for DCPP is an acceptable methodology per NEI 08-05, Section 2.3, and the NRC safety evaluation of NEI 08-05 (Reference 22). Details of the analysis and results are provided in Reference 23.

The dynamic impact model includes a system of non-linear springs and masses of the falling head with the upper service structures, and impact on an integrated spring mass model of the RV target structures. The target structures consist of the RV shell, nozzles, the RCS loop piping, reactor coolant pumps (RCPs), and the steam generators (SGs). An impact damping ratio of 5 percent was applied in the analysis to address the plastic deformation of the contact surfaces of the head and the vessel flanges. The load-deflection curve of the non-linear spring elements are developed from a non-linear finite element analysis model of the components.

The dynamic impact analysis results indicate, for all drop cases, stresses in the RV shell, RV nozzles, RCS loop piping, bottom mounted instrumentation piping and all RCS attached piping meet the allowable stresses per ASME BPVC Section III, Appendix F, for faulted conditions per the NEI 08-05 Guideline as endorsed by the NRC. Stresses in the reactor coolant pressure retaining components are below the ASME Section III, Appendix F, stress limits for faulted conditions.

Concrete under the RV support base steel plate exceeds the local bearing stress limit, thus the RV supports were considered to be ineffective after the concrete crushing. Local concrete under the pipe whip saddle inside the wall penetration also partially crushed, however, the strain in pipe whip support plates remains below the allowable strain. Thus, the RVCH, RV, and RCS piping remain supported after the impact.

The SG and RCP nozzle loads and support loads due to head drop load impact are less than or approximately the same as the component faulted design loads (based on combined LOCA, seismic, pressure, and deadweight loads). Thus, the RCP and SG support structures would remain functional to support the weight of the RCS components after the head drop accident.

9.1.4.2.6 Cask Loading Operations

A summary description of the FHS equipment for cask loading operations is provided in this section. Additional details are included in References 10, 11, and 12. Refer to Section 4.4.1.3 of Reference 12 for the division of safety considerations inside and outside the 10 CFR Part 50 facility.

The FHS equipment needed to load spent fuel into a cask for transfer to the Diablo Canyon ISFSI consists of cranes, lifting and handling devices including tools, a low profile transporter (LPT), and an SFP transfer cask restraint cup. The primary structures associated with this equipment are the SFP, the cask loading area of the SFP, and the Unit 2 cask washdown area (CWA).

Decay heat, generated by the spent fuel assemblies in the SFP, is removed by the SFP cooling and cleanup system. When fuel is to be moved to the Diablo Canyon ISFSI, selected assemblies are removed from the racks and loaded into a multi-purpose canister (MPC) that is located inside a transfer cask (the HI-TRAC 125D cask shown in Figure 9.1-4). The cask handling route within the FHB is shown in Figure 9.1-7. The minimum cooling time for fuel to be loaded into the transfer cask is 5 years.

The cranes, lifting and handling devices, and tools used for cask loading operations are essentially the same as those used for refueling and fuel transfer operations (described in Section 9.1.4.2). They are, however, configured differently for cask loading operations, as further described below in Section 9.1.4.2.6.1. The LPT is used to move the empty transfer cask into the FHB. The transfer cask is detached from the LPT and moved into the Unit 2 CWA seismic restraint using the fuel handling area crane. While located in the CWA, the empty transfer cask/MPC is restrained as shown in Figure 9.1-24. When the transfer cask is moved to the cask loading area, it is lowered into the SFP transfer cask restraint cup (refer to Figure 9.1-6), which provides lateral support of the cask while it is lowered, loaded with fuel, and lifted from the SFP. Once the loaded cask is raised out of the SFP, it is moved to the Unit 2 CWA for decontamination, MPC processing, and preparation for transport to the cask transfer facility (CTF). When ready, the loaded cask is lifted out of the Unit 2 CWA, fastened onto the LPT, and moved out of the FHB for subsequent transport to the CTF using the cask transporter.

9.1.4.2.6.1 Component Descriptions

The following sections describe major components of the FHS as they relate to cask loading operations in the FHB. The cranes, lifting and handling devices, and tools described in Section 9.1.4.2.1 for refueling and fuel transfer operations are essentially the same as those used for cask loading operations. There are, however, some configuration differences. The component descriptions in this section focus on the configuration differences for the cranes, lifting and handling devices, and tools, as well as describing those components unique to cask loading and handling in the FHB. Other FHS components not addressed in this section remain the same as described in Sections 9.1.4.2.1 and 9.1.4.2.2.

9.1.4.2.6.1.1 HI-STORM 100 Interchangeable Multi-Purpose Canisters

The HI-STORM MPC provides for confinement of radioactive materials, criticality control, and the means to dissipate decay heat from the stored fuel. It is a welded cylindrical canister with a honeycombed fuel basket, which contains a Boral or Metamic neutron absorber for criticality control. Although MPC-24s and MPC-32s are certified for use at DCPP, only the MPC-32 is used. The MPC-32 is designed for intact spent fuel. All MPCs certified for use at DCPP have the same outside dimensions and use the same transfer cask.

9.1.4.2.6.1.2 HI-TRAC 125D Transfer Cask

The HI-TRAC 125D transfer cask (refer to Figure 9.1-4) contains the MPC during loading, unloading, and transfer operations. The physical characteristics of the 125D cask are shown in Table 4.2-3 of the Diablo Canyon ISFSI UFSAR . It provides shielding and structural protection of the MPC in the FHB and from the SFP to the CTF. The transfer cask is a multi-walled (carbon steel/lead/carbon steel) cylindrical vessel with a built-in exterior water jacket. The maximum weight including the lifting yoke during any loading, unloading, or transfer operation does not exceed 125 tons.

9.1.4.2.6.2 Transfer Cask/Multi-Purpose Canister Loading Process

The transfer cask/MPC loading process is briefly summarized in this section. Additional detail is provided in Section 3.2 of Reference 10 and in Section 5.1 of Reference 12.

An empty MPC is loaded into the transfer cask in the FHB using the fuel handling area crane. Borated water is added to the MPC and the transfer cask is lifted above the SFP wall using the fuel handling area crane. The cask is then traversed into position over the cask loading area of the SFP and SFP transfer cask restraint cup. The cask is lowered into the SFP transfer cask restraint cup, which rests on a platform near the bottom of the cask loading area. The restraint precludes tipping or damage to adjacent fuel racks.

After fuel loading is complete, a lid is placed on the MPC, the transfer cask is lifted out of the SFP, traversed and lowered into the Unit 2 CWA seismic restraint, and decontaminated. The MPC lid is welded to the MPC shell, hydro leak tested, water is drained from the MPC, and the MPC is dehydrated, filled with Helium, isolated, Helium leak tested, and seal welded to closure. The top lid is then installed on the transfer cask.

The fuel handling area crane is then used to lift the loaded transfer cask out of the CWA restraint and to place the loaded transfer cask onto the LPT. The cask is fastened to the LPT and the LPT is moved out of the FHB on removable tracks to a position where it is rigged to the cask transporter for movement to the CTF.

9.1.4.2.6.3 Unloading Operations

While unlikely, certain conditions described in Reference 12 may require unloading of the fuel assemblies from the transfer cask/MPC. The unloading process is generally the reverse order of the loading process.

9.1.4.3 Safety Evaluation

9.1.4.3.1 General Design Criterion 2, 1967 – Performance Standards

The FHB, auxiliary building, and containment structure, which contain the FHS are PG&E Design Class I (refer to Section 3.8). These buildings or applicable portions thereof are designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), and earthquakes (refer to Section 3.7). These designs protect the FHS, ensuring its safety functions are performed.

The cranes and movable partition walls of the FHS have been seismically qualified to either DEs, DDEs, or HEs, as discussed in Section 9.1.4.2.1.4.

The LPT, with a loaded transfer cask attached, is qualified for DE, DDE, and HE events.

Refer to Section 9.1.4.3.9 for further discussion of the seismic qualifications of the FHS equipment.

<u>Tornado Winds and Tornado Generated Missiles</u>

The overall tornado resistance of the FHB is addressed in Section 9.1.2.3.1. The effects of tornado wind loads acting on the cask suspended from the FHB crane are enveloped by the seismic analysis for this configuration. However, cask handling introduces new tornado missile targets. Analysis results demonstrate that the 125-ton transfer cask satisfies all functional requirements under postulated impact scenarios and the system will not be subject to a loss of load due to a missile impact. For discussions of wind and tornado loading and associated tornado generated missiles, refer to Sections 3.3 and 3.5, respectively.

9.1.4.3.2 General Design Criterion 4, 1967 – Sharing of Systems

The FHB is a steel framed building anchored to the concrete auxiliary building structure (refer to Section 3.8.2.3.4.1). The FHB encloses the fuel handling areas of Unit 1 and Unit 2 (refer to Figures 1.2-4 and 1.2-10). Therefore, the FHB is a shared structure.

One fuel handling area crane services both units (refer to Section 9.1.4.2.1.3). DCPP administrative controls are used to restrict the path of loads, such as the handling of spent fuel casks, which could damage fuel assemblies. The restrictions on fuel handling prevent a failure of the fuel handling area crane from negatively impacting the safety of both units (refer to Section 9.1.4.3.10).

Two movable partition walls are used in the FHB to allow the fuel handling area crane to transition from one unit to the other unit. The movable partition walls serve to separate the Unit 1 and Unit 2 areas of the FHB during fuel handling operations (refer to Section 9.1.4.2.1.4).

The function of the FHBVS is not impacted by the shared use of the movable partition walls, due to the fact that these walls are moved into their required positions before fuel handling begins. This allows the FHBVS to maintain the required negative atmospheric pressure for the fuel handling process (refer to Section 9.1.4.2.1.4).

Movement of fuel assemblies is controlled by DCPP administrative controls, ensuring the safe handling of fuel in the shared FHB. When the fuel handling area crane is not needed in the Unit 1 or Unit 2 fuel handling areas, the movable partition walls are positioned such that they provide a ventilation boundary between the hot shop and the fuel handling area of each unit (refer to Section 9.1.4.2.1.4).

9.1.4.3.3 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Electrical interlocks (i.e., limit switches) are provided for minimizing the possibility of damage to the fuel during fuel handling operations. The electrical interlocks for the manipulator crane are not specifically designed to the requirements of Reference 4 because of the primary dependence on mechanical stops.

The manipulator crane design includes the following provisions to ensure safe handling of fuel assemblies:

(1) Bridge, trolley, and winch drives are mutually interlocked, using redundant interlocks, to prevent simultaneous operation of any two drives.

(2) Bridge and trolley drive operation is prevented except when both gripper tubeup position switches are actuated.

(3) An interlock is supplied that prevents the operation of either the engaging or disengaging solenoid valves unless the load and elevation requirements are satisfied. As backup protection for this interlock, the mechanical weight-actuated lock in the gripper prevents operation of the gripper under load even if air pressure is applied to the operating cylinder.

(4) An excessive suspended weight switch opens the hoist drive circuit in the up direction when the loading is in excess of 110 percent of a fuel assembly weight.

(5) An interlock of the hoist drive circuit in the up direction permits the hoist to be operated only when either the open or closed indicating switch on the gripper is actuated.

(6) An interlock of the bridge and trolley drives prevents the bridge drive from traveling beyond the edge of the core unless the trolley is aligned with the refueling canal centerline. The trolley drive is locked out when the bridge is beyond the edge of the core.

The following safety features are provided for in the fuel transfer system control circuit:

(1) Transfer car operation is possible only when both upenders are in the down position as indicated by the limit switches.

(2) The remote control panels have a permissive switch in the transfer car control circuit that prevents operation of the transfer car in either direction when either switch is open; i.e., with two remote control panels, one in the refueling canal and one in the SFP, the transfer car cannot be moved until both "go" switches on the panels are closed.

(3) An interlock allows upender operation only when the transfer car is at either end of its travel.

(4) Transfer car operation is possible only when the transfer tube gate valve position switch indicates the valve is fully open.

(5) The refueling canal upender is interlocked with the manipulator crane. The upender cannot be operated unless the manipulator crane gripper tube is in the fully retracted position or the crane is over the core.

Fuel handling devices have provisions to avoid dropping or jamming of fuel assemblies during transfer operations.

9.1.4.3.4 General Design Criterion 49, 1967 – Containment Design Basis

The fuel transfer system, where it penetrates the containment, has provisions to preserve the integrity of the containment pressure boundary. The fuel transfer tube that connects the refueling canal (inside the containment) and the SFP (outside the containment) is closed on the refueling canal side by a quick opening hatch at all times except during refueling operations. Two seals are located around the periphery of the quick opening hatch with leak-check provisions between them. The terminus of the tube outside the containment is closed by a gate valve.

The portion of the fuel transfer tube inside the containment is considered to be part of the containment liner and is a Type B containment test penetration (refer to Table 6.2-39). In accordance with Technical Specifications, the fuel transfer tube penetration is tested up to expected LOCA pressure to ensure that the design leakage rate is not exceeded. Other than during refueling operations, the transfer tube remains empty therefore there are no means for affecting the functional design of the penetration.

9.1.4.3.5 General Design Criterion 66, 1967 – Prevention of Fuel Storage Criticality

As discussed in Section 9.1.4.2.6.1.1, the design of the Holtec HI-STORM MPC provides criticality control for spent fuel stored in the MPC.

The boron concentration of the water that communicates with the reactor pressure vessel during the refueling process is controlled by Technical Specifications and DCPP administrative controls to ensure that the fuel assemblies located inside the reactor pressure vessel remain subcritical.

During refueling operations when the SFP water mixes through the fuel transfer tube with the reactor refueling cavity water, SFP chemistry is controlled to ensure compatibility with the RCS and residual heat removal (RHR) system chemistry requirements to preclude a boron dilution accident.

For the period of time that an MPC is located in the MPC cavity of one of the SFPs, the water is free to communicate between the MPC and the SFP. During cask loading operations in the SFP, boron concentrations in the SFP (and thus, the MPC cavity) are maintained in accordance with the ISFSI Technical Specifications. For an MPC-32, a boron concentration range between 2000 ppm and 2600 ppm is required depending on the initial enrichment (wt%) of the loaded fuel assemblies. Boration levels up to 3,000 ppm in the MPC and SFP have been evaluated and determined to have no adverse effects on materials or thermal performance.

Protection from boron dilution events is discussed in Section 9.1.4.3.8.1.

9.1.4.3.5.1 Fuel Assembly Drop into Loaded Transfer Cask

While very unlikely, analysis of this event shows that a fuel assembly drop into the MPC could result in physical deformation that would challenge criticality margins. However, the results of the analysis show that the criticality margins would continue to meet the licensing basis (k_{eff} <0.95) and that the radiological consequences are enveloped by the existing fuel assembly drop accident described in Section 15.5.22.1.

9.1.4.3.6 General Design Criterion 68, 1967 – Fuel and Waste Storage Radiation Shielding

Cranes and hoists used to lift spent fuel assemblies have a limited maximum lift height so that the minimum required depth of water shielding is maintained. During all phases of spent fuel transfer, the gamma dose rate at the surface of the water is limited by maintaining a minimum of 8 feet of water above the top of the fuel assembly during all handling operations. This corresponds to about 9 feet of water shielding over the active fuel.

The two cranes used to lift spent fuel assemblies are the manipulator crane and the SFP bridge hoist. The manipulator crane contains positive stops, which prevent the top of a fuel assembly from being raised to within 8 feet of the water level in the refueling cavity. The hoist on the SFP bridge moves spent fuel assemblies with a long-handled tool. Hoist travel and tool length likewise limit the maximum lift of a fuel assembly to assure 8 feet of water shielding in the SFP.

When handling irradiated fuel rods, during fuel repairs and post-irradiation examinations, nine feet of water is maintained above the active fuel.

9.1.4.3.7 10 CFR 50.55a(g) – Inservice Inspection Requirements

The FHS has a periodic ISI program in accordance with the ASME BPVC Section XI.

9.1.4.3.8 10 CFR 50.68(b) – Criticality Accident Requirements

Based on the alarms, procedures, administrative controls, assumption of zero burnup fuel, and availability of trained operators described in Reference 13, the NRC has granted an exemption from the criticality requirements of 10 CFR 50.68(b)(1) during loading, unloading, and handling of the MPC in the DCPP SFP (Reference 14).

In accordance with 10 CFR 50.68(b)(1), plant administrative controls prohibit handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical.

Refer to Section 9.1.4.3.5 for a discussion on measures used during fuel handling activities to ensure the K_{eff} of the fuel assemblies remains below the limits established in 10 CFR 50.68(b)(4).

Implementation of 10 CFR 50.68(b)(8) was completed with the submittal of Revision 15 of the UFSAR.

9.1.4.3.8.1 Mulit-Purpose Canister Boron Dilution

The original boron dilution analysis was performed and submitted to the NRC (Reference 13) to determine the time available for operator action to ensure criticality does not occur in an MPC-32 during fuel loading and unloading operations. The results of the analysis of record show that operators have 4.7 hours available to identify and terminate the source of unborated water flow from the limiting boron dilution event to ensure criticality in the MPC-32 does not occur. To minimize the possibility of a dilution event, a temporary administrative control is implemented while the MPC is in the SFP that will require, with the exception of the 1-inch line used to rinse the cask as it is removed from the SFP, at least one valve in each potential flow path of unborated water to the SFP to be closed and tagged out. During the cask rinsing process, the MPC will have a lid in place that will minimize entry of any unborated water into the MPC. The flow path with the highest potential flow rate of 494 gpm is doubly isolated by having two valves closed and tagged out while the MPC is in the SFP.

9.1.4.3.9 Safety Guide 13, March 1971 – Fuel Storage Facility Design Basis

Regulatory Position 1:

The FHB, auxiliary building, and containment structure, which contain the FHS, are designed to withstand the effects earthquakes (refer to Section 9.1.4.3.1).

Fuel lifting and handling devices are capable of supporting maximum loads under seismic conditions. The fuel handling equipment will not fail so as to cause damage to any fuel elements should the seismic event occur during a refueling operation. The earthquake loading of the fuel handling equipment is evaluated in accordance with the seismic considerations addressed in Section 9.1.4.3.1. However, several components used in the fuel transfer system are not seismically qualified. This includes the upenders (one inside containment and one inside the FHB), the fuel transfer car, and the rails that the fuel transfer car rides on. Refer to Table 3.2-3 for further details on the classification of the fuel transfer system.

The maximum design stress for the fuel handling area crane structures and for all parts involved in gripping, supporting, or hoisting the fuel assemblies is 1/5 ultimate strength of the material. This requirement applies to normal working load and emergency pullout loads, when specified, but not to earthquake loading. To resist earthquake forces, the fuel handling area crane structures are designed to limit the stress in the load bearing parts to either 0.96 times the yield stress for a combination of normal working load plus DDE forces, or 1 times the yield strength for a combination of normal working load plus HE forces, whichever is greater. To ensure the ability of the lifting and handling devices to support maximum loads under seismic conditions, the design safety factor of the lifting and handling devices has been evaluated against the actual calculated HE loads. The evaluation indicates the inherent design safety factor is sufficient to envelope the combined normal working load plus DDE or HE loads.

For the manipulator crane, restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing due to the DDE and HE. The manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper under the DDE and HE. The SFP bridge crane is seismically qualified for the DDE and HE.

While in the Unit 2 CWA, the HI-TRAC is seismically restrained by the CWA seismic restraint system. This system includes a wall mounted restraint and service platform and a floor restraint plate.

The potential impact of seismic events on cask loading, handling, closure, and transport activities has been considered in the evaluation of the cask system components and in the design and evaluation of the interfaces with 10 CFR Part 50 facilities. Two structures, the SFP transfer cask restraint cup and the Unit 2 CWA seismic restraint structure, are designed to preclude unacceptable movement of the cask system components, assuring all involved SSCs remain within their design bases.

Seismic analyses have been performed that demonstrate the adequacy of the SFP transfer cask restraint cup and Unit 2 CWA restraint to preclude unacceptable movement or impact on the 10 CFR Part 50 facilities.

Regulatory Position 2:

Section 9.1.4.3.1 provides a discussion of the DCPP environmental protections, such as protection from cyclonic winds and external missiles, for the PG&E Design Class I portions of the FHS.

Regulatory Position 3:

Electrical interlocks (i.e., limit switches) are provided for minimizing the possibility of damage to the fuel during fuel handling operations. Mechanical stops are provided as the primary means of preventing FHAs. For example, safety aspects of the manipulator crane depend on the use of electrical interlocks and mechanical stops. The electrical interlocks for the manipulator are not specifically designed to the requirements of Reference 4 because of the primary dependence on mechanical stops.

Section 9.1.4.3.3 provides a more detailed overview of the interlocks used to ensure that fuel assemblies are safely handled.

Regulatory Position 5(b):

Crane operation in the fuel handling area is such that the heavy loads cannot traverse over the spent fuel storage racks in the SFP. Redundant electrical interlocks are installed on the fuel handling area crane to prevent movement of heavy loads over the area of the SFP which can contain spent fuel. The backup interlock is connected to a different circuit than the primary interlock to preclude heavy loads movement over stored fuel resulting from a single failure.

These limitations on fuel handling area crane travel preclude the possibility of dropping heavy objects from above the spent fuel storage racks. The spent fuel bridge hoist and the movable partition wall monorail are the only cranes capable of moving objects over the spent fuel storage racks.

The rated capacity of the Unit 2 spent fuel bridge hoist is 2500 pounds. The Equipment Control Guideline (ECG 42.4) prohibits movement of any load greater than 2500 pounds over fuel assemblies in the spent fuel pool. As stated in the ECG, the 2500 pounds is the approximate weight of a fuel assembly, control assemble, and associated handling tool. This is the assumed load in the fuel handling accident. The restriction on loads in excess of 2500 pounds over the other fuel assemblies in the spent fuel pool (SFP) ensures in the event this load is dropped that: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of the fuel in the storage racks will not result in a critical array.

The rated capacity of the Unit 1 spent fuel bridge hoist is 2000 pounds. An object of this weight dropped on the racks will not affect the integrity of the racks.

The rated capacity of the movable partition wall monorail is 4000 pounds, however, physical restrictions (trolley stops) are provided to prevent movement of loads over the SFP. Lighting fixtures or other components of the building above the racks are not sufficiently massive to cause damage to the racks if they are assumed to fall into the pool. Protection of nuclear fuel assemblies from overhead load handling is a key element of the Control of Heavy Loads Program described in Section 9.1.4.3.10.

9.1.4.3.10 NUREG-0612, July 1980 – Control of Heavy Loads at Nuclear Power Plants

The objective of the Control of Heavy Loads Program is to ensure that all load handling systems are designed, operated, and maintained such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The program is based on all seven general guideline areas of NUREG-0612, July 1980 Section 5.1.1, also known as Phase I (Safe Load Paths; Load Handling Procedures; Crane Operator Training; Special Lifting Devices; General Lifting Devices; Crane Inspection, Testing and Maintenance, and Crane Design).

The Control of Heavy Loads Program for refueling and fuel transfer operations is described in Section 9.1.4.3.10.3. The Control of Heavy Loads Program for cask loading operations, which includes revisions for loading the HI-STORM System components within the 10 CFR Part 50 facility, complies with the guidelines of NUREG-0612, July 1980. Details specific to cask loading and handling in the FHB are provided in References 10 and 11.

Generic Letter 80-113, December 1980 (Reference 18) required PG&E to review their provisions for handling and control of heavy loads at DCPP to determine the extent to which the guidelines of NUREG-0612, July 1980 Phase I and II were satisfied and to commit to mutually agreeable changes and modifications that would be required to fully satisfy these guidelines. An overview of the DCPP heavy loads program is presented in this section.

PG&E has developed and is maintaining a robust heavy loads control program at DCPP to minimize the potential for adverse interaction between overhead load handling operations and: 1) nuclear fuel assemblies to ensure a subcritical configuration and preclude radiological consequences and; 2) SSCs selected to ensure safe, cold shutdown of the plant following a postulated heavy load drop event. The bases of the NRC-accepted program are summarized in Reference 7.

Implementation of Sections 5.1.2 through 5.1.6 of NUREG-0612, July 1980, also known as Phase II, was determined by the NRC in Generic Letter 85-11, June 1985 to not require NRC review. While not a requirement, the NRC encouraged the implementation of any licensee actions identified in Phase II that are considered appropriate.

To accomplish the program, PG&E defined as heavy load targets nuclear fuel assemblies and selected SSCs necessary to safely shut down the plant and maintain the plant in a safe, cold shutdown condition. Initial plant operating modes of normal operation, shutdown, and refueling were considered in the selection of the target equipment. Overhead load handling operations and heavy load target SSCs were then evaluated for potential interaction. Mitigation measures for minimizing adverse interactions include as applicable: (1) to the extent possible, changing methods, routes or scheduling of the overhead load handling operation to avoid the interaction; (2) analyzing the intervening floor structural capacity for protection of target SSCs against postulated damage due to a load drop, and restricting overhead loads in the plant area by weight and handling height above the intervening floor (i.e., restricted area); and (3) excluding the plant area from non-essential overhead load handling operations (i.e., exclusion area). PG&E plans and capabilities to handle heavy loads at DCPP are described in PG&E correspondence to the NRC in response to NUREG-0612, July 1980 and are summarized in Reference 7. Sections 2.2.4, 2.3.4, and 2.4.2 of PG&E's NUREG-0612, July 1980 submittals (References 19 and 20) provide the results of various load drop analyses.

The results of these evaluations are used to create administrative controls for overhead load handling operations in plant areas where heavy load targets exist. The Plant Staff Review Committee is responsible for reviewing administrative controls for overhead load handling operation in exclusion areas (refer to Section 17.2.4). Additional controls for the training of crane operators, design, operation, maintenance and inspection of rigging, lifting devices, and overhead load handling systems are administered through plant administrative controls.

The movable wall partitions, which have monorail lifting devices attached, are designed and configured as described in Section 9.1.4.2.1.4.

9.1.4.3.10.1 Reactor Vessel Closure Head Load Drop

Controls implemented by NUREG-0612, July 1980 Phase I elements make the risk of a load drop very unlikely, and, in the event of a postulated RVCH drop, the load drop analysis demonstrates that the consequences are acceptable (refer to Section 9.1.4.2.5). Restrictions on load height, load weight, and medium under the load consistent with analysis assumptions are reflected in plant administrative controls.

9.1.4.3.10.2 RCP Motor Load Drop

While the likelihood of a load drop is very unlikely under DCPP's Control of Heavy Loads Program, in the event of a postulated drop of an RCP motor heavy load component, a load drop analysis demonstrates that the consequences are acceptable. Restrictions on load height, load weight, intervening structures, and plant conditions consistent with analysis are reflected in plant administrative controls. These administrative controls provide additional assurance that the core will remain covered and cooled in the event of a postulated drop of an RCP motor heavy load component. The consequences of dropping a fully assembled motor have not been analyzed.

RCP motor load drop analysis have been performed for DCPP Unit 1 and Unit 2 in accordance with NEI 08-05, which was endorsed by the NRC, with exceptions, in the NRC safety evaluation of NEI 08-05 (Reference 22). This is the same non-linear dynamic impact analysis methodology used for the RPVH drop analysis. The results of the analysis determined that the consequences satisfy the acceptance criteria of NUREG-0612 Section 5.1.3 part (3), using the guidelines of NUREG-0612 Appendix A. In the analysis, the RCS integrity is evaluated with three postulated drop locations, which are considered to be the worst consequences to the RCS pressure boundary components and their support components during a RCP motor load handling activity. The results of the analysis apply only in plant Modes 5 or 6 while the RCS is depressurized with the RPVH on or off the vessel.

The purpose of the analyses was to evaluate the consequences of a postulated heavy load drop of the heaviest RCP motor component, the upper end bracket (UEB), while raising or lowering the load over the RCP hatch area during plant Modes 5 and 6. The maximum analyzed below-the-hook load is 52,000 lbs lifted up to 5 feet above the 140-foot containment elevation, which conservatively bounds lifting the UEB and all lighter RCP motor components in any sequence. A design analysis of a Rotor Impact Absorber (RIA) is also performed to protect the RCP casing from excessive stresses if the rotor is lifted separately from the UEB and is concentrically dropped through the stator and the lower end bracket. Procedures and design documents control the use and configuration of the RIA.

9.1.4.3.10.3 Cask Loading Operations

Potential spent fuel cask accidents and off-normal events related to handling and loading (or unloading) of the MPC in the HI-TRAC 125D transfer cask, including safe handling as related to use of cranes and lifts; potential drops and tipovers; operational errors and mishandling events; support system malfunctions; and fires, are addressed in this section. Additional details of the methodology, acceptance criteria, and results are provided in Section 4.3 of Reference 10.

9.1.4.3.10.3.1 Defense-in-Depth Measures

9.1.4.3.10.3.1.1 Safe Handling

The safe handling discussion in Section 9.1.4.3.3 for the FHB cranes and lifts applies to cask loading operations. In addition, the FHB crane is configured and operated as described in Section 9.1.4.2.1.3 when performing cask loading and handling operations.

Structural, mechanical, and electrical design of the fuel handling area crane is in accordance with ASME NOG-1-2004, as conformed to the requirements of NUREG-0554, May 1979 per the guidance of NUREG-0612, July 1980, Appendix C. Furthermore, the crane structural design has been demonstrated to envelope the structural requirements of the original design codes as enumerated below. The main hoist, trolley, and bridge members meet ASME NOG-1-2004, Type I, design standards. Original bridge members and components retained for the current single-failure-proof design has been reanalyzed and inspected in accordance with the guidance provided by NUREG-0612, July 1980, Appendix C. The 15-ton auxiliary hoist meets ASME NOG-1-2004, Type II standards and therefore is not considered as single-failure-proof.

9.1.4.3.10.3.1.2 Drops and Tipovers

Upgraded cranes and hoists used during cask handling and loading minimize the potential for load drops. In particular, the FHB crane bridge and main hoist have been upgraded to meet single-failure-proof criteria, as described in Section 9.1.4.2.1.3.

The transfer cask, MPC and its internals, MPC lids, and spent fuel assemblies must be handled in and around the SFP and spent fuel (in the SFP and MPC). With the exception of the spent fuel assemblies, all of these items represent heavy loads.

The potential for drops or tipovers of any of these heavy loads is extremely small due to DCPP's Control of Heavy Loads Program and fuel-handling operations administrative controls. The Control of Heavy Loads Program provides procedures, training, and designs to minimize the potential for load drops, meets PG&E's commitments to NUREG-0612, July 1980, and has been accepted by the NRC (Reference 28). The single-failure-proof upgrade to the FHB crane further reduces the potential for a crane-related failure or mishandling event that could result in the drop of a cask.

Nonetheless, the following potential heavy-load drops have been postulated and evaluated, where credible, in accordance with the guidance of NUREG-0612, July 1980, Section 5.1, demonstrating defense in depth.

9.1.4.3.10.3.2 Cask Loading Accident Analyses

9.1.4.3.10.3.2.1 Loaded Transfer Cask Drops

PG&E has provided defense in depth through crane enhancements (described in Section 9.1.4.2.6) in those locations where a drop could have unacceptable consequences. Use of a single-failure-proof FHB crane ensures that an uncontrolled drop onto the edge of the SFP wall, which could allow the cask to tip or tumble horizontally into the SFP or into the CWA, is not credible. The single-failure-proof FHB crane also precludes drops during the placement of the transfer cask/MPC onto the LPT.

Further, movement of heavy loads over fuel in the SFP, or over any other safe shutdown systems or equipment identified in PG&E's NUREG-0612, July 1980 submittals, is controlled by administrative controls and considers the design of the single-failure-proof crane system and/or travel limit devices.

To ensure the cask does not adversely affect the stored spent fuel in the adjacent racks, the cask is inserted and seated in the transfer cask restraint cup during fuel loading or unloading operations. Structurally separating the transfer cask restraint cup from the fuel storage racks is the spent fuel cask restraint. The restraint is made of 12-inch Schedule 80S Type 304 stainless steel pipe, as shown in Figure 9.1-5.

The evaluations for postulated cask drops do not consider orientations for a tipped cask. This exception to NUREG-0612, July 1980, Appendix A, Consideration (1) is acceptable, due to cask restraints in the SFP and CWA.

9.1.4.3.10.3.2.2 Multi-Purpose Canister or Transfer Cask Lids, AWS Baseplate Shield, or Lifting Yoke Drops into Loaded Cask

While these components are classified as heavy loads and handling them over the MPC is required for dry fuel operations, drops are not considered credible because they are handled with a single-failure-proof overhead load handling system and rigged in accordance with the Control of Heavy Loads Program.

9.1.4.3.10.3.2.3 Operational Errors and Mishandling Events

The design of the dry cask handling system and associated administrative controls provide assurance that operational errors and mishandling events will not result in an increase in the probability or consequences of an accident previously analyzed. The following operational errors and mishandling events were evaluated and found to either have consequences within the design or licensing basis of DCPP, be precluded by compliance with the Control of Heavy Loads Program and/or operations procedures, or to not be credible.

- SFP Liner Breach Due to Cask Drop

- Crane Mishandling Operation with Transfer Cask/MPC Resulting in Horizontal Impact or Drops Outside of the Analyzed Lift Points

- Loss of the Transfer Cask Water Jacket Water During MPC and Cask Handling Operations

- Boron Dilution of the SFP and Associated Criticality Concerns

- Loading of an Unauthorized Fuel Assembly

9.1.4.3.10.3.2.4 Support System Malfunctions

The following support system malfunctions have been evaluated and found to not adversely affect plant safety:

- Loss of Electrical Power or Component Failures During Handling Operations

- Rupture of MPC Dewatering, Vacuum, FHD, or Related Closure System Lines or Equipment

- Failure of the LPT or Crane Handling Systems

9.1.4.3.10.3.3 Fires

The DCPP fire protection program is described in Section 9.5.1. The program incorporates the requirements of the ISFSI fire analyses, such that the required controls are provided to ensure the plant and the ISFSI components remain within their licensing bases.

Inside the FHB

The transporter and its associated fuel tank remain outside of the buildings. However, transient materials brought into the FHB associated with dry cask storage activities could provide additional fire loading. These activities and materials are under the control of DCPP's fire protection program. The current program ensures that ignition sources are monitored and that combustible loading requirements for the FHB areas are followed. To the extent practical, combustibles are kept away from the transfer cask to minimize the effects of any potential fire.

Outside the FHB

The fire protection program ensures potential fires during the transport and storage are handled consistently with the plant program requirements and meet the assumptions described in the Diablo Canyon ISFSI UFSAR. Prior to any cask transport, a walkdown is performed to ensure local combustible materials, including transient combustibles, are controlled in accordance with ISFSI fire protection requirements.

9.1.4.4 Tests and Inspections

As part of normal plant operations, the fuel handling equipment is inspected for operating conditions prior to each refueling operation. During the operational testing of this equipment, administrative controls are followed that will affirm the correct performance of the FHS interlocks.

Prior to each cask loading operation, the fuel and cask handling equipment is inspected for operating conditions During operational testing of the equipment, procedures are followed to affirm the correct performance of interlocks and controls.

9.1.4.5 Instrumentation Applications

For instrumentation and control systems for the FHS, refer to Section 9.1.4.3.3.

9.1.5 REFERENCES

1. Deleted in Revision 24.

2. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended.

3. U.S. Atomic Energy Commission, "Fuel Storage Facility Design Basis," Safety Guide 13, March 1971.

4. IEEE Standard, 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations, The Institute of Electrical and Electronics Engineers, Inc.

5. NRC Letter to PG&E, dated May 30, 1986, granting License Amendments No. 8 to Unit 1 and No. 6 to Unit 2.

6. License Amendment Request 95-01, submitted to the NRC by PG&E Letters DCL-95-28, dated February 6, 1995; DCL-95-063, dated March 23, 1995; DCL-95-112, dated May 22, 1995; and DCL-95-178, dated August 22, 1995.

7. PG&E Letter (DCL-96-111) to the NRC, “Response to NRC Bulletin 96-02, ‘Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment,’” dated May 13, 1996.

8. License Amendment Request 01-02, Credit for Soluble Boron in the Spent Fuel Pool Criticality Analysis, PG&E Letter DCL-01-096, dated September 13, 2001, supplemented by PG&E Letter DCL-02-022, dated February 27, 2002.

9. License Amendments 154/154, Credit for Soluble Boron in the Spent Fuel Pool Criticality Analysis, issued by the NRC, September 23, 2003.

10. License Amendment Request 02-03, Spent Fuel Cask Handling, PG&E Letter DCL-02-044, dated April 15, 2002.

11. License Amendments 162 and 163, Spent Fuel Cask Handling, issued by the NRC, September 26, 2003.

12. Diablo Canyon ISFSI Final Safety Analysis Report Update.

13. PG&E Letter (DCL-03-126) to the NRC, “Request for Exemption from 10 CFR 50.68, Criticality Accident Requirements, for Spent Fuel Cask Handling,” dated October 8, 2003, supplemented by PG&E Letters (DCL-03-150 and DIL-03-014), “Response to NRC Request for Additional Information Regarding Potential Boron Dilution Events with a Loaded MPC in the DCPP SFP,” dated November 25, 2003.

14. NRC Letter to PG&E, dated January 30, 2004, “Exemption from the Requirements of 10 CFR 50.68(b)(1).”

15. License Amendment Request 04-07, Revision to Technical Specifications 3.7.17 and 4.3 for Cycles 14-16 for a Cask Pit Spent Fuel Storage Rack, PG&E Letter DCL-04-149 dated November 3, 2004.

16. Deleted in Revision 20.

17. Deleted in Revision 20.

18. NRC Generic Letter GL 80-113, “Control of Heavy Loads,” December 22, 1980.

19. PG&E Letter to NRC, “Control of Heavy Loads (NUREG-0612),” September 30, 1982.

20. PG&E Letter to NRC, “Control of Heavy Loads (NUREG-0612),” May 9, 1983.

21. NEI 08-05, “Industry Initiative on Control of Heavy Loads,” July 2008.

22. NRC Safety Evaluation, “NEI 08-05, Revision 0, Industry Initiative on Control of Heavy Loads,” September 5, 2008.

23. PG&E Calculation SAP DIR No. 9000040722, Legacy HID-12, Binder No. MR-11 (S&L Calculation 2008-13003, “Analysis of Postulated Reactor Vessel Head Drop for DCPP,” Proprietary and Confidential, Sargent & Lundy, LLC).

24. Westinghouse Calculation No. A-DP1 -FE-0001, "DCPP Units 1 &2 Spent Fuel Criticality Analysis," September 2001.

25. PG&E Vendor Document 6021773-88, "WCAP-16985-P Revision 2, DCPP Tavg & Tfeed Ranges Program NSSS Engineering Report," April 2009.

26. Holtec Report HI-931 077 Revision 3, "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel storage Racks w/ 5% Enrichment," June 1995.

27. PG&E Letter (DCL-12-117) to NRC, “Summary of Regulatory Commitment Changes January 1, 2011, Through December 31, 2011,” Dated November 20, 2012.

28. NRC Supplemental Safety Evaluation Report (SSER) 31 for Diablo Canyon Power Plant, April 1985.

29. PG&E Calculation SAP Dir No. 9000041782, Legacy / S&L No. 2015-05116, “Analysis of Postulated Reactor Coolant Pump Motor Drop For DCPP,” Proprietary and Confidential, Sargent & Lundy, LLC.

9.1.6 REFERENCE DRAWINGS

Figures representing controlled engineering drawings are incorporated by reference and are identified in Table 1.6-1. The contents of the drawings are controlled by DCPP procedures.

9.2 WATER SYSTEMS

This section describes all of the auxiliary water supply and cooling water systems in the plant except for the fire protection system, which is discussed in Section 9.5.1 and the seawater supply to the service cooling water (SCW) heat exchangers, which is described in Section 10.4.5. Water used in the plant is a combination of processed seawater and well water.

9.2.1 SERVICE COOLING WATER SYSTEM

The SCW system, shown in Figure 3.2-15, is a closed system used to cool equipment in the secondary or steam and power conversion portion of the plant, as well as miscellaneous air conditioning systems. The SCW system is classified as PG&E Design Class II (refer to Section 3.2). Since no PG&E Design Class I components are cooled by the SCW system, complete shutdown of the system does not affect safe operation or shutdown of the reactor. The design requirements for flow are based on the heat load demands of the various components cooled by the SCW system.

9.2.1.1 Design Bases

9.2.1.1.1 General Design Criterion 4, 1967 – Sharing of Systems

The SCW system is not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.2.1.1.2 Service Cooling Water System Function Requirements

(1) Protection from Moderate Energy Pipe Rupture Effects – Outside Containment

SCW system failures are evaluated for the effects of moderate energy pipe failure on PG&E Design Class I equipment to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(2) Protection from Flooding Effects

SCW system header failure is evaluated for the effects of flooding on PG&E Design Class I equipment.

9.2.1.2 System Description

The SCW system is a closed system that supplies buffered cooling water for the following plant equipment:

(1) Post-accident sample room air conditioning

(2) Onsite technical support center (TSC) air conditioning condenser

(3) Personnel access control area air conditioning

(4) Plant air compressors and aftercoolers

(5) Plant air dryer cooler

(6) Main Steam lead, SG blowdown, and feedwater heater 1 outlet header sample coolers

(7) Main turbine lube oil reservoir coolers

(8) Condenser vacuum pump seal water cooler

(9) Electrohydraulic control coolers

(10) Feedwater sample coolers

(11) Heater 2 drain tank pump discharge sample coolers

(12) Heater No. 2 drain pump

(13) Secondary process control room isothermal bath water chiller

(14) Main feedwater pumps 1 and 2 lube oil coolers

(15) Condensate pump motor upper bearing coolers

(16) Condensate booster pump lube oil coolers

(17) Air conditioning for operation ready room (Unit 2 only)

(18) Generator exciter coolers

(19) Fuse wheel cooler

(20) Stator coil cooling water corrosion sampler and sample cooler (Unit 1 only)

(21) Generator seal oil coolers

(22) Isophase bus coolers

The SCW heat exchangers are normally cooled by the circulating water system (CWS) described in Section 10.4.5; however, alternative cooling supplies are available. Makeup water to the system is from the PG&E Design Class II portion of the MWS described in Section 9.2.3. This is controlled automatically by the level in the service water head tank.

9.2.1.3 Safety Evaluation

9.2.1.3.1 General Design Criterion 4, 1967 – Sharing of Systems

The DCPP Unit 1 and Unit 2 SCW systems have the ability to cross-connect through the SCW system headers. The SCW system is PG&E Design Class II and not designed to remain functional when subjected to DE, DDE, or HE seismic loads. No PG&E Design Class I components are cooled by the SCW system. Therefore, safety is not impaired by sharing because the SCW system is not necessary to assure: the integrity of the reactor coolant pressure boundary (RCPB); the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents which could result in the release of substantial amounts of radioactivity.

9.2.1.3.2 Service Cooling Water System Function Requirements

(1) Protection from Moderate Energy Pipe Rupture Effects – Outside Containment

The PG&E Design Class I systems are designed to be protected against the effects of SCW system moderate energy pipe failures to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(2) Protection from Flooding Effects

The low operating pressure and temperature of the SCW system minimizes the probability of line failures. The physical location of the lines and components cooled by the SCW system is such that the failure of a SCW system header would not create an adverse environment for any PG&E Design Class I components.

9.2.1.4 Tests and Inspections

The operating components are in either continuous or intermittent use during normal plant operation, and no additional periodic tests are required. Periodic visual inspections and preventive maintenance are conducted in accordance with normal plant operating practices.

9.2.1.5 Instrumentation Applications

The operation of the system is monitored with the following instrumentation:

(1) High and low level alarms in the service water head tank

(2) Automatic pump start pressure switches on the common pump discharge

(3) Temperature sensing devices at:

(a) Main turbine reservoir lube oil coolers outlet

(b) SCW supply header

(c) Isophase bus cooler fans hot air inlet

(d) Generator exciter and fuse wheel cooler water return

(e) Air compressor cooling water returns

(f) Post-accident sample room air conditioning system inlet and outlet

(g) Onsite TSC air conditioning cooling water inlets and outlets

(h) Personnel access control area air conditioning inlets and outlets

(4) Pressure sensing devices at:

(a) SCW pumps outlet

(b) CCW chemical addition system

(c) SCW heat exchanger inlets and outlets

(d) Main turbine lube oil coolers water supply

(e) Post-accident sample room air conditioning system inlet and outlet

(f) SCW filter inlets and outlets

(g) Onsite TSC air conditioning cooling water inlets and outlets

(h) SCW booster pumps inlet and outlet

9.2.2 COMPONENT COOLING WATER SYSTEM

The CCW system, shown in Figure 3.2-14, is a closed-cycle cooling system that transfers heat from nuclear (primary) plant equipment and other systems/components (refer to Table 9.2-4) during normal plant operation, plant cooldown, and following a LOCA or main steam line break (MSLB) to the auxiliary saltwater (ASW) system. Except for normally closed makeup lines and seal water make-up to the waste gas compressor, there is no direct connection between the CCW system and other systems. The CCW system provides a monitored intermediate barrier between equipment and components handling radioactive fluids and the ASW system.

9.2.2.1 Design Bases

9.2.2.1.1 General Design Criterion 2, 1967 - Performance Standards

The CCW system is designed to withstand the effects of or is protected against natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis, and other local site effects.

9.2.2.1.2 General Design Criterion 3, 1971 - Fire Protection

The CCW system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.2.2.1.3 General Design Criterion 4, 1967 - Sharing of Systems

The CCW systems or components are not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.2.2.1.4 General Design Criterion 11, 1967 - Control Room

The CCW system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.2.2.1.5 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain CCW system variables within prescribed operating ranges.

9.2.2.1.6 General Design Criterion 17, 1967 - Monitoring Radioactivity Releases

The CCW system is designed to provide means for monitoring the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients and from accident conditions.

9.2.2.1.7 General Design Criterion 40, 1967 – Missile Protection

The containment isolation portion of the CCW system is designed to be protected against dynamic effects and missiles that might result from plant equipment failures.

9.2.2.1.8 General Design Criterion 53, 1967 - Containment Isolation Valves

CCW system containment penetrations that require closure for the containment isolation function are protected by redundant valving and associated apparatus.

9.2.2.1.9 General Design Criterion 57, 1967 - Provisions for Testing Isolation Valves

The CCW system provides capability for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

9.2.2.1.10 Component Cooling Water System Safety Function Requirements

(1) Waste Heat Removal

The CCW system is designed to remove waste heat from the nuclear (primary) plant equipment and components during normal plant operation, plant cooldown, and design basis accidents.

(2) Single Failure

The CCW system and ASW system are essentially considered a single heat removal system for the purpose of assessing the ability to sustain either a single active or passive failure and still perform design basis heat removal.

(3) Redundancy

The CCW system components considered vital are redundant.

(4) Isolation

The CCW system includes provision for isolation of system components and may be split into separate trains during long term post-LOCA conditions.

(5) Protection from Missiles

Vital portions of the CCW system are designed, located, or protected against the effects of missiles which may result from plant equipment failure and from events and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(6) Protection Against High Energy Pipe Rupture Effects

Vital portions of the CCW system are designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(7) Protection from Moderate Energy Pipe Rupture Effects – Outside Containment

The outside containment portion of the vital CCW system is designed to be protected against the effects of moderate energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(8) Protection from Jet Impingement – Inside Containment

The inside containment portion of the vital CCW system is designed to be protected against the effects of jet impingement which may result from high energy pipe rupture to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(9) Protection from Flooding Effects – Outside Containment

The outside containment portion of the vital CCW system is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(10) Leak Detection

The CCW system serves as an intermediate system between normally or potentially radioactive systems and the ASW system.

9.2.2.1.11 10 CFR 50.49 - Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

CCW system electric components that require environmental qualification (EQ) are qualified to the requirements of 10 CFR 50.49.

9.2.2.1.12 10 CFR 50.55a(f) - Inservice Testing Requirements

CCW system ASME Code components are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

9.2.2.1.13 10 CFR 50.55a(g) - Inservice Inspection Requirements

CCW system ASME Code components (including supports) are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.2.2.1.14 10 CFR 50.63 - Loss of All Alternating Current Power

The CCW system is required to provide cooling water to the reactor coolant pump (RCP) thermal barriers, the operating CCW pump cooler, the operating centrifugal charging pump coolers, and the seal water heat exchanger in the event of a Station Blackout (SBO).

9.2.2.1.15 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The CCW system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.2.2.1.16 Regulatory Guide 1.97 Revision 3, May 1983 - Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Condition During and Following an Accident

The CCW system provides instrumentation to monitor CCW flow and temperature and containment isolation valve (CIV) position indication on the monitor light box (for applicable CCW valves) during and following an accident.

9.2.2.1.17 NUREG-0737 (Item II.K.3.25), November 1980 – Clarification of TMI Action Plan Requirements

Item II.K.3.25 – Effect of Loss of Offsite AC Power on RCP Seals: The CCW system is designed such that the RCP seals can withstand a complete loss of offsite alternating current power for at least two hours.

9.2.2.1.18 Generic Letter 89-10, June 1989 - Safety Related Motor-Operated Valve Testing and Surveillance

The CCW system safety related and position changeable motor-operated valves (MOVs) meet the requirements of Generic Letter 89-10, June 1989 and associated Generic Letter 96-05, September 1996.

9.2.2.1.19 Generic Letter 89-13, July 1989 - Service Water System Problems Affecting Safety Related Equipment

The CCW system heat exchangers are subject to monitoring and maintenance programs to ensure capability to perform their safety function as an alternative to a testing program. Maintenance practices, operating and emergency procedures, and training ensure effectiveness of these programs.

9.2.2.1.20 Generic Letter 96-06, September 1996 - Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

The CCW system is designed so that it is not subject to the hydrodynamic effects of water hammer, reduced cooling effectiveness due to two phase flow conditions, or thermal overpressurization of isolated piping sections that could affect containment integrity during design basis accident conditions, as described in Generic Letter 96-06, September 1996.

9.2.2.2 System Description

The CCW system is designed to provide cooling water to vital and nonvital components and to operate in all plant operating modes, including normal power operation, plant cooldown, and emergencies, including a LOCA or MSLB.

The CCW system includes three CCW pumps, two CCW heat exchangers, and an internally baffled CCW surge tank as described in Table 9.2-3. The piping system consists of three parallel headers. Two are separable redundant vital service headers A and B, which serve only the unit's engineered safety feature (ESF) equipment and the post-LOCA sample cooler (header A only). A miscellaneous service loop C serves nonvital equipment. Except for normally closed makeup lines and seal water make-up to the waste gas compressor, there is no direct connection between the CCW system and other systems. The equipment cooled is tabulated in Table 9.2-4. Nominal flows for major CCW system operating modes are tabulated in Table 9.2-5.

Cooling water for the CCW heat exchangers is supplied from the ASW system. Together, CCW/ASW support heat transfer to the ultimate heat sink (UHS). The CCW system serves as an intermediate system between the RCS and ASW system, ensuring that any leakage of radioactive fluid from the components being cooled is contained within the plant.

Operation under normal and accident conditions will be as follows:

(1) Normal Operation

During normal operation, all loops are in operation. Two CCW pumps and one or two CCW heat exchangers are in use and are capable of serving all operating components. The third pump and the second heat exchanger generally provide backup during normal operation.

(2) Plant Cooldown

During the cooldown phase of a unit shutdown, all loops are operated with two or three pumps and two heat exchangers used for the removal of residual and sensible heat from the RCS through the RHR system (refer to Figure 3.2-10). If one of the pumps or one of the heat exchangers is inoperative, orderly shutdown is not affected, but the time for cooldown is extended.

(3) Accident Conditions

In the event of a LOCA or MSLB, three CCW pumps are placed in service to provide protection against an active failure. The safety injection (SI) signal initiates an automatic start signal for the standby CCW pump. When containment pressure reaches the high containment pressure setpoint, a Phase A isolation signal is generated and CCW flow to the excess letdown heat exchanger is isolated. When the containment pressure reaches the high-high containment pressure setpoint (containment Phase B isolation), a signal to close the C (non-vital) header isolation valve is generated because components on the C header are not required for post-accident cooling. The portion of the non-vital header that serves the RCPs and vessel support coolers located within the reactor primary shield wall is independently isolated (Phase B), due to its vulnerability during a LOCA, to assure isolation

The CCW system is required to provide cooling water to the ESF pump coolers and the containment fan cooler units (CFCUs) during the injection and recirculation phase of a LOCA and during an MSLB. The CCW system is flow balanced to ensure that adequate flow is maintained to each component. Following the post-LOCA injection phase, the CCW system is realigned for the recirculation phase by valving in the RHR heat exchangers to cool the water collected in the containment sump. Additionally, if a containment Phase B isolation signal has occurred but the C header does not automatically isolate during the injection phase, it is manually isolated prior to realignment for recirculation.

The additional heat load on the CCW system is controlled by plant operators by limiting the heat input equipment (operating CFCUs and RHR heat exchangers) based on the available heat removal equipment (operating CCW heat exchangers and ASW pumps) to prevent the CCW system supply temperature from exceeding its design basis limit.

During long-term post-accident recirculation operation, the CCW system may be manually realigned into two separate redundant loops. Each loop has a pump and a heat exchanger and is capable of fulfilling the minimum long-term cooling requirements. This provides protection against a passive failure in one loop. Should one loop fail, the other loop is unaffected, and the ESF components that it serves remain operative.

Due to its vulnerability to a loss of inventory, the CCW system should be split into separate trains as soon as possible after aligning for long-term post

LOCA recirculation if plant conditions are acceptable. The decision to split CCW trains will be made by the TSC based on the physical integrity of the trains, the availability of active components, and the reliability of power systems. This long-term post-accident alignment provides further assurance of the capability to withstand a passive failure (Reference 8).

Design data for some major CCW system equipment are listed in Table 9.2-3. The CCW system consists of the following major pieces of equipment.

9.2.2.2.1 Component Cooling Water Pumps

The three CCW pumps that circulate CCW through the CCW system are horizontal, double suction, centrifugal units. The pumps operate on electric power from the Class 1E 4.16-kV buses that can be supplied from either normal or emergency sources.

9.2.2.2.2 Component Cooling Water Heat Exchangers

The two CCW heat exchangers are shell and tube type. Seawater circulates through the tube side. The shell is carbon steel, and the tubes are 90-10 Cu-Ni.

9.2.2.2.3 Component Cooling Water Surge Tank

The CCW surge tank, which is connected by two surge lines to the vital headers on the pump suction piping, is constructed of carbon steel. The tank is internally divided into two compartments by a partial height partition to hold two separate volumes of water. This arrangement provides redundancy to accommodate a passive failure when the CCW system is manually realigned into two trains.

The surge tank accommodates thermal expansion and contraction, and in- or out-leakage of water from the system. The tank is normally pressurized with nitrogen to a minimum of 17 psig to provide sufficient static head to prevent boiling and two-phase flow conditions in the CCW to the CFCUs during a postulated large break LOCA coincident with a loss of offsite power. The primary source of nitrogen is the Class II nitrogen system (Reference 7).

A back-pressure regulator is provided downstream of the surge tank vent valve to prevent the pressure in the surge tank from exceeding the desired pressure range. This back-pressure control valve maintains surge tank pressure at all times at its setpoint by relieving excess nitrogen to the atmosphere. The surge tank vent valve closes when a high radiation level is detected by radiation monitors provided in the CCW pump discharge headers. The monitor also actuates an alarm in the control room.

In the event of a low level in the surge tank, makeup water is automatically added to the system through control valves from the MWS (refer to Section 9.2.3).

9.2.2.2.4 Chemical Addition System

The closed chemical addition system supplies various treatment chemical solutions to the CCW. The system contains chemical addition tanks, an injection pump, pressure and flow indication, and a fume hood for personnel protection. The system is common to Unit 1 and Unit 2 CCW and also supplies chemicals to the SCW.

9.2.2.2.5 Component Cooling Water Corrosion Monitor

A corrosion test loop is provided to monitor the effectiveness of the corrosion inhibitor used in the CCW system. The test loop consists of four coupon locations and two spare connections for future use. Test coupons of representative materials are exposed to CCW system conditions for a period of time and then analyzed to determine the overall corrosion rates.

9.2.2.2.6 Residual Heat Removal Heat Exchangers

Control room operated air-actuated valves control the CCW flow to the RHR heat exchangers (refer to Chapter 5) in order to place these components in service during plant cooldown or after a LOCA. The valves, which open on loss of air, are provided with a PG&E Design Class I backup air supply to allow positive operator control after loss of the plant compressed air system.

9.2.2.2.7 Containment Fan Coolers

CCW is supplied to the containment fan coolers (refer to Chapter 6) by the two vital headers. Two fan coolers are on loop A and three on header B. Drain and isolation valves are provided on each side of the fan coolers allowing each cooler to be isolated individually for leakage testing. The flow of CCW through the fan coolers is throttled (position fixed) by a manual valve downstream of each fan cooler to ensure adequate flow to support design basis accident analyses as discussed below. The air-actuated temperature control valves, provided originally to limit the flow to the CFCUs, are not required. Therefore, instrument air to these valves has been isolated and the valves remain in the fully open position.

The above valve alignment assures minimum required flow through each fan cooler under accident conditions without any immediate automatic or operator action. The only required manual action would be header C isolation when transferring to post-LOCA recirculation.

The ability of the CCW system to adequately remove heat from containment without overheating the CCW fluid is demonstrated by several transient analyses. For determining adequate containment heat removal, the minimum CCW flow rate to CFCUs is 1600 gpm to the cooling coils. For determining maximum CCW temperature, the maximum CCW flow rate to CFCU cooling coils is 2500 gpm. Data presented in Tables 9.2-5 and 6.2-26 are nominal data that are enveloped by these extremes. CCW is also supplied to a separate cooling coil located in the CFCU motor enclosure. This cooling flow path is in parallel to the CFCU cooling coil flow path.

Specific conditions, such as inlet and outlet temperatures to the CFCUs, are dynamically calculated and vary over the course of the transient according to the scenario assumptions.

9.2.2.2.8 Reactor Vessel Supports

Cooling water is provided to the RV supports (refer to Chapter 3) to prevent overheating and dehydration of the concrete for the RV support shoes. This is accomplished by the use of water-cooled steel blocks between each of the four vessel support pads and its support shoe. CCW flows in labyrinth flow passages in the blocks providing heat removal sufficient to prevent the concrete from dehydrating.

9.2.2.2.9 Valves

The valves in the CCW system are standard commercial valves constructed of carbon steel with carbon steel, bronze, or stainless steel trim. Since the CCW is normally not radioactive, special features to prevent leakage to the atmosphere are not provided. Self-actuated spring-loaded relief valves are provided for lines and components that may be pressurized to above their design pressure by improper operation or malfunction. The valves associated with CCW pumps, the CCW heat exchangers, large piping and associated instrumentation are located outside the containment and are therefore available for maintenance and inspection during power operation.

CCW valves associated with containment isolation are discussed as a part of the containment isolation system (CIS) (refer to Section 6.2.4).

The equipment vent and drain lines have manual valves that are normally closed (the surge tank vent line has an automatic back-pressure regulator that is also normally closed) unless the equipment is being vented or drained for maintenance or repair operations.

The relief valves on the CCW lines downstream of the sample, letdown, seal water, SFP, and RHR heat exchangers are sized to relieve the volumetric expansion occurring if the exchanger shell-side is isolated and high-temperature coolant flows through the tube side. The relief pressures do not exceed 150 psig.

Relief valves for volumetric expansion are provided on the downstream side of the waste gas compressor heat exchanger and the abandoned in place boric acid and waste evaporator condenser. Relief pressures do not exceed 150 psig. Waste gas compressor relief and evaporator packages relief is back to the return line downstream of the respective shutoff valve.

The relief valve on the CCW surge tank is sized to relieve the maximum flow-rate of water that could enter the surge tank following an RHR heat exchanger tube rupture. The discharge from this valve is directed to the skirted area under the surge tank and then enters a floor drain routed to the auxiliary building sump.

9.2.2.2.10 Piping

All piping components of the CCW system are designed to the applicable codes and standards listed in Table 3.2-3. CCW system piping is carbon steel, with welded joints and flanged connections at components, and designed to USAS B31.7 for Class II and Class III pipe. A molybdate blend solution is added to the CCW as a corrosion inhibitor.

9.2.2.2.11 Component Cooling Water Filter Housing

The CCW side-stream filter is provided in order to remove particulates in the CCW system. The filter housing is designed for 150 psig at 200°F.

9.2.2.2.12 Radiation Monitoring

Leaks in components being cooled are detected by radiation monitors located in the two CCW pump discharge headers. Because the discharge of all three CCW pumps is into these two headers, the flow from any combination of pumps placed in operation is monitored continuously. During normal plant operations, the inservice CCW pump discharge header is determined by which of the two CCW heat exchangers is operating (refer to Figure 3.2-14). A single failure in the radiation monitoring system on the discharge header in operation alarms in the control room. The operator can take action to correct the problem or to put into operation the redundant heat exchanger and header. This action places the redundant radiation monitoring system into operation. During operation with both heat exchangers in service, both radiation monitors are in service continuously. Because of the discharge piping configuration, only the radiation monitor associated with the in-service CCW heat exchanger is sampling flow representative of the bulk system.

The CCW may become contaminated with radioactive water from any of the following sources:

(1) Leakage in any heat exchanger tube or tubesheet in the CVCS, the nuclear steam supply system (NSSS) sampling system, the RHR system, the SFP cooling system, or the gaseous radwaste system

(2) Leakage in a cooling coil for the thermal barrier cooler on a RCP

(3) Leakage in a containment fan cooler coil following an accident

9.2.2.3 Safety Evaluation

Refer to Section 3.1 for a more comprehensive discussion of General Design Criteria applicable to DCPP. In addition to Section 3.1, other sections are referenced, where appropriate, for individual design basis requirements discussed under 9.2.2.3.

9.2.2.3.1 General Design Criterion 2, 1967 - Performance Standards

The buildings that contain the majority of the CCW system SSCs (containment, auxiliary and FHB, and the turbine building) are PG&E Design Class I or QA Class S (refer to Section 3.8). These buildings or applicable portions thereof are designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunami (refer to Section 3.4), external missiles (refer to Section 3.5), earthquakes (refer to Section 3.7), and other appropriate natural phenomena and to protect CCW SSCs, and their safety functions, from damage due to these events. The loss of CCW components that are not contained within these buildings, and are directly exposed to potential wind and tornado loads, has been evaluated. Loss of this equipment does not compromise the capability to safely shut down the plant (refer to Section 3.3.2.3).

The CCW system SSCs important to safety are designed to perform their safety functions under the effects of earthquakes. The PG&E Design Class II portions of CCW (except the chemical addition system) as well as components from other systems served by the non-vital CCW header (header C) have been analyzed to the same seismic requirements as for the CCW PG&E Design Class I components to ensure pressure boundary integrity is maintained. The chemical addition system is not required for the system to perform its safety function and is normally isolated at the PG&E Design Class I to Class II code break boundary.

The makeup water for the CCW surge tank is provided by the MWS. This source is backed up by several alternative sources, including the CST. Seismic design capability of makeup sources is discussed in Section 9.2.3.

The primary source of nitrogen pressurization for the CCW surge tank is the nonseismically-qualified PG&E Design Class II nitrogen system. If this supply is lost, PG&E Design Class I nitrogen is automatically supplied from dedicated bottles. The Class II plant instrument air supply (with a normally closed valve) is also available to provide the required pressurization of the tank.

9.2.2.3.2 General Design Criterion 3, 1971 - Fire Protection

The CCW system is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.2.2.3.3 General Design Criterion 4, 1967 - Sharing of Systems

The design basis of the CCW system does not require sharing of SSCs between Unit 1 and Unit 2 because each unit has its own CCW system. A means to cross-tie the Unit 1 and Unit 2 CCW systems (valving associated with the spare waste gas compressor piping) is available in the event of a loss of surge tank for supplying CCW from a unit with an operating CCW system to a unit with an inoperable system (e.g., to permit shutdown). However, the associated valves to Unit 2 CCW are normally closed and cross-connection is procedurally controlled. Therefore, safety is not impaired by the sharing.

9.2.2.3.4 General Design Criterion 11, 1967 - Control Room

Appropriate CCW system instruments and controls are provided to permit system operation from the control room (refer to Section 9.2.2.5). The CCW pumps are designed to be remotely operated from the hot shutdown panel (HSP) in the event that the main control room is uninhabitable (refer to Section 7.4.2.1.2.2).

9.2.2.3.5 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Required controls for CCW components are provided for system operation. Instrumentation is provided for monitoring CCW system parameters during normal operations and accident conditions (refer to Section 9.2.2.5).

9.2.2.3.6 General Design Criterion 17, 1967 - Monitoring Radioactivity Releases

A radiation monitor associated with each of the two CCW pump discharge headers monitors the CCW system for radioactive in-leakage. Because the discharge of all three CCW pumps is into these two headers, the flow from any combination of pumps placed in operation is monitored continuously.

9.2.2.3.7 General Design Criterion 40, 1967 – Missile Protection

The provisions taken to protect the containment isolation portion of the CCW system from damage that might result from missiles and dynamic effects associated with equipment and high-energy pipe failures are discussed in Sections 3.5, 3.6, and 6.2.4.

9.2.2.3.8 General Design Criterion 53, 1967 - Containment Isolation Valves

The CCW system containment penetrations that are credited as part of the CIS include CCW supply/return for containment fan cooler (Group D), RCP (Group A), and excess letdown heat exchangers (Group C). These lines can be isolated remotely from the control room. The configuration / requirements for each Group are described in Section 6.2.4.2 and a description of the isolation valves / piping configuration for each penetration is provided in Table 6.2-39.

9.2.2.3.9 General Design Criterion 57, 1967 - Provisions for Testing Isolation Valves

CCW system piping that penetrates containment is provided with the capability for leak detection and operability testing. Most of the piping, valves, and instrumentation inside the containment, including the vital components, are located outside the crane wall at an elevation above the water level in containment following an accident. Exceptions are the cooling lines for the RCPs and the RV support which are on miscellaneous nonvital header "C." This location affords radiation shielding which permits maintenance and inspection during power operation if required.

9.2.2.3.10 Component Cooling Water System Safety Function Requirements

A malfunction analysis of pumps, heat exchangers, and valves is presented in Table 9.2-7.

(1) Waste Heat Removal

The CCW system is designed to remove waste heat from nuclear (primary) plant equipment and components during normal plant operation, plant cooldown, and accident conditions. Analytical results show that the CCW system performs adequately during design basis accidents while providing cooling to all safety-related components cooled by CCW. In the event of a LOCA or MSLB, non-vital / unnecessary heat loads are isolated and analyses demonstrate that the CCW system does not exceed its design basis temperature limit under maximum mechanistically calculated heat loads. At least two CCW pumps must be in operation to ensure that the minimum CCW flow rates are achieved.

Safety analyses for containment peak pressure demonstrate that only one ASW pump and one CCW heat exchanger is required to provide sufficient heat removal from containment to mitigate a MSLB or LOCA (refer to Section 6.2). The analyses were performed assuming minimum CCW flow rate to the CFCUs. Other critical assumptions incorporated into those analyses include the CCW flow rate to the CCW heat exchanger, and CCW heat exchanger UA (heat transfer index and area of the heat exchanger, refer to Section 6.2).

Analyses that demonstrate the CCW system does not exceed its design basis temperature limit following a LOCA or MSLB credit one or two ASW pumps, depending on the assumed single failure. A single CCW heat exchanger was assumed to be in service throughout the transient (except as noted in Section 9.2.2.2). These analyses assume single failures that maximize heat input to the CCW system and maximum flow rates consistent with the system flow balance. The limiting post-LOCA injection phase CCW temperature transient is a solid state protection system Train A failure scenario. The limiting post-LOCA recirculation phase CCW temperature transient results from a solid state protection system Train A failure scenario that conservatively assumes that only three CFCUs are in operation during the injection phase. The highest peak CCW temperature following an MSLB results from a split rupture at 30 percent power with the failure of a main steamline isolation valve. All of the limiting CCW temperature analyses assume 64°F ocean water and a single CCW heat exchanger in service. A separate set of analyses assuming a 70°F ocean water temperature credit two CCW heat exchangers in service to address operation with an elevated UHS temperature (Reference 3). Technical Specifications require that the second CCW heat exchanger be placed in service when the UHS temperature is greater than 64°F.

The CCW system is qualified for a maximum post-accident supply temperature of 140°F for a period of up to 6 hours, and a long-term continuous supply temperature of 120°F. Therefore, predicted CCW temperatures during both normal and accident conditions are within the limits of the CCW system temperature qualification.

(2) Single Failure

The CCW system is designed to continue to perform its safety function following an accident assuming a single active failure during the short-term recovery period and either a single active or passive failure during the long-term recovery period. Refer to Section 3.1.1 for a description of DCPP single failure criteria and definition of terms. During normal operation and up to 24 hours after an accident (the short-term recovery period), the CCW headers are crosstied. This configuration will withstand a single active failure without the loss of safety function. For a passive failure (up to a 200 gpm leak for 20 minutes), operator mitigation action (consisting of valve manipulations) is credited to stop leaks (refer to Table 9.2-7). The CCW headers are evaluated for separation, per procedure, during long term post LOCA recirculation. When separated during the remainder of the recovery period, this configuration will withstand either an active failure or a passive failure without the loss of safety function.

(3) Redundancy

The CCW system components that are considered vital are redundant. The redundant vital CCW headers served by headers A and B supply cooling water to the containment fan coolers, the RHR heat exchangers, each redundant set of ESF pumps (SI, centrifugal charging, and RHR), and the CCW pumps. The automatic flow cutoff (closure of inlet valve) of the non-vital header (served by header C) on containment Phase B isolation is not redundant and operator action is credited prior to realignment for recirculation if automatic isolation fails. However, analyses that assume a failure to isolate during the LOCA injection phase, or during an MSLB, demonstrate that CCW heat removal capability continues to support post-accident cooling requirements without exceeding design temperature limitations (140°F peak and 120°F for six hours).

The three CCW pump motors are on separate Class 1E 4.16-kV buses that have diesel generator standby power sources. The CCW surge tank, which is connected by two surge lines to the vital headers near the pump suction, is internally divided into two compartments by a partial height partition to hold two separate volumes of water. This arrangement provides redundancy to accommodate a passive failure when the CCW system is manually realigned into two trains. In the event of loss of the PG&E Design Class II nitrogen supply, PG&E Design Class I nitrogen is supplied from dedicated bottles, or the plant instrument air system will be available to provide the required pressurization of the tank.

Makeup water is supplied to the CCW system through two redundant makeup valves feeding into the two redundant CCW surge lines, described in Section 9.2.3 and schematically shown in Figure 3.2-14. These air-actuated level control valves (LCVs) open automatically when surge tank level decreases below the associated setpoint, and normal operating conditions for the MWS allow immediate makeup to the CCW system through the makeup valves whenever they open.

Redundant radiation monitors are provided in the system for detection of radioactivity entering the CCW system from the RCS and its associated auxiliary systems.

(4) Isolation

The CCW piping design includes valving for isolating cooling water flow associated with individual components and for complete isolation of a header. In addition to facilitating maintenance and testing, valving is used to: 1) stop leaks from / into the CCW system, 2) prevent an unmonitored release in the event of a radiation monitor alarm, 3) isolate non-vital header C to accommodate higher heat load under accident conditions, 4) separate CCW into two trains to enhance protection against passive failure for long term accident recovery.

Leaks from the CCW system arise from open drain valves or severed piping, ruptured heat exchanger tube, or other malfunction. The location of a leak can be determined by sequential isolation or visual inspection of equipment and hence stopped by closing the appropriate valve(s). Refer to Table 9.2-7 for an evaluation of leakage from the system.

Leaks into the CCW system can result from heat exchanger tube failures. For the RCP thermal barrier, the system design is to contain the in-leakage to the CCW system within the containment structure. This is accomplished by closure of the outboard CIV associated with return of CCW from all RCP thermal barriers on a high flow signal. All piping and valves required to contain this in-leakage are designed for an RCS design pressure of 2485 psig. Should a coolant leak develop from the postulated failure mode that does not result in automatic flow isolation, the corresponding increase in CCW volume is accommodated by the relief valve on the CCW surge tank. The four relief valves on the CCW returns from thermal barriers are sized to relieve volumetric expansion and are set to relieve at RCS design pressure.

Table 9.2-6 shows components in the CCW system with a single barrier between CCW and reactor coolant water.

As shown in Table 9.2-6, the pressure and temperature design requirements of the barriers in the RHR heat exchangers, the letdown heat exchanger, and the seal water heat exchanger are less than the RCS pressure and temperature during full power operation. For the letdown heat exchanger and the seal water heat exchanger, this condition results because the pressure and temperature of the reactor coolant water are reduced to the values shown in the table before the flow reaches the components.

In the case of the RHR heat exchangers and RHR pumps (seal coolers), the RCS pressure and temperature are reduced to less than or equal to 390 psig and 350°F before the RHR system is brought into service to complete the cooldown of the reactor. The RHR system is protected from overpressurization as discussed in Section 5.5.6.4.10. The controls and interlocks provided for the isolation valves between the RCS and the RHR system are described in Section 7.6.2.1.

Tube failure in components with design pressures and temperatures less than RCS design condition may initiate a leak into the CCW system. The radioactivity associated with the reactor coolant would actuate the CCW system radiation monitor. The monitor in turn would annunciate in the control room and close the vent valve located just upstream of the CCW surge tank back-pressure regulator to prevent the regulator from venting after sensing high radiation. The operator would then take the appropriate action to isolate the failed component. In addition to the radiation monitoring system, the operator would also receive high level and high-pressure alarms from the surge tank as it filled. If the in-leakage continued after the vent valve closed, the surge tank pressure would increase until the high surge tank pressure alarm was received and then the relief valve setpoint was reached. The relief valve on the surge tank will protect the system from overpressurization. The maximum postulated in-leakage into the CCW system is based on an RHR heat exchanger tube rupture. The relief valve will accommodate this flow. Relief valve discharge from the CCW system surge tank is routed to the skirted area under the surge tank, which then enters a floor drain routed to the auxiliary building sump. Refer to Table 9.2-7 for an evaluation of leakage into the CCW system.

Under accident conditions, automatic isolation is initiated for CCW flow to the excess letdown heat exchanger on a containment Phase A isolation signal and for the C (non-vital) header on a containment Phase B isolation signal. Independently, the portion of the C header serving the RCPs and vessel support coolers is also isolated.

(5) Protection from Missiles

The provisions taken to protect the vital CCW system from missiles resulting from plant equipment failures and from events and conditions outside the plant are discussed in Sections 3.5.

(6) <u>Protection Against High Energy Pipe Rupture Effects</u>

The plant is designed so that a postulated piping failure will not cause the loss of needed functions of safety related systems and structures that would prevent safe shutdown. The measures taken in design and construction of the plant for protection of the vital CCW system against dynamic effects associated with a postulated rupture of high-energy piping both inside and outside containment are discussed in Section 3.6.

CCW SSCs important to safety are designed, located, or protected against dynamic effects. With respect to post-accident conditions in containment, most of the piping, valves, and instrumentation are located outside the crane wall at an elevation above the water level in containment following an accident. Exceptions are the cooling lines for the RCPs and the RV support which are on portions of header C inside the crane wall. The vital portions of the CCW system within the containment are protected during accidents from dynamic effects associated with accidents by routing piping away from high energy lines and from credible internal missiles by separation / barriers (refer to Section 6.2.4.4.6).

(7) <u>Protection from Moderate Energy Pipe Rupture Effects – Outside Containment</u>

The provisions taken to provide protection of the vital portion of the CCW system located outside containment from the effects of moderate energy pipe failure are discussed in Section 3.6.

(8) <u>Protection from Jet Impingement – Inside Containment</u>

The provisions taken to provide protection of the vital portion of the CCW system located inside containment from the effects of jet impingement which may result from high energy pipe rupture are discussed in Section 3.6.

(9) <u>Protection from Flooding Effects – Outside Containment</u>

The provisions taken to provide protection of the vital portion of the CCW system from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

The CCW pumps, heat exchangers and associated valves, and all of the large piping and instrumentation are located outside containment. Each CCW pump is protected against flooding due to rupture of another because they are located in separate compartments with a raised curb in the doorway to prevent water in the rest of the auxiliary building from entering the compartment. Check valves are provided on each pump discharge to prevent back leakage into a compartment from an operating pump.

Flooding of the CCW heat exchangers is highly improbable because of their location on the turbine building ground level where there are large door openings to allow water to run out, several floor drains, sumps, and a large condenser pit below the elevation of the heat exchangers. Based on this, operation of the heat exchangers would not be impaired by flooding.

(10) Leak Detection

Leakage from the CCW system can be detected by a decreasing level in the CCW surge tank. Using the tank geometry, an estimate of leakage rate can be determined by timing the change in indicated level. A maximum 200 gpm leak or rupture is postulated. Refer to Table 9.2-7 for a discussion of leakage from the system.

A radiation monitor associated with each of the two CCW pump discharge headers is provided for the CCW system to detect radioactivity entering the CCW system from the RCS and its associated auxiliary systems. In-leakage from components being cooled is detected by a radiation monitor associated with each of the two CCW pump discharge headers. Because the discharge of all three CCW pumps is into these two headers, the flow from any combination of pumps placed in operation is monitored continuously. Leaks can also be detected by surge tank level instrumentation and alarms.

9.2.2.3.11 10 CFR 50.49 - Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

CCW system SSCs required to function in harsh environments under accident conditions are qualified to the applicable environmental conditions to ensure that they will continue to perform their safety functions. Section 3.11 describes the DCPP EQ Program and the requirements for the environmental design of electrical and related mechanical equipment. The affected components include valves, switches and flow transmitters and are listed on the EQ Master List.

9.2.2.3.12 10 CFR 50.55a(f) - Inservice Testing Requirements

The inservice testing (IST) requirements for CCW system components are contained in the IST Program Plan and comply with the ASME Code for Operation and Maintenance of Nuclear Power Plants.

9.2.2.3.13 10 CFR 50.55a(g) - Inservice Inspection Requirements

The ISI requirements for CCW system components are contained in the ISI Program Plan and comply with the ASME BPVC Section XI.

9.2.2.3.14 10 CFR 50.63 - Loss of All Alternating Current Power

The CCW system provides for safe shutdown and cooldown of the reactor by removing heat from safety-related system components after a SBO. The CCW system is used to cool the RCP provides cooling water in support of safe shutdown of the plant (Mode 3) following a SBO. Using a single CCW pump and heat exchanger, the system provides cooling water to the RCP thermal barriers to prevent overheating and degradation of the RCP seals following an SBO cooling water to the operating CCW pump and centrifugal charging pump coolers to support proper pump operation, and cooling water to the seal water heat exchanger in the RCP seal leakoff flowpath.

The CCW pumps can be provided with alternate ac (AAC) power within 10 minutes of an SBO event. Refer to Section 8.3.1.6 for further discussion of station blackout.

9.2.2.3.15 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The CCW System is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.2.2.3.16 Regulatory Guide 1.97 Revision 3, May 1983 - Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Condition During and Following an Accident

CCW post-accident instrumentation for meeting Regulatory Guide 1.97, May 1983 guidelines consist of flow indication for CCW supply headers A & B, temperature indication for each CCW heat exchanger outlet, and CIV position indication on the monitor light box for applicable CCW valves (refer to Section 7.5.3.6).

9.2.2.3.17 NUREG-0737 (Item II.K.3.25), November 1980 – Clarification of TMI Action Plan Requirements

Item II.K.3.25 - Effect of Loss of Alternating Current Power on Reactor Coolant Pump Seals: Required confirmation that RCP thermal barriers can withstand a loss of CCW cooling water to the RCP seal coolers due to a loss of ac power for at least two hours. This requirement is accommodated because the CCW pumps are supplied from Class 1E buses that have emergency on-site backup power (refer to Section 9.2.2.2.1). The associated CIVs are also supplied with emergency on-site backup power or are check valves (refer to Section 6.2.4.2).

9.2.2.3.18 Generic Letter 89-10, June 1989 - Safety Related Motor-Operated Valve Testing and Surveillance

CCW system MOVs are addressed by the DCPP MOV Program Plan, and the plan applies the recommendations of Generic Letter 89-10, June 1989 and associated Generic Letter 96-05, September 1996.

9.2.2.3.19 Generic Letter 89-13, July 1989 - Service Water System Problems Affecting Safety Related Equipment

The applicable recommendations of Generic Letter 89-13, July 1989 for ongoing surveillance and control have been applied to the CCW system, including a monitoring program combining flow testing, trending, inspection, and frequent preventive maintenance. Corrosion inhibitors and additives to prevent biofouling are included as part of preventive maintenance. The CCW heat exchangers provide pressure differential indication in the control room to alert operators to the need for cleaning and sample coupons are used to assess conditions and effectiveness of the program.

9.2.2.3.20 Generic Letter 96-06, September 1996 - Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions

Generic Letter 96-06, September 1996 identified the potential for waterhammer or two-phase flow in the portion of the CCW system serving the CFCUs and for overpressurization of piping in systems that penetrate containment during accident conditions. The CCW nitrogen pressurization system was installed in response to the waterhammer concern to mitigate the possibility of flashing and subsequent waterhammer.

Subsequent review concluded that a limited amount of cavitation was possible during normal operation in the CCW flow downstream of the exit from the CFCUs, but that post-accident conditions would not result in a significant increase in the condition. Because CCW flow balance, and thus operability, is not affected, there is no impact on the ability of the CCW system to perform its design basis function.

A comprehensive review identified all containment mechanical piping and tubing penetrations and isolated piping segments inside containment. For CCW, other than the nitrogen pressurization system, no other actions were required to be taken.

9.2.2.4 Tests and Inspections

The active components of the CCW system are in either continuous or intermittent use during normal plant operation, and no additional periodic tests are required. Periodic visual inspections and preventive maintenance are conducted in accordance with normal plant operating practice.

9.2.2.5 Instrumentation Applications

The operation of the system is monitored with the following major or vital instrumentation:

(1) Temperature detectors at the inlet and at the outlet of each CCW heat exchanger, with control room temperature indication and alarm for heat exchanger outlet high/low temperatures

(2) A control room flow indicator and low flow alarm for each header

(3) Low-pressure switches with alarms and auto pump start near the inlet to each vital supply header

(4) Radiation monitor and alarm in the two pump discharge headers

(5) Control room level indicator and high/low level alarm for each half of the CCW surge tank

(6) Flow indication, temperature indication, or pressure indication on the equipment return lines

(7) Surge tank low and high pressure alarms in the control room

(8) Valve position indications in the control room

Design flowrates for normal, loss of coolant, and cooldown conditions are listed in Table 9.2-5.

9.2.3 COMMON MAKEUP WATER SYSTEM

The MWS, shown in Figure 3.2-16, supplies demineralized makeup water of the quality and quantity necessary for normal reactor coolant services, secondary system makeup, firewater, and miscellaneous plant uses. The system has the capacity necessary to meet the water requirements of a cold plant shutdown and subsequent startup from cold conditions at a time late in core life. The MWS provides makeup to the CCW surge tank. The MWS also supplies water to the CST, which provides a supply of water for the auxiliary feedwater (AFW) system.

For discussion specific to the condensate storage facilities (including the Unit 1 and Unit 2 CSTs and the common transfer tank), refer to Section 9.2.6. Refer to Section 6.5 for discussion regarding the AFW system. Refer to Section 9.2.2 for discussion regarding the CCW system.

9.2.3.1 Design Bases

9.2.3.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the MWS is designed to withstand the effects of, or is protected against, natural phenomena such as earthquakes, tornadoes, flooding, winds, tsunamis, and other local site effects.

9.2.3.1.2 General Design Criterion 3, 1971 – Fire Protection

The PG&E Design Class I portion of the MWS is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.2.3.1.3 General Design Criterion 4, 1967 – Sharing of Systems

The MWS is not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.2.3.1.4 General Design Criterion 11, 1967 – Control Room

The PG&E Design Class I portion of the MWS is designed to or contains instrumentation and controls that support actions to maintain and control the safe operational status of the plant from the control room or from a remote location if control room access is lost due to fire or other cause.

9.2.3.1.5 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain MWS variables within prescribed operating ranges.

9.2.3.1.6 General Design Criterion 67, 1967 – Fuel and Waste Storage Decay Heat

The PG&E Design Class I portion of the MWS from the CST to the SFP provides a reliable and adequate method to deliver makeup water for decay heat removal to prevent damage to the fuel in the SFP that could result in radioactivity release to plant operating areas or the public environs.

9.2.3.1.7 Makeup Water System Safety Function Requirements

(1) <u>Protection from Missiles</u>

The PG&E Design Class I portion of the MWS is designed to be protected against the effects of missiles that may result from equipment failures and from events and conditions outside the plant.

(2) <u>Internal Flooding Protection</u>

Rupture of the PG&E Design Class II portion of the MWS will not impact PG&E Design Class I SSCs required for safe (cold) shutdown.

(3) Makeup Capabilities

The MWS is designed to provide a PG&E Design Class I flow path for makeup water from the CST to the CCW surge tank.

(4) Water Purification

The MWS is designed to filter and demineralize water to provide water that meets the quality requirements of the primary system.

9.2.3.1.8 10 CFR 50.55a(f) – Inservice Testing Requirements

The PG&E Design Class I MWS ASME Code pumps and valves are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

9.2.3.1.9 10 CFR 50.55a(g) – Inservice Inspection Requirements

The PG&E Design Class I MWS ASME Code components are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.2.3.1.10 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The MWS is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.2.3.1.11 Safety Guide 13, March 1971 – Fuel Storage Facility Design Basis

The MWS is designed to provide a PG&E Design Class I flow path for makeup water from the CST to the SFP in accordance with Regulatory Position 8 of Safety Guide 13, March 1971.

9.2.3.1.12 IE Bulletin 80-10, May 1980 – Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment

The MWS is designed to maintain control over boundaries between potentially radioactive and nonradioactive portions of the system to prevent the spread of contaminated water into the nonradioactive portions of the system.

9.2.3.2 System Description

The MWS provides the following levels of water quality:

(1) Raw reservoir water

(2) Demineralized water

The MWS makes use of well water and seawater as sources of raw water. The well water is filtered and then discharged to the raw water storage reservoirs by a rental pretreatment system. The seawater is treated in the rental seawater reverse osmosis system and then pumped to the raw water storage reservoirs.

The seawater evaporators are abandoned in place. The water quality produced by the rental makeup water treatment system (MWTS) meets the specification of various plant operating services, which fall under the following categories:

(1) Makeup water for the primary system

(2) Makeup water for the secondary system

(3) Makeup water for the CCW system and SCW system

(4) Water in adequate quantity for fire fighting

(5) Water supply to the AFW system

(6) Provide an adequate reserve of water for startup and upset conditions for the secondary systems

(7) Supply water for dilution, flushing, and cleanup

Seawater is filtered, sterilized, and desalinated by the seawater reverse osmosis system. The desalinated product water is pumped to a 5.0-million gallon open reservoir system with plastic lined concrete walls.

The well water is pumped to a 100,000-gallon raw water storage tank. From the raw water storage tank, the water is processed through the pretreatment system and then discharged to the open raw water storage reservoirs. The reservoir water is treated with sodium hypochlorite to retard algae growth.

The reservoir water is treated by the MWTS. The system is capable of producing up to 600 gpm of deoxygenated/demineralized water for makeup to the CST or primary water storage tank (PWST). The MWTS consists of reverse osmosis, vacuum deaerator, and mixed-bed demineralizers.

The water produced by the MWTS is distributed to the CST, PWST or transfer tank for storage. The CST is used to supply the secondary system makeup, the AFW system, the auxiliary boiler, and the CCW system.

9.2.3.2.1 Raw Water Storage Reservoirs

The raw water storage reservoirs have a combined capacity of 5.0 million gallons. They have concrete-lined walls and are primarily intended to serve as fresh water storage for fire protection. The raw water storage reservoirs also serve as a source to the MWS and the AFW pumps, providing a large water storage reserve when the raw water supply is lower than the MWS demand.

9.2.3.2.2 Transfer, Distribution, and Storage

Most of the piping in the MWS, except the MWTS piping, is constructed in accordance with ANSI B31.1-1967, except the lines supplying water to the firewater pumps header, AFW pumps, and CCW system. The design classifications for these various SSCs are discussed in Table 3.2-3.

The water for decay heat removal by the AFW pumps is reserved in the CSTs (one for each unit) and supplied to the pump suction through PG&E Design Class I piping. Refer to Section 9.2.6.2 for the capacity of the CST and to Section 6.5.3.7 for additional information on usable inventory. The raw water storage reservoir is also used as a backup source for the AFW pumps.

There is no direct connection between the raw water supply header in the plant and the CST such that any single failure of a component could cause the loss of both CST AFW and reservoir water. A failure of the normal supply header from the CST to the AFW pumps would require opening the manually-operated valves to use the raw water storage reservoirs as a source of AFW. Check valves prevent back-flow of raw water through the failed header. Back-flow of CST water from the AFW pumps suctions through the connections to a postulated break in the raw water supply header is prevented by check valves and normally closed manually-operated stop valves.

The 200,000 gallon PWST is diaphragm-sealed to maintain the low oxygen content required for the reactor makeup water. For a description of reactor coolant water chemistry refer to Section 5.2.2.3.4. Design and operating parameters of the PWST are given in Table 9.2-9.

The PWSTs have been designed and erected to PG&E Design Class II standards and should contain only highly purified water.

9.2.3.2.2.1 Component Cooling Water System Makeup Capabilities

The available sources of makeup water to the CCW system are listed below with their respective makeup capacities. The design classification for each of the tanks and pumps is identified in Table 3.2-3.

Source	Pumps

The following makeup water source and pumps provide a PG&E Design Class I flow path for MWS supply to the CCW surge tank:

(1) CST	Makeup water transfer pumps, two 250 gpm pumps

The following makeup water sources and pumps provide alternative methods for MWS supply to the CCW surge tank:

(2) Transfer tank	Makeup water transfer pumps, two 250 gpm pumps
(3) PWST (Unit 1)	Primary water makeup pumps (Unit 1), two 150 gpm pumps
(4) PWST (Unit 2)	Primary water makeup pumps (Unit 2), two 150 gpm pumps
(5) Firewater tank, 300,000 gallons	Makeup water transfer pumps, two 250 gpm pumps
(6) MWTS can supply water to CCW system directly or via other storage tanks (CSTs, PWSTs, transfer tank and firewater tank)	Makeup water transfer pumps, two 250 gpm pumps Primary water makeup pumps (Unit 1 and Unit 2), four 150 gpm pumps

Flowpaths associated with the source and pump combinations 2, 3, 4, 5, and 6 are not completely PG&E Design Class I, but the number of methods does provide considerable redundancy in backup provisions for makeup water to the CCW system.

9.2.3.3 Safety Evaluation

9.2.3.3.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portions of the MWS which are located within the PG&E Design Class I auxiliary building are protected from the effects of natural phenomena ensuring their safety functions will be performed. The auxiliary building, or applicable portions thereof, is designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), and earthquakes (refer to Section 3.7).

The PG&E Design Class I portions of the MWS are designed to remain functional when subjected to DE, DDE, and HE seismic loads (refer to Section 3.2.2.4.1). The seismic requirements are defined in Sections 3.7 and 3.10, and the provisions to protect the system from seismic damage are discussed in Sections 3.7, 3.9, and 3.10.

Certain PG&E Design Class I portions of the MWS are located outside and are vulnerable to tornadoes. However, they are not required for plant safe shutdown (refer to Sections 3.3.2.5.2.2 and 3.3.2.5.2.9).

Leakage from the PWSTs due to a tornado or missile-induced damage will not result in the flooding of PG&E Design Class I equipment in the auxiliary building since essentially watertight cover plates are installed over the pipe entranceway from each tank into the auxiliary building. Gross leakage from the tanks can be detected by level indication or visual inspection. The tank level is continuously monitored by the plant computer, and a high or low water level will initiate a control room alarm.

9.2.3.3.2 General Design Criterion 3, 1971 – Fire Protection

The MWS is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.2.3.3.3 General Design Criterion 4, 1967 – Sharing of Systems

The CSTs for Unit 1 and Unit 2 are cross-connected so that additional makeup is available from the other unit if required. Manual isolation valves and administrative controls are used to prevent unwanted flow between the Unit 1 and Unit 2 CSTs.

The two PG&E Design Class I makeup water transfer pumps are shared between Unit 1 and Unit 2. Each pump can draw water from the shared fire water storage tank, the shared transfer tank, or the Unit 1 or Unit 2 CSTs. The discharge from the makeup water transfer pumps can be directed to Unit 1 or Unit 2.

A portion of the suction piping to each of the three AFW pumps for each unit is common to the two feedwater sources, the raw water storage reservoirs, and the CST. A failure in this portion of the suction piping to the AFW pumps could draw from both sources. However, the manual stop valve to the raw water supply header is normally closed so the supply of raw reservoir water to the other unit would be unaffected by such a failure. In the affected unit, only one type of AFW pump, either turbine-driven or motor-driven, would be made inoperable by such a failure.

The MWTS is common for both units. No PG&E Design Class I systems are dependent on the output of the MWTS for their operation.

Water can be transferred between the Unit 1 and Unit 2 PWSTs and the tanks can be cross-tied to allow the two tanks to act as one source. The PWSTs are PG&E Design Class II and are not the seismically qualified sources for providing makeup water to the CCW surge tank or SFP. Manual isolation valves and administrative controls are provided to ensure that the sharing of the Unit 1 and Unit 2 PWSTs does not impair the safety of either unit.

In the event that the makeup water transfer pumps are not operable, the MWTS equipment can provide demineralized water to the SFP. Makeup water to the CCW surge tank can be provided using water from the PWST using the primary water makeup pumps. Through the use of administrative controls, the use of shared makeup water components does not impair the safety of either unit.

9.2.3.3.4 General Design Criterion 11, 1967 – Control Room

The PG&E Design Class I portion of the MWS is designed to support actions to maintain and control the safe operational status of the plant from the control room.

The operation of the makeup water transfer pumps is controlled from the control room.

No PG&E Design Class I MWS components are required to be operated from the control room or remotely for safe shutdown.

Refer to Section 9.2.3.3.7 for a description of the instrumentation used to monitor the CCW surge tank makeup valves.

9.2.3.3.5 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation is provided as required to monitor and maintain the MWS variables within prescribed operating ranges. Instrumentation is installed to measure the water level inside the PG&E Design Class II PWST. The water levels in the two raw water storage reservoirs are monitored through the use of instrumentation that indicates the water levels on the main control board and the HSP. Local instrumentation, in the form of pressure indicators and flow indicators, is provided to monitor the operational status of the MWS. Controls to start and stop the MWS transfer pumps are located on the main control board. Controls to start and stop the primary water makeup pumps are provided at a local control board.

Refer to Section 9.2.3.3.8 for discussion regarding instrumentation used to meet IST requirements.

9.2.3.3.6 General Design Criterion 67, 1967 – Fuel and Waste Storage Decay Heat

The MWS provides a PG&E Design Class I flow path for makeup water from the CST to the SFP to ensure adequate decay heat removal in the event of a failure of the SFP cooling and cleanup system. Makeup water to the SFP is supplied by the MWS as described in Section 9.1.3.3.9.

9.2.3.3.7 Makeup Water System Safety Function Requirements

(1) Protection from Missiles

The design function of the PG&E Design Class I portion of the MWS will not be prevented by internal missiles and/or missiles resulting from external events. Provisions taken to protect the PG&E Design Class I portion of the MWS from the effects of missiles are discussed in Section 3.5.

(2) Internal Flooding Protection

Although the raw water storage reservoirs are not PG&E Design Class I items, their location, on a bench excavated into the ridge above the power plant at elevation 310 feet, as shown in Figure 1.2-1, poses a small potential flood risk to the site. Since a portion of the reservoir bench drains toward the power plant, the raw water storage reservoirs were lowered by excavating the basin entirely in rock, eliminating the risk of flooding due to dike failure. The discussion of slope stability in Section 2.5.6 provides assurances that the slope between the power plant yard and the reservoir bench will not fail. Slope failure in any other direction, which results in a reservoir rupture will release the water into the Diablo Canyon.

The discussion of flooding in Section 2.4.4.5 provides assurance that the drainage capacity of the Diablo Canyon is sufficient to pass the entire volume of the raw water storage reservoirs safely by the plant in approximately 1 minute. The raw water reservoirs have a plastic lining on reinforced concrete to prevent leakage. The water levels in the raw water reservoirs are indicated in the control room and at the HSP and low levels are annunciated in the control room.

The two pipelines (12 and 6 inches) between the raw water storage reservoirs and the plant have been examined for their potential for flooding. The maximum combined flow from these lines if ruptured would be 7000 to 8000 gallons per minute. This flow-rate would be intercepted by the site storm drainage systems and diverted from PG&E Design Class I equipment.

The 8 inch raw water supply header in the auxiliary building presents the potential flooding source with the largest volume of water (the raw water storage reservoirs). Flooding of the auxiliary building from this header would be recognized by the annunciation of the auxiliary building sump high-level alarm. The flow would be terminated by an operator using the appropriate manual isolation valve. A volume of 345,000 gallons in the auxiliary building pipe tunnel for sump overflow storage is available to receive water flooding from this source. This storage capacity allows the operator sufficient time to close the stop valve in the yard to prevent overflooding.

The flow rate of water flooding from the raw water supply header in the auxiliary building is defined by moderate energy line break (MELB) analysis criteria, and is bounded by that from other high energy line break (HELB)/MELB sources.

(3) Makeup Capabilities

Refer to Section 9.1.3.3.9 for discussion regarding the use of the MWS to provide makeup water to the SFP.

Normal operating conditions for the MWS allow immediate makeup to the CCW system through the makeup valves whenever they open (shown in the CCW system piping schematic, refer to Figure 3.2-14). These air-actuated LCVs open automatically in the event of low level in the CCW surge tank. They close automatically when the normal operating level in the surge tank is restored or on loss of air. Opening these valves is annunciated in the control room, indicating that CCW system makeup is required. Makeup water to the CCW system is normally supplied from the transfer tank through the makeup water transfer pumps. The MWS supplies water to the 150,000 gallon transfer tank.

The most demanding transient on the water inventory of the condensate system and the CST is a natural circulation cooldown. After such a transient, as much as 225,000 gallons available remain in the CST for makeup to the CCW system. This inventory provides for more than one complete refill of the CCW system. One refill is considered adequate makeup reserve capacity. The nozzle for MWS on the CST draws water from a level above the CST water volume dedicated as a source of AFW supply water. Nozzles for non-seismically qualified PG&E Design Class II systems are attached to the CST at the same elevation as the MWS. The use of CST water for AFW purposes or the failure of non-MWS piping attached to the CST could lower the water inside the CST to a level below the MWS nozzle. In such cases, an alternative water source would need to be used. The firewater tank, which is also seismically qualified, can be aligned as an additional makeup water supply for the CCW system.

The makeup water from the CSTs, both Unit 1 and Unit 2, is pumped from the tank to the CCW system by the PG&E Design Class I makeup water transfer pumps. All piping and valves in the makeup path from the CSTs (including their cross-connections) through the makeup water transfer pumps up to and including the makeup valves on the CCW system lines, are PG&E Design Class I. Two redundant, full capacity, makeup water transfer pumps are each capable of delivering approximately 250 gpm makeup to the CCW system. Each pump is powered from the Class 1E 480-V electrical buses, which are energized by either normal sources or the emergency diesel engine-generator units. A 250 gpm makeup rate is considered to be greater than any credible leakage from the CCW system during normal operation or post-accident injection. This conclusion is based on the low operating pressures and the protection afforded large piping and equipment and the flow rate of the largest postulated leak.

Makeup water for the CCW system can be provided from various water sources and through various pumps and flowpaths in the MWS. Refer to Section 9.2.3.2.2.1 for a list of available alternative PG&E Design Class I and PG&E Design Class II sources and pumps that the MWS can utilize to deliver water to the CCW surge tank.

Refer to Section 9.2.6.2 for the capacity of the CST and to Section 6.5.3.7 for additional information on usable inventory.

(4) Water Purification

The system supplies makeup water of the quality and quantity necessary for normal reactor coolant services, secondary system makeup, and miscellaneous plant uses.

The PWST, which has a diaphragm seal to minimize O_2 contact, supplies water for the primary system.

9.2.3.3.8 10 CFR 50.55a(f) – Inservice Testing Requirements

The PG&E Design Class I MWS pumps and valves comply with the ASME code for Operation and Maintenance of Nuclear Power Plants and are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

Instrumentation is installed to measure the flow rate of the PG&E Design Class I makeup water transfer pumps for IST purposes.

9.2.3.3.9 10 CFR 50.55a(g) – Inservice Inspection Requirements

The PG&E Design Class I MWS ASME BPVC Section XI components are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.2.3.3.10 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The MWS System is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.2.3.3.11 Safety Guide 13, March 1971 – Fuel Storage Facility Design Basis

Regulatory Position 8:

The MWS provides a PG&E Design Class I flow path for makeup water from the CST to the SFP, in accordance with Regulatory Position 8 of Safety Guide 13, March 1971. Refer to Section 9.1.3.3.9 for discussion regarding the use of the MWS to provide makeup water to the SFP.

9.2.3.3.12 IE Bulletin 80-10, May 1980 – Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment

Isolation valves and administrative controls are used to prevent contamination of the CCW. Water from the PWSTs is not used for normal makeup to the CCW system as a result of IE Bulletin 80-10, May 1980. The PWST makeup to CCW system isolation valve is normally locked closed. The lock was installed to preclude opening the valve and contaminating the CCW system with tritiated water. If the valve is to be opened, the plant operator must obtain concurrence from the chemistry and radiation protection group.

9.2.3.4 Tests and Inspections

The operating components of the MWTS are in either continuous or intermittent use during normal plant operation and no additional periodic tests are required. Periodic visual inspections and preventive maintenance are conducted in accordance with plant procedures for plant controlled distribution system components, or in accordance with vendor procedures for the vendor-owned water treatment facilities.

9.2.3.5 Instrumentation Applications

Refer to Section 9.2.3.3.5 for discussion on the instrumentation used to monitor the function of the MWS.

The vendor-owned makeup water treatment facility is equipped with dissolved oxygen, conductivity, silica, and organic carbon monitoring instrumentation. In the event the product from the makeup water plant exceeds the values established in plant procedures, the product is automatically diverted to the raw water storage reservoirs until corrected.

9.2.4 POTABLE WATER SYSTEM

There is no separate potable water system. Potable water is supplied by the domestic water system (DWS) as discussed in Section 9.2.8.

9.2.5 ULTIMATE HEAT SINK

The UHS dissipates residual heat after normal and emergency shutdown conditions. The following sections provide information on (a) design bases, (b) system description, (c) safety evaluation, (d) tests and inspections, and (e) instrumentation applications.

9.2.5.1 Design Bases

9.2.5.1.1 General Design Criterion 2, 1967 - Performance Standards

The UHS is designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.2.5.1.2 General Design Criterion 4, 1967 - Sharing of Systems

The UHS (Pacific Ocean) is not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.2.5.1.3 General Design Criterion 11, 1967 - Control Room

The UHS is available to support safe shutdown from the control room or an alternate location if control room access is lost due to fire or other causes.

9.2.5.1.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation is provided as required to monitor UHS system variables.

9.2.5.1.5 Safety Guide 27, March 1972 - Ultimate Heat Sink for Nuclear Power Plants

The UHS will provide in excess of 30-day supply of cooling water to be available for shutdown and cooldown after normal and emergency conditions.

9.2.5.2 System Description

The Pacific Ocean is the UHS. The Pacific Ocean is the source of cooling water to the safety-related ASW system, along with other non-safety related cooling water systems discussed in Sections 9.2.3 and 10.4.5. The seawater from the Pacific Ocean passes through screening equipment located in the intake upstream of the pumps for which it supplies cooling water.

The ocean water supply to the ASW system provides the cooling and heat absorption capability required to remove waste heat under normal and emergency conditions from the NSSS. The waste heat from containment and other plant equipment is transferred to the CCW system. The heat picked up by the CCW system is transferred to the ASW system by the CCW heat exchangers. The ASW flows into the main condenser circulating water discharge structure and then into the ocean.

9.2.5.3 Safety Evaluation

9.2.5.3.1 General Design Criterion 2, 1967 - Performance Standards

The availability of the heat sink to provide cooling when required under severe conditions is discussed in detail in Section 2.4.12.6. The most severe oceanographic phenomenon to consider is a tsunami as discussed in Section 2.4.7. Estimates of wave runup on the plant facility are referenced in Section 2.4.6.6.

The expected downsurge during short periods of time would be to 9 feet below mean lower low water (MLLW). The arrangement of the intake channel and the design of the ASW pumps allow operation down to 17.4 feet below MLLW in the normal one-pump one-heat exchanger alignment. For reference, MLLW equals mean sea level (MSL) minus 2.6 feet. MSL is ground elevation zero.

The ASW portion of the intake structure and piping systems associated with the UHS are designed to the seismic conditions and requirements described in Section 2.5 and Sections 3.7 through 3.10, respectively. These components are constructed of materials compatible with the saltwater environment, or provided with protective features, to ensure the functionality of the components required for delivering the required cooling water supply to the ASW system and CCW heat exchangers.

9.2.5.3.2 General Design Criterion 4, 1967 - Sharing of Systems

The Pacific Ocean is the UHS. The Pacific Ocean is the source of cooling water to the safety-related ASW system. Because of the location of the plant on the ocean and the separation of intake and discharge structures, insignificant recirculation occurs.

9.2.5.3.3 General Design Criterion 11, 1967 - Control Room

Temperature of the UHS is measured at the circulating water pumps discharge and monitored in the control room.

9.2.5.3.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Temperature of the UHS is measured at the circulating water pumps discharge and monitored in the control room. Also, the temperature is measured outside the bar racks and recorded near the HSP.

9.2.5.3.5 Safety Guide 27, March 1972, - Ultimate Heat Sink for Nuclear Power Plants

Maximum temperature limits exist on the UHS to ensure the heat removal capability of the ASW/CCW system in normal and accident conditions. When the UHS exceeds 64°F, both CCW heat exchangers must be placed in service. Operation with elevated

UHS temperatures as high as 70°F is acceptable with two CCW heat exchangers in service. It has also been confirmed that the CCW heat exchangers will operate in a one pump two heat exchanger configuration. The limiting condition for operation and surveillance requirements of the UHS is discussed in the Technical Specifications (refer to Section 3.7.9 of Reference 1).

The ocean as a single water source for the UHS will provide in excess of 30 days of cooling water during normal and emergency shutdown conditions as required by Safety Guide 27, March 1972.

9.2.5.4 Tests and Inspections

Tests and inspections of piping systems between the reactor heat source and the UHS are discussed in their respective sections.

9.2.5.5 Instrumentation Applications

Temperature of the UHS is measured at the circulating water pumps discharge and monitored in the control room. Also, the temperature is measured outside the bar racks and recorded near the HSP.

9.2.6 CONDENSATE STORAGE FACILITIES

The ESF function of the CST in support of the AFW system is discussed in Section 6.5. The safety function of the CST is to provide condensate storage for RCS cooldown.

The condensate storage facilities, shown in Figure 3.2-16, provide for the storage and transfer of demineralized water from the MWS to the AFW system and to supply the normal makeup and rejection requirements of the steam plant.

The CST provides, as available, makeup water for CCW and SFP makeup (refer to Section 9.2.3).

The boundary of the condensate storage facilities consists of one CST per unit, a single makeup water transfer tank, the hydrazine mixing pumps, and interconnected valves and piping.

9.2.6.1 Design Bases

9.2.6.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the condensate storage facilities are designed to withstand the effects of, or are protected against, natural phenomena such as earthquakes, flooding, tornadoes, winds, tsunamis and other local site effects.

9.2.6.1.2 General Design Criterion 3, 1971 – Fire Protection

The condensate storage facilities are designed and located to minimize, consistent with other safety requirements, the probability of fires and explosions.

9.2.6.1.3 General Design Criterion 4, 1967 – Sharing of Systems

The condensate storage facilities are not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.2.6.1.4 General Design Criterion 11, 1967 – Control Room

The PG&E Design Class I portion of the condensate storage facilities are designed to or contain instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.2.6.1.5 General Design Criterion 12, 1967 – Instrumentation and Controls

Instrumentation is provided as required to monitor the condensate storage facilities variables within prescribed operating ranges.

9.2.6.1.6 Condensate Storage Facilities System Safety Function Requirements

(1) <u>Condensate Storage</u>

The CST is designed to provide condensate storage to support RCS decay heat removal and cooldown and as a backup for makeup water to CCW and the SFP.

(2) <u>Protection from Missiles</u>

The PG&E Design Class I portion of the condensate storage facilities are designed to be protected from missiles.

9.2.6.1.7 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The condensate storage facilities are designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.2.6.1.8 10 CFR 50.55a(f) – Inservice Testing Requirements

The condensate storage facilities ASME Code components are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

9.2.6.1.9 10 CFR 50.55a(g) – Inservice Inspection Requirements

The condensate storage facilities ASME Code components are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.2.6.1.10 10 CFR 50.63 – Loss of All Alternating Current Power

The CST provides a sufficient inventory of water to support RCS cooldown during a SBO event.

9.2.6.1.11 NUREG-0737 (Item II.E.1.1), November 1980 –Clarification of TMI Action Plan Requirements

Item II.E.1.1 – Auxiliary Feedwater Reliability System Evaluation: The CST is designed to provide redundant level indication and low level alarms in the control room to allow the operator 20 minutes to anticipate the need for makeup water or transfer to an alternate water supply to support AFW pump operation.

9.2.6.2 System Description

Design and operating parameters for the condensate storage facilities are given in Table 9.2-9. The condensate storage facilities consist of a CST for each unit and a common transfer tank located outside the east end of the auxiliary building. The CST has a floating roof to minimize oxygen absorption by the stored water. The capacity of each CST is 425,000 gallons, which includes a minimum reserve of approximately 225,000 usable gallons for AFW pump operation and a drainable but not usable volume of approximately 13,000 gallons. Refer to Section 6.5.3.7 for additional information on usable inventory (refer to Section 6.5 for a description of the AFW system). The capacity of the transfer tank is 150,000 gallons. The CST is used for condensate makeup and rejection. The DCPP Unit 1 CST serves as a source of water to the auxiliary boiler. Makeup water to other plant systems can be supplied from the CSTs by use of the makeup water transfer pump (refer to Section 9.2.3). The transfer tank provides a holding storage capacity while transferring water. The condensate storage facilities are shown in Figure 3.2-16.

The CST capacity is based on supplying the normal makeup and rejection requirements of the steam plant, and providing a source of feedwater for the AFW system. The seismic evaluation of these tanks is discussed in Sections 3.7.2.2.1.5, 3.8.2.4 and 3.9.2.2.3. All outdoor tanks are designed for atmospheric pressure and for 32 to 200°F design temperature.

The PG&E design classifications of the CST and the transfer tank are given in Table 3.2-3.

A gravity flow line is also provided to allow water to flow between the transfer tank and CST when required.

In an emergency, water can be supplied from other sources through a connection to the CST hydrazine recirculation line.

The makeup water transfer pumps can be used to pump water to the CSTs of DCPP Unit 1 and Unit 2 and the transfer tank (refer to Section 9.2.3).

The suction lines from the CSTs to the makeup water transfer pumps, which can supply makeup water to the CCW system are crosstied between DCPP Unit 1 and Unit 2 but are isolated by two manual valves. If necessary, the DCPP Unit 1 and Unit 2 CSTs can provide an additional source of makeup water to the DCPP Unit 1 and DCPP Unit 2 CCW systems. The DCPP Unit 1 and Unit 2 crosstie provides an operating flexibility that minimizes the possibility of system failure. The CST also provides a source of makeup water to the SFP (refer to Section 9.1.3.2).

9.2.6.3 Safety Evaluation

9.2.6.3.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portions of the condensate storage facilities are designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), and earthquakes (refer to Sections 3.7 and 3.8.2.4).

The tornado wind and associated missiles capability of the CST is given in Table 3.3-2. As discussed in Section 3.3.2.5.2.9, a tornado-induced tank failure causing an instantaneously large leak and flood is extremely unlikely. Loss of this tank does not significantly compromise safe shutdown capability, since the PG&E Design Class II raw water reservoir provides a backup supply of water for the AFW pumps. Leakage from the CSTs and transfer storage tank due to tornado or missile-induced damage will not result in the flooding of PG&E Design Class I equipment in the auxiliary building since essentially watertight cover plates are installed over the pipe entranceway from each tank into the auxiliary building. The water will drain away from the building via the plant yard drainage system.

The storage tanks are located outside the auxiliary building and, therefore, will not be subject to any unusual post-accident environment. The codes to which the tanks were built consider normal atmospheric conditions such as wind and rain in the design guides. Protective coats of paint are applied to the outside of each tank.

9.2.6.3.2 General Design Criterion 3, 1971 – Fire Protection

The condensate storage facilities are designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.2.6.3.3 General Design Criterion 4, 1967 – Sharing of Systems

The DCPP Unit 1 and Unit 2 CSTs are crosstied through a 4- inch line. The crosstie lines nozzles on each tank are located above the Technical Specification (Reference 1) volume requirement for AFW, thereby ensuring that failure of the crosstie line cannot reduce the condensate storage capacity for SG makeup from the AFW pumps. Additional sources of water for the AFW pumps are described in Section 6.5.2.

9.2.6.3.4 General Design Criterion 11, 1967 – Control Room

The PG&E Design Class I portion of the condensate storage facilities is designed to support actions to maintain and control the safe operational status of the plant from the control room.

The water level in each CST is displayed on a PG&E Design Class I local indicator and redundant PG&E Design Class I recorders in the control room. High, low, and low-low water levels are alarmed in the control room. The low level alarm on both DCPP Unit 1 and Unit 2 annunciates when the tank level is approaching the top of the internal plenums of the PG&E Design Class II nozzles. Refer to Section 6.5.3.7 for additional information on usable inventory. CST water level indication is provided locally at the tank, at the HSP, and in the control room.

9.2.6.3.5 General Design Criterion 12, 1967 – Instrumentation and Controls

Instrumentation is provided as required to monitor the condensate storage facilities variables within prescribed operating ranges as discussed in Section 9.2.6.3.4.

Refer to Section 6.5.3.20 for Regulatory Guide 1.97, Revision 3 CST level indication compliance.

9.2.6.3.6 Condensate Storage Facilities System Safety Function Requirements

(1) <u>Condensate Storage</u>

The CST provides storage of condensate to support RCS decay heat removal and cooldown as discussed in Section 6.5. The usable minimum reserve in the CST of approximately 225,000 usable gallons is discussed in Section 6.5.3.7. The usable volume CST reserve is ensured by internal plenums at the connections of all consumers of CST inventory in the usable volume region.

(2) Protection from Missiles

The condensate storage facilities design is such that physical protection is adequately provided against physical hazards in areas through which the system is routed. The CSTs and transfer tank are cylindrical steel tanks protected by reinforced concrete encasements (refer to Section 3.5). The location of the tanks also provides protection from physical hazards.

9.2.6.3.7 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The CST is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition (refer to Section 9.5.1).

9.2.6.3.8 10 CFR 50.55a(f) – Inservice Testing Requirements

The IST requirements for the applicable condensate storage facilities components are contained in the DCPP IST Program Plan.

9.2.6.3.9 10 CFR 50.55a(g) – Inservice Inspection Requirements

The ISI requirements for the applicable condensate storage facilities components are contained in the DCPP ISI Program Plan.

9.2.6.3.10 10 CFR 50.63 – Loss of All Alternating Current Power

During an SBO event, decay heat is removed from the core by natural circulation of the reactor coolant. This heat is then transferred to the secondary side of the SGs and discharged to the atmosphere through the 10 percent atmospheric dump valves. The CST provides storage of makeup water to the AFW pumps to support decay heat removal (refer to Section 6.5.3.18).

9.2.6.3.11 NUREG-0737 (Item II.E.1.1), November 1980 – Clarification of TMI Action Plan Requirements

Item II.E.1.1 – Auxiliary Feedwater Reliability System Evaluation: The AFW pumps take water from the CST, which is the preferred source of water for AFW. The CST provides redundant level indication and low level alarms in the control room. Additional sources of water for the AFW pumps are described in Section 6.5.2.1.1.

9.2.6.4 Tests and Inspections

The water in the CSTs will be sampled periodically to determine chemical quality. In addition, the activity level of the water will be checked, and if SG leakage is suspected, the frequency of the activity samples of water will be increased.

9.2.6.5 Instrumentation Applications

Instrumentation for the condensate storage facilities is discussed in Sections 9.2.6.3.4 and 9.2.6.3.5.

9.2.7 AUXILIARY SALTWATER SYSTEM

The ASW system, shown in Figure 3.2-17, is an open-cycle system that supplies cooling water to the CCW heat exchangers from the UHS, the Pacific Ocean, during normal operation, plant cooldown, and following a LOCA or MSLB. It transfers waste heat from the CCW system, via each CCW heat exchanger to the UHS.

9.2.7.1 Design Bases

9.2.7.1.1 General Design Criterion 2, 1967 - Performance Standards

The ASW system is designed to withstand the effects of or is protected against natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.2.7.1.2 General Design Criterion 3, 1971 - Fire Protection

The ASW system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.2.7.1.3 General Design Criterion 4, 1967 - Sharing of Systems

The ASW systems or components are not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.2.7.1.4 General Design Criterion 11, 1967 - Control Room

The ASW system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.2.7.1.5 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain ASW system variables within prescribed operating ranges.

9.2.7.1.6 Auxiliary Saltwater System Safety Function Requirements

(1) <u>Waste Heat Removal</u>

The ASW/CCW systems are designed to remove waste heat from the nuclear (primary) plant equipment and components during normal plant operation, plant cooldown, and design basis accidents.

(2) Single Failure

The ASW system and CCW system are essentially considered a single heat removal system for the purpose of assessing the ability to sustain either a single active or passive failure and still perform design basis heat removal.

(3) Redundancy

Vital ASW system components are redundant.

(4) Isolation

The ASW system includes provision for isolation of system components and may be split into separate trains during long term post-LOCA conditions.

(5) Protection from Missiles

Vital portions of the ASW system are designed, located, or protected against effects of missiles which may result from plant equipment failure and from events and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(6) Protection Against High Energy Pipe Rupture Effects

Vital portions of the ASW system are designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(7) Protection from Moderate Energy Pipe Rupture Effects – Outside Containment

The outside containment portion of the vital ASW system is designed to be protected against the effects of moderate energy pipe failure.

(8) Protection from Flooding Effects – Outside Containment

The outside containment portion of the vital ASW system is designed to be protected from the effects of internal flooding.

(9) Leak Detection

The CCW system serves as an intermediate system between normally or potentially radioactive systems and the ASW system, which is an open-cycle system that discharges to the UHS (Pacific Ocean).

9.2.7.1.7 10 CFR 50.55a(f) - Inservice Testing Requirements

ASW system ASME Code components are tested to the requirements of 10 CFR 50.55 a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical

9.2.7.1.8 10 CFR 50.55a(g) - Inservice Inspection Requirements

ASW system ASME Code components (including supports) are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.2.7.1.9 10 CFR 50.63 - Loss of All Alternating Current Power

The ASW system is required to provide cooling water to the CCW System following a SBO.

9.2.7.1.10 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The ASW system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.2.7.1.11 Generic Letter 89-10, June 1989 - Safety Related Motor-Operated Valve Testing and Surveillance

The ASW system safety-related and position-changeable MOVs meet the requirements of Generic Letter 89-10, June 1989 and associated Generic Letter 96-05, September 1996.

9.2.7.1.12 Generic Letter 89-13, July 1989 - Service Water System Problems Affecting Safety-Related Equipment

The CCW heat exchangers cooled by the ASW system are subject to monitoring and maintenance programs to ensure capability to perform their safety function as an alternative to a testing program. Maintenance practices, operating and emergency procedures, and training ensure effectiveness of these programs.

9.2.7.1.13 Generic Letter 91-13, September 1991 - Essential Service Water System Failures at Multi-Unit Sites

The DCPP procedures establish periodic flow testing, surveillance, and operability of the ASW cross-tie valve FCV-601.

9.2.7.2 System Description

There is a separate ASW system for Unit 1 and Unit 2. Each unit is provided with two ASW trains with crosstie capability. Each train consists of a full capacity electric motor-driven pump, the tubeside of the CCW heat exchanger and associated supply and discharge piping for the CCW heat exchanger. Upstream of the pumps, there is a unit ASW traveling water screen and a suction bay gate for each pump. There is a vacuum relief system on each ASW supply header piping to prevent water hammer. In addition, the Unit 1 and Unit 2 ASW piping system is arranged with interunit crosstie capability.

Each train is designed with the capability of providing adequate cooling to the CCW system during normal operation, plant safe shutdowns following normal operation, and refueling modes. Equipment design margins and system redundancy allow either an active or a passive failure of any component without degrading the system's cooling function under all modes of operation, including a design basis accident.

All system boundary components are located within the turbine building, the vacuum breaker vault, and the intake structure. These locations provide access for inspections and maintenance during either normal or post-accident operation.

The components are connected via buried, plastic-lined, carbon steel pipes between these structures. The buried piping is accessible for inspections and maintenance during train outages of sufficient duration (typically refueling).

9.2.7.2.1 Auxiliary Saltwater Pumps

The ASW pumps are powered from separate Class 1E 4.16-kV buses, which can be energized by either the normal source or the emergency diesel generators (EDGs). All train components satisfy PG&E Design Class I criteria. The pumps are single stage, vertical, wet pit type driven by 4.16-kV motors. The design data for the ASW pumps are tabulated in Table 9.2-1. The piping and other essential lines (power, sensing, and control) that pass from the pumps to the main portion of the plant are shown in Figure 9.2-3.

9.2.7.2.2 Electrical Conduits

The route of circuits F, G, and H carrying these Class 1E power and instrument signals to the ASW pumps parallels the piping between the turbine building and the intake structure, as shown in Figure 9.2-3. Embedded plastic (ABS) conduits are used. These conduits are encased in a sand envelope with a reinforced concrete slab cover throughout the entire run except at structure crossings and the portion from bluff penetration to the intake structure where they are encased in reinforced concrete envelopes. All connections to pull boxes or structures are flexible. The electrical aspects of these safety-related circuits are described in Section 8.3.

9.2.7.2.3 Intake Structure and Equipment

The ASW pumps are installed at the intake structure, as shown in Figure 9.2-2. This arrangement provides a separate bay and intake bay gate for each pump. The design classification of the intake structure is given in Table 3.2-3 and the seismic analysis is presented in Section 3.7.2. The PG&E Design Class I equipment located in the intake structure are the ASW pumps, ASW MOVs, ASW piping, including valves in the piping, ASW pump compartment heating, ventilation and air-conditioning (HVAC), and some ASW instrumentation.

Each unit's pair of ASW pump trains share a common traveling screen to remove floating debris from the incoming seawater. If the common screen for a unit becomes clogged with debris, seawater may be supplied to the ASW pump bays from the unit's circulating water pump bays via the demusseling valves. Level transmitters are provided on both the inlet and outlet of the ASW common traveling screen in each unit for the purpose of indication and annunciation of water level differential across the common screen and for automatic screen start. The level transmitters are shown in Figure 3.2-17.

Provisions exist to control marine fouling buildup in the ASW system pump forebays, piping and the CCW heat exchanger to minimize flow blockage, and slime buildup in tubes. Flow testing is routinely performed and heat exchanger differential pressure is monitored to ensure adequate flow for heat removal capabilities. Biofouling is controlled by continuous chlorination.

The ASW pumps, traveling screens, gates, and guides are cathodically protected to protect the equipment from corrosion.

9.2.7.2.4 Piping

The design classification for ASW piping is given in Table 3.2-3. The ASW piping is designed to perform its function and maintain its integrity considering the effects of the environment and load combinations due to varying pressures, temperatures, and seismic conditions. The arrangement of the ASW system buried supply piping between the intake structure and turbine building is shown in Figure 9.2-3. The supply lines exit the east wall of the intake structure and are supported by thrust blocks and surrounding soil in the filled area. The supply lines are typically anchored to the circulating water conduits and are buried in the same trenches except for the new bypass piping just outside of the intake structure that is buried in soil and supported by large reinforced concrete thrust blocks. Refer to Section 2.5.5.8 for discussion of potential liquefaction of soil beneath a portion of buried ASW piping. The supply lines are anchored in concrete to the circulating water conduits at 40-foot intervals. The pipe trench is backfilled with compacted granular fill between the anchors. Within the turbine building, the pipes are embedded in concrete.

A separate ASW line from each CCW heat exchanger discharges to the ocean at the discharge structure.

The pipe used in the ASW system is standard weight ASTM A53 and A106 seamless steel with a 1/8 inch thick layer of polyvinyl chloride thermally bonded to the pipe's interior surface and over the full face of the flanged ends. The exterior surface of the pipe is coated with epoxy for corrosion resistance.

Portions of the buried ASW supply and discharge pipes are cathodically protected by an impressed current system or embedded in concrete. In addition, the ASW supply pipes near the turbine building are also protected by a sacrificial anode system.

Due to the vulnerability of the buried portions of the ASW system supply piping to potential corrosion damage, corrosion protection for the piping is provided by an internal lining and external coating applied directly to the piping and cathodic protection systems installed at selected locations.

9.2.7.2.5 Discharge Structure

The discharge structure is a massive energy dissipating device located in the coastal bluff. The arrangement of the structure and the ASW pipe discharge is illustrated in Figure 11.2-9. The structure is divided into two chambers (one for each unit) that are open to the ocean under all conditions. The two ASW return lines for each unit discharge into the chamber of that unit. The base slab of the discharge structure is keyed into and poured on sound rock. Where possible, the walls were formed directly against sound rock.

9.2.7.2.6 Heat Exchangers

The design details of the CCW heat exchangers are given in Table 9.2-3. Performance of the CCW heat exchanger is based on performance curves provided by the manufacturer.

9.2.7.3 Safety Evaluation

9.2.7.3.1 General Design Criterion 2, 1967 - Performance Standards

The design classifications for PG&E Class I ASW SSCs are identified in Table 3.2-3. The PG&E Design Class I equipment located in the intake structure are the ASW pumps, ASW MOVs, ASW piping, including valves in the piping, ASW pump compartment HVAC, and some ASW instrumentation. In order to provide assurance that the function of PG&E Design Class I equipment will not be adversely affected even in the unlikely event of a seismic event, the intake structure (QA Class S) was reviewed to ensure that it would not collapse. The failure analysis was based on the HE discussed in Section 3.7.1. The capability of the intake structure to withstand winds, tornadoes, and associated missiles is discussed in Section 3.3, and to withstand design flood events is discussed in Sections 2.4 and 3.4.

The invert depth in the ASW channel is 31.5 feet below MSL. The ASW pump's intake bells are at 23 feet below MSL, and the pumps are designed to operate with a water level at 20 feet below MSL, which envelopes the postulated tsunami drawdown conditions (refer to Sections 2.4.6 and 2.4.12). The pump's mounting plates are located at the pump deck 2.1 feet below MSL with the motor drivers at 4 feet above MSL. Pumps and motors are situated in watertight compartments that are ventilated by forced air through a roof ventilation shaft with vent extensions (snorkels) having a high point of 49.4 feet above MSL and a low point of 45.57 feet above MSL. The ASW pump room vents are extended with steel snorkels to prevent seawater ingestion due to splash-up during the design flood event as described in Section 3.4. PG&E Design Class I equipment is thus ensured of operation during extreme tsunami drawdown and combined tsunami and storm wave runup conditions.

During tsunami drawdown, each ASW pump will deliver about 85 percent of design flow due to increased static head losses. This is a temporary condition and would not result in excessive temperatures in components to be cooled. If a second ASW pump is available, the temporary condition can be avoided by running two pumps in parallel. For operation with two CCW heat exchangers with no second pump available, operator action is required to isolate one of the heat exchangers during the tsunami drawdown as described in Sections 2.4.12.5 and 2.4.12.6.

Design flow can readily be maintained during high water conditions. MSL is ground elevation zero; MSL (in feet) equals MLLW (in feet) plus 2.6 feet.

The ASW system is the only safety-related system that has components within the projected sea wave zone. The ASW pumps are housed in watertight compartments in the intake structure. These compartments are ventilated by forced air through a roof ventilator as described in Section 2.4.6.7.

ASW pump vents are extended with steel snorkels that face eastward to prevent seawater ingestion due to splash-up during the design flood event and are described in Section 2.4.6.7. The pump and piping are designed to handle the increased pressure in the system due to the combination of normal system operating and temporary-high wave conditions. The expected downsurge during short periods of time would be to 9 feet below MLLW. The arrangement of the intake channel and the design of the ASW pumps allow operation down to 17.4 feet below MLLW in the normal one-pump one-heat exchanger alignment. For operation with two CCW heat exchangers in service, operator action would be required to isolate one of the heat exchangers during the tsunami drawdown as described in Sections 2.4.12.5 and 2.4.11.6. PG&E Design Class I equipment is therefore assured of operation during extreme tsunami drawdown and combined tsunami and storm wave runup conditions. For reference, MLLW equals MSL minus 2.6 feet. MSL is ground elevation zero.

The piping and Class 1E electrical circuits associated with the ASW system are buried except for short exposed portions at the intake structure vacuum breaker vault and at the turbine building, and therefore not subject to damage due to missiles from rotating equipment or tornadoes, or due to collapse of nonseismic structures. Seismic design for the PG&E Design Class I piping (including buried and embedded portions) is provided in the applicable ASW piping stress analysis and discussed in Section 3.7. Since no surface fault movement is postulated for the site (as discussed in Section 2.5) and since consideration has been given to the ductility of the material (as discussed in Section 3.8.2.3.5) and the method of construction for the conduit runs, they can be assured to remain in service during and following seismic events.

Wave protection measures at ground level and below to protect the ASW system buried piping and electrical conduits from tsunami/storm conditions include concrete covers, revetments, roadway slabs, pavement and gabion mattresses.

The ASW pumps are housed in watertight compartments preventing flooding from occurring from sources external to the compartments.

Differential rock movement would be required to overload the discharge structure. Differential rock movement or faulting is not a design criterion (refer to Section 2.5.4). However, if a collapse of the structure were postulated, it is not likely that the ASW flow would be obstructed. There is insufficient rubble from such a postulated collapse to block the flow from both ASW pipes.

9.2.7.3.2 General Design Criterion 3, 1971 - Fire Protection

The ASW system is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.2.7.3.3 General Design Criterion 4, 1967 - Sharing of Systems

A normally closed MOV provides separation between the Unit 1 and Unit 2 ASW supply headers and is shown in Figure 3.2-17. Since the valve is normally closed, the crosstie does not expose either unit to an additional active failure. The unit crosstie provides operating flexibility in that it is possible to have the Unit 2 standby pump provide water to Unit 1 in the event the Unit 1 standby pump is inoperable and vice versa. The operating condition of Unit 2 will be considered before crosstying to prevent jeopardizing the safety of Unit 2. If Unit 2 is already in a shutdown condition during a postulated accident on Unit 1, then the Unit 2 standby saltwater pump can provide backup to Unit 1.

9.2.7.3.4 General Design Criterion 11, 1967 - Control Room

Appropriate ASW system instruments and controls are provided to permit system operation from the control room (refer to Section 9.2.7.5). The ASW pumps are designed to be remotely operated from the HSP in the event that the main control room is uninhabitable (refer to Section 7.4.2.1.2.2).

9.2.7.3.5 General Design Criterion 12, 1967 - Instrumentation and Control Systems

The ASW system is monitored by the instrumentation listed in Section 9.2.7.5.

9.2.7.3.6 Auxiliary Saltwater System Safety Function Requirements

(1) <u>Waste Heat Removal</u>

The ASW system is designed to provide sufficient heat removal to maintain the CCW system within its design basis temperature limits for normal operation, plant cooldown and design basis accident conditions.

The CCW system transfers heat from both vital and non-vital systems to the UHS via the ASW system during normal operations and reactor shutdown. During normal operation, both ASW pumps and one supply header are aligned with the operating CCW heat exchanger. Only one pump is required to run; the second pump, being on standby, provides backup against an active failure. By means of unit and redundant supply header crosstie MOVs, the standby pump for one plant unit may act as a second standby for the other unit.

The ASW system capability to perform its design basis function assumes the ASW pumps are capable of providing the minimum required flow under conditions of low tide, high CCW heat exchanger tube side differential pressure and supply temperatures up to 64°F. As discussed in Section 9.2.5, the Technical Specifications require a second CCW heat exchanger be placed in service when UHS temperature exceeds 64°F. The ASW flow rate and minimum acceptable flow are a function of the number of ASW pumps and CCW heat exchangers in service based on operating conditions and assumed single failure.

The ASW system is designed to provide sufficient heat removal to maintain the CCW system within its design basis temperature limits for normal CCW system conditions.

During plant cooldown the ASW and CCW systems operate together to remove heat from vital equipment as follows:

- Reactor decay heat (RHR)
- Equipment and cooling

During the cooldown phase of a routine plant shutdown, both ASW pumps and CCW heat exchangers are in operation. If one pump or supply header is inoperative during cooldown, cooling would be accomplished safely, but the cooldown time would be extended.

Following design basis accidents with a postulated single active or passive failure, the ASW and CCW systems operate together to remove heat from vital equipment including heat loads as follows:

- Reactor Decay Heat
- Containment Accident Heat Loads
- Vital Loads

The ASW/CCW system must be able to remove the minimum required heat in order to ensure that the containment design pressure and temperature is not exceeded. Additionally, the ASW system must be able to remove sufficient heat from the CCW system so as to not exceed the CCW system design basis temperature limits when the containment heat removal equipment is operating at maximum predicted heat removal rates. The adequacy of the heat sink provided by the ASW/CCW systems has been evaluated to ensure that the minimum heat removal function is satisfied following a LOCA or MSLB (References 5 and 6). The ability of the ASW/CCW system to support the maximum containment heat removal without exceeding the CCW system design basis temperature limits following LOCA or MSLB has also been demonstrated (Reference 3).

During the SI phase or upon loss of the offsite power supply, both ASW pumps receive a start signal. On a bus transfer with no SI signal or loss of the offsite power supply, the previously operating ASW pump will immediately be restarted and the standby pump will receive a start signal. This design ensures both pumps in operation following the event of accident or upset condition, excluding the condition of a Class 1E F or G bus failure.

In the injection and post-LOCA recirculation phases of the accident, no operator action is required for operation or reconfiguration of the ASW system and its components. A decision to split the ASW system into separate trains to mitigate a passive failure would be made by the TSC if it became required. (Reference 8)

The capacity of the ASW system is based on post-design basis accident heat rejection requirements. The ASW and CCW systems operate together to remove heat from containment and safety-related loads following a design basis accident. Together the ASW and CCW systems must be able to remove the minimum required heat loads to ensure that the containment design pressure and temperature limits are not exceeded. The ASW system is designed to provide sufficient heat removal to maintain the CCW system within its design basis temperature limits for post-accident CCW system conditions.

The ASW system and CCW system are essentially considered a single heat removal system for the purpose of assessing the ability to sustain either a single active or passive failure and still perform design basis heat removal. The heat removal capability of the ASW/CCW system has been evaluated to ensure that the minimum containment heat removal function is satisfied following a LOCA or MSLB (References 5 and 6). A single train of ASW (one ASW pump and one CCW heat exchanger) provides sufficient heat removal from containment to mitigate an MSLB or LOCA. The ability of the ASW and CCW systems to support the maximum containment heat removal without exceeding the CCW maximum supply temperature design basis limit following a LOCA or MSLB has also been demonstrated (Reference 3). The mechanistic analyses credited one or two ASW pumps, depending on the assumed single failure. A single CCW heat exchanger was assumed to be in service throughout the transient (except when the UHS temperature exceeds 64°F, two CCW heat exchangers are assumed in service). No credit was taken for operator action to align the second CCW heat exchanger or an ASW pump from the opposite unit.

The design basis for ASW system performance to support analysis for peak containment pressure is described in Section 6.2. The design basis for ASW system performance to support the analysis for peak CCW temperature transients is described in Section 9.2.2. Critical ASW system assumptions include maximum ASW temperature, minimum ASW flow rate to the CCW heat exchanger, and single pump/heat exchanger operation (refer to References 3, 5, and 6).

(2) Single Failure

A malfunction analysis is presented in Table 9.2-2.

The ASW system and CCW system are essentially considered a single heat removal system for the purpose of assessing the ability to sustain either a single active or passive failure and still perform design basis heat removal. Refer to Section 3.1.1 for a description of DCPP single failure criteria and definition of terms.

During post-LOCA long term recirculation, the ASW trains should remain cross-tied to assure that any active failure in the ASW or CCW system would not result in the loss of CCW system cooling. While vulnerable to a passive failure in this configuration, the ASW system capacity is such that the ASW system function would not be affected. A decision to split the ASW system into separate trains to mitigate a passive failure would be made by the TSC if it became required. (Reference 8)

The ASW system is comprised of active components for which design classifications are given in Table 3.2-3. The ASW system can sustain either an active or a passive failure and still perform its function.

(3) Redundancy

Redundancy is provided by having two ASW pumps, one running and one on standby, and two CCW heat exchangers, with one normally in service and one in standby. The ASW system can be cross-connected within trains and between units so that various pump-heat exchanger combinations can be used for cooling. Redundant vacuum breakers are installed at the vertical bend of each line to eliminate water hammer.

Each unit's pair of ASW pump trains shares a common traveling screen to remove floating debris from the incoming seawater. If the common screen for a unit becomes clogged with debris, seawater may be supplied to the ASW pump bays from the unit's circulating water pump bays via the demusseling valves. Level transmitters are provided on both the inlet and outlet of the ASW common traveling water screen in each unit for the purpose of indication and annunciation of water level differential across the common screen and for automatic screen start. The level transmitters are shown in Figure 3.2-17.

(4) Isolation

The design classification of the CCW heat exchangers is listed in Table 3.2-3. Rupture of the heat exchanger tubes or channel is considered highly unlikely because of low operating pressures and the use of corrosion-resistant materials. However, a leaking heat exchanger can be identified by sequential isolation or visual inspection. If the leak should be in the operating heat exchanger, the standby heat exchanger will be placed in operation and the leaking heat exchanger isolated and repaired.

(5) Protection from Missiles

The provisions taken to protect the vital CCW system from missiles resulting from plant equipment failures and from events and conditions outside the plant are discussed in Sections 3.5.

(6) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect the vital portion of the ASW system from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Section 3.6.

(7) Protection from Moderate Energy Pipe Rupture Effects – Outside Containment

The provisions taken to provide protection of the vital portion of the ASW system located outside containment from the effects of moderate energy pipe failure are discussed in Section 3.6.

(8) Protection from Flooding Effects – Outside Containment

The provisions taken to provide protection of the vital portion of the CCW system from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

No systems that are required for safe shutdown are rendered inoperable due to flooding caused by a postulated break in the ASW piping. The low operating pressure and temperature of the saltwater system minimizes the possibility of a line severance. However, a severance would be detected and alarmed to the control room as low differential pressure across the heat exchanger and a high temperature rise across the CCW system, and possibly a pump motor failure. Sufficient valving is provided to isolate the units and their redundant trains from the failed section of piping.

Most of the ASW piping is buried except for short sections in the intake structure, the vacuum breaker vaults and the turbine building. A pipe break inside an ASW pump room or outside the boundary of both unit rooms in the Intake would not jeopardize the other pump motors. Each pump is housed in its own watertight compartment; therefore a pipe break would only flood one compartment. No components required to be operated for safe shutdown are located in the vacuum breaker vault. Failure of the ASW supply inside the turbine building would result in draining to the turbine building sumps (TBSs); a break in the ASW system discharge piping to the ocean would not result in flooding of the turbine building unless flow blockage in the line occurs, since the line pressure is negative.

In the event that the entire contents of the hotwell and heater drain tanks are discharged to the turbine building, the operability of PG&E Design Class I equipment (CCW heat exchangers) in the building is not endangered. The volume of water that would be discharged is within the capacity of the turbine building drain system. This system includes one 18-inch drain line from the TBS of each unit to the CWS discharge structure (refer to Figure 3.2-27 and Table 1.6-1). If this drain were clogged, the water flow would begin to fill the TBSs and equipment pits below 85 feet (refer to Figures 1.2-16 and 1.2-20). However, the capacity (58,000 cubic feet) below this elevation is more than the potential flooding volume. Refer to Section 10.4.7 for further discussion of flooding in the turbine building.

The ASW system is physically separated from all piping carrying high-energy fluid. The ASW system is a moderate-energy system as described in Section 3.6.

(9) Leak Detection

Provisions exist to isolate the CCW heat exchanger on both the ASW (tube-side) and CCW (shell-side). Large leakage could be detected by differential pressure across the tube side of the heat exchanger, by a decrease in CCW flow out of the heat exchanger, or by makeup to the CCW system.

The ASW discharges directly to the UHS. The ASW system provides cooling to the CCW heat exchanger. The CCW system is normally non-radioactive; however, the radiation level of the CCW system is monitored and an alarm is provided with a predetermined radiation level setpoint (refer to Section 11.4). In the event of an alarm for radiation on the CCW system, immediate efforts would be made to isolate the in-leakage. If the in-service CCW heat exchanger developed a leak at the same time, the heat exchanger would be isolated and the standby heat exchanger placed in service. Potential radioactive leakage into the CCW system is monitored to prevent/minimize release to the environment, via the ASW system, in the event of a concurrent leak in the heat exchanger.

Molybdate concentration in the CCW system is maintained by procedure for corrosion prevention. In the event of a leak, small or large, the chemical concentration in plant effluent would be greatly reduced by dilution with the ASW system and then by the main CWS.

9.2.7.3.7 10 CFR 50.55a(f) - Inservice Testing Requirements

The ASW system contains pumps and valves classified as PG&E Design Class I. The IST requirements for these components are contained in the IST Program Plan and comply with the ASME Code.

9.2.7.3.8 10 CFR 50.55a(g) - Inservice Inspection Requirements

The ISI requirements for CCW system components are contained in the ISI Program Plan and comply with the ASME Code.

9.2.7.3.9 10 CFR 50.63 – Loss of All Alternating Current Power

The ASW system provides cooling water in support of safe shutdown of the plant (Mode 3) following a SBO. Using a single ASW pump and heat exchanger, the system provides cooling water to the CCW system. Refer to Section 9.2.2.3.19 for description of components cooled by the CCW system.
Refer to Section 8.3.1.6 for further discussion of station blackout.

9.2.7.3.10 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The ASW system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition (refer to Section 9.5.1).

9.2.7.3.11 Generic Letter 89-10, June 1989 - Safety-Related Motor-Operated Valve Testing and Surveillance

The ASW system MOVs subject to the requirements of Generic Letter 89-10, June 1989 and associated Generic Letter 96-05, September 1996 meet the requirements of the DCPP MOV Program Plan.

9.2.7.3.12 Generic Letter 89-13, July 1989 - Service Water System Problems Affecting Safety-Related Equipment

Design fouling is considered in accident analyses. Fouling is a combination of tube microfouling and tube flow blockage resulting from marine life. Mechanical tube plugging is limited to two percent of the tubes before the performance of the heat exchanger, as defined by the curves, is impacted. As noted in Section 9.2.7.2.3, provisions exist to control marine fouling on the tube side (ASW) of the CCW heat exchanger. Cathodic protection is provided on the tube side of the heat exchanger in the waterboxes.

9.2.7.3.13 Generic Letter 91-13, September 1991 - Essential Service Water System Failures at Multi-Unit Sites

A normally closed MOV provides separation between the Unit 1 and Unit 2 ASW supply headers and is shown in Figure 3.2-17. Since the valve is normally closed, the crosstie does not expose either unit to an additional active failure. The unit crosstie provides operating flexibility in that it is possible to have the Unit 2 standby pump provide water to Unit 1 in the event the Unit 1 standby pump is inoperable and vice versa. The operating condition of Unit 2 will be considered before crosstying to prevent jeopardizing the safety of Unit 2. If Unit 2 is already in a shutdown condition during a postulated accident on Unit 1, then the Unit 2 standby saltwater pump can provide backup to Unit 1. Equipment Control Guideline 17.1, ”Auxiliary Saltwater Cross-Tie Valve FCV-601,” provides requirements for the valve operability.

The ASW system provides cooling water in support of safe shutdown of the plant (Mode 3) following a SBO. Using a single ASW pump and heat exchanger, the system provides cooling water to the CCW system. Refer to Section 9.2.2.3.19 for description of components cooled by the CCW system. Refer to Section 8.3.1.6 for further discussion of station blackout.

9.2.7.4 Tests and Inspections

The operating components are in either continuous or intermittent use during normal plant operation. Periodic testing of the standby feature of the ASW pump, testing of alarm setpoints, and visual inspections and preventive maintenance will be conducted in accordance with normal plant operating practices.

9.2.7.5 Instrumentation Applications

The ASW system is monitored by the following instrumentation:

(1) Differential pressure transmitters for both CCW heat exchangers

(2) ASW header pressure indicators

(3) Automatic start feature of the standby pump on loss of header pressure with pump status indicator in the control room

(4) Temperature indicators on the CCW heat exchanger, ASW (tube side) inlet and outlet

(5) Valve position indicators in the control room

(6) Differential level across the traveling water screens

(7) ASW pump room high water level alarm

(8) ASW pump room watertight door alarm

(9) High-temperature alarm for the ASW pump room

(10) ASW pump motor temperature indicators: upper/lower bearings and motor stator winding

(11) Inlet gate position indicator

(12) CCW heat exchanger, CCW (shell side) outlet temperature indicators

(13) ASW pump bay level indication

(14) ASW pump bay level alarm

9.2.8 DOMESTIC WATER SYSTEM

The DWS processes raw water from the raw water storage reservoirs to provide water suitable for human consumption.

9.2.8.1 Design Bases

The DWS is designed to provide drinking water at the plant. It is a PG&E Design Class III system.

9.2.8.2 System Description

The DWS receives its water from the domestic water treatment system. The domestic water treatment system takes water from the raw water storage reservoirs and processes it through a multi-media filter, a reverse osmosis module, a neutralizing-media filter, an activated carbon filter, and finally through a 10-micron cartridge filter. Prior to transferring the water to the domestic water storage and distribution piping system, it is disinfected using chlorine. The water is then supplied to the plant for drinking.

The radioactively uncontaminated utilities that receive domestic water include:

(1) Lavatories

(2) Water heaters

(3) Showers

(4) Maintenance connections

(5) Emergency eye wash

(6) Hose bibs

(7) Kitchen sinks

(8) Chemical laboratory sink in the chlorination building

(9) Drinking fountains

(10) Chemical laboratory sinks in maintenance shop buildings

(11) Landscape irrigation

After use, the domestic water passes into the plant sewage system where it is treated in a sewage treatment plant before being discharged to the ocean.

The potentially radioactively contaminated utilities that use domestic water are:

(1) Hot showers

(2) Laundry facilities

(3) Laboratory sinks

(4) Washdown area in hot machine shop

(5) Utility water connections in the radiologically controlled area

This water, after being used, drains to the liquid radwaste system (LRS) (refer to Section 11.2) for treatment.

9.2.8.3 Safety Evaluation

Failure of the DWS will not affect nuclear safety. Back-flow preventers are provided to prevent contamination of the domestic water by back-siphonage from potentially radioactively contaminated areas.

9.2.8.4 Tests and Inspections

Water quality tests and system integrity inspections will be performed periodically in accordance with normal plant operating procedures.

9.2.8.5 Instrumentation Applications

Local flow, pressure, and temperature indicators are used to monitor the system condition.

9.2.9 REFERENCES

1. Technical Specifications, Diablo Canyon Power Plant Unit 1 and Unit 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended.

2. Deleted in Revision 8

3. Evaluation of Peak CCW Temperature Scenarios for Diablo Canyon Units 1 and 2, WCAP-14282, Revision 1.

4. Deleted in Revision 12.

5. Analysis for Containment Response Following Loss-of-Coolant Accidents for Diablo Canyon Units 1 and 2, December 1993, WCAP-13907.

6. Analysis for Containment Response Following Main Steam Line Break for Diablo Canyon Units 1 and 2, December 1993, WCAP-13908.

7. NRC Letter, dated May 13, 1999, “Issuance of Amendments for Diablo Canyon Nuclear Power Plant, Unit No. 1 (TAC No. M98829) and Unit No. 2 (TAC No. M98830) and Close-out of Generic Letter 96-06 (TAC Nos. M96804 and M96805) (License Amendments 134/132)

8. NRC Letter, dated January 13, 2000, “Issuance of Amendments for Diablo Canyon Nuclear Power Plant Unit Nos. 1 and 2 (TAC Nos. MA 1406 and MA 1407) (License Amendments 138/138)

9.2.10 REFERENCE DRAWINGS

Figures representing controlled engineering drawings are incorporated by reference and are identified in Table 1.6-1. The contents of the drawings are controlled by DCPP procedures.

9.3 PROCESS AUXILIARIES

The process auxiliaries consist of those auxiliary systems associated with the reactor process systems. These systems include:

(1) the compressed air systems, including the backup air/nitrogen supply system (refer to Section 9.3.1);

(2) the sampling system (refer to Section 9.3.2);

(3) equipment and floor drainage systems (refer to Section 9.3.3);

(4) the CVCS (refer to Section 9.3.4);

(5) failed fuel detection (performed using the sampling system) (refer to Section 9.3.5);

(6) nitrogen and hydrogen systems (refer to Section 9.3.6);

(7) miscellaneous process auxiliaries (i.e., the auxiliary steam system, and the oily water separator [OWS] and TBS system) (refer to Section 9.3.7).

9.3.1 COMPRESSED AIR SYSTEM

The compressed air system (refer to Figure 3.2-25) provides compressed air for process control systems and for station service throughout Unit 1 and Unit 2 under normal operating conditions. The compressed air system consists of instrument air, service air, and the backup air/nitrogen systems. The backup air/nitrogen supply system is described in Section 9.3.1.2.2. The compressed air system is shared by Unit 1 and Unit 2. Table 3.2-3 lists its design classification.

The PG&E Design Class I portions of the instrument air and service air systems include the piping penetrating containment between the CIVs. The backup air/nitrogen supply system supplies compressed gas to the PG&E Design Class I pneumatic-operated components that are required to perform an active PG&E Design Class I function after the loss of the compressed air system. The backup air/nitrogen supply system also supplies air/nitrogen to selected components, pursuant to PG&E's commitments to the NRC to provide a seismically qualified backup air/nitrogen supply;
hence these components are classified as PG&E Design Class I even though operability of the valves is not strictly a PG&E Design Class I function.

9.3.1.1 Design Bases

9.3.1.1.1 General Design Criterion 2, 1967 – Performance Standards

Portions of the compressed air system are designed to withstand the effects of, or are protected against, natural phenomena such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.3.1.1.2 General Design Criterion 3, 1971 – Fire Protection

The compressed air system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.3.1.1.3 General Design Criterion 4, 1967 – Sharing of Systems

The compressed air system is not shared by the DCPP units unless it is shown that safety is not impaired by the sharing.

9.3.1.1.4 General Design Criterion 11, 1967 – Control Room

The compressed air system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room.

9.3.1.1.5 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the compressed air system variables within prescribed operating ranges.

9.3.1.1.6 General Design Criterion 40, 1967 – Missile Protection

The containment isolation portion of the compressed air piping system that penetrates containment is designed to be protected against dynamic effects and missiles that might result from plant equipment failures.

9.3.1.1.7 General Design Criterion 49, 1967 - Containment Design Basis

The compressed air system piping penetrating containment is designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of ECCSs.

9.3.1.1.8 General Design Criterion 54, 1971 - Piping Systems Penetrating Containment

The portions of the instrument air and service air piping systems that penetrate containment are provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. These piping systems are designed with a capability to test periodically that operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

9.3.1.1.9 General Design Criterion 56, 1971 - Primary Containment Isolation

The instrument air system contains valves in piping that penetrates containment and is connected directly to the containment atmosphere. One automatic isolation valve is provided outside containment and one check valve inside containment to ensure containment integrity is maintained.

The service air system contains valves in piping that penetrates containment and is connected directly to the containment atmosphere. One normally sealed closed manual valve is provided outside containment and one check valve inside containment to ensure containment integrity is maintained.

9.3.1.1.10 Compressed Air System Safety Function Requirements

(1) <u>Protection from Missiles</u>

The PG&E Design Class I backup-air/nitrogen supply portion of the compressed air system is designed to be protected against the effects of missiles and dynamic effects which may result from equipment failures and postulated rupture of piping. In addition, PG&E Design Class I components are protected against the effects of missiles resulting from the postulated failure of compressed air cylinders and associated piping.

(2) <u>Protection Against High Energy Pipe Rupture Effects</u>

The PG&E Design Class I backup-air/nitrogen supply portion of the compressed air system are designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(3) <u>Protection from Moderate Energy Pipe Rupture Effects – Outside Containment</u>

The outside containment portion of the PG&E Design Class I backup-air/nitrogen system is designed to be protected against the effects of moderate energy pipe failure.

(4) Protection from Jet Impingement – Inside Containment

The inside containment portion of the PG&E Design Class I backup-air/nitrogen system is designed to be protected against the effects of jet impingement which may result from high energy pipe rupture.

(5) Protection from Flooding Effects – Outside Containment

The outside containment portion of the PG&E Design Class I backup-air/nitrogen system is to be protected from the effects of internal flooding.

(6) Backup Air/Nitrogen Capacity

The backup air/nitrogen system is designed to have sufficient capacity to allow pneumatically operated devices to perform their design functions.

9.3.1.1.11 10 CFR 50.49 – Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

Instrument air components that require EQ are qualified to the requirements of 10 CFR 50.49.

9.3.1.1.12 10 CFR 50.55a(f) – Inservice Testing Requirements

The PG&E Design Class I compressed air system ASME pump and valves are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

9.3.1.1.13 10 CFR 50.55a(g) – Inservice Inspection Requirements

The PG&E Design Class I compressed air system ASME BPVC Section XI components are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.3.1.1.14 10 CFR 50.63 – Loss of All Alternating Current Power

The backup air bottles supplying the 10 percent atmospheric dump valves contain sufficient pressure to support decay heat removal to maintain hot standby during a SBO event.

9.3.1.1.15 Regulatory Guide 1.97, Revision 3, May 1983 - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

The compressed air system provides instrumentation in the control room to monitor instrument air CIV status post-accident.

9.3.1.1.16 Generic Letter 88-14, August 1988 – Instrument Air Supply Problems Affecting Safety-Related Equipment

The PG&E Design Class I portions of the instrument air system address the air quality requirements of Generic Letter 88-14, August 1988.

9.3.1.2 System Description

9.3.1.2.1 Compressed Air System

Compressed air is provided to each unit via two subsystems, instrument air and service air.

The compressed air system arrangement for Unit 1 and Unit 2 is shown in Figure 3.2-25. Table 9.3-1 lists design data for the components. Portions of the compressed air system are utilized by both the instrument air system and the service air system.

During normal operation the service air system is isolated from the instrument air system. A cross-tie line between the service air and the instrument air system can be used to supply instrument air in the event of a failure of the instrument air compressors. To maintain instrument air purity, the backfeed air enters the instrument air system upstream of the instrument air after filters.

The compressed air system is required for startup and normal operation of the plant. Except for the portion of the backup air/nitrogen supply system described in Section 9.3.1.2.2, and the portion of the air distribution piping penetrating the containment, the compressed air system is PG&E Design Class II.

The plant can be taken to and maintained at hot shutdown without the use of air-operated valves. However, some air-operated valves are required for going from hot shutdown to cold shutdown.

Two water-cooled rotary screw compressors, rated 650 scfm each are located in the turbine area of Unit 1, and one air-cooled rotary compressor, rated at 650 scfm is located at the Unit 1 west buttress. These compressors supply instrument air to Unit 1 and Unit 2. Normally, one of the rotary screw compressors operates as a base loaded machine and the other rotary compressors are on automatic standby - start control. In addition, four reciprocating air compressors, rated 334 scfm each, located in the turbine area of Unit 1, are normally maintained in automatic standby-start mode to start in the event all three rotary compressors fail to maintain the required supply header pressure. Each rotary screw compressor has its own control panel and local loading/unloading switch sensing pressure at the system supply header. Start and stop operation of rotary screw compressors is manual from their local control panel. Once started, these compressors load and unload automatically. A master compressor loading controller automatically loads and unloads the reciprocating compressors at preselected system supply header pressures.

Normally, the reciprocating compressors are on automatic standby - start control.

Since the compressed air system is not required for proper operation of pneumatically operated devices which have PG&E Design Class I functions, the system is not automatically switched to EDG power in the event of a loss of power. However, if diesel generator loading conditions permit, the air compressors can be manually restarted on EDG power.

The compressors have nonlubricated cylinders to provide an oil-free air system. Except for the rotary screw compressor at the Unit 1 west buttress, each compressor has a water-cooled aftercooler separator. The rotary screw compressor at the Unit 1 west buttress is an air-cooled machine identical to the water-cooled rotary compressors inside the turbine building. An additional water-cooled after cooler and moisture separator with drain trap is provided downstream of the rotary screw compressors in the common line going to pre-filters to further reduce the water load of the air entering the air dryers. A bypass around the above referenced after cooler and moisture separator is provided to perform maintenance activities.

Prefilters upstream of the air dryers protect the desiccant beds from contamination by entrained water, pipe scale etc.; thereby extending desiccant life. The system has two air dryers, one of which is an "adsorbent heat-regenerative" type and the other a "heatless" type. Both air dryers are designed to produce -40°F pressure dewpoint with a dryer inlet air temperature of 100°F and a pressure of 100 psig. However, the system components served by instrument air do not require that the system pressure dewpoint be maintained at -40°F. The air system pressure dewpoint is maintained at least 18°F below the minimum temperature at any point in the instrument air system to preclude water blockage of instrument air lines and to prevent a buildup of rust that can break free and block instrument air lines. The system pressure dewpoint is monitored by a direct reading dewpoint indicator, with a "high moisture content" local alarm and "common trouble alarm" in the main control room.

Dry air leaving the air receiver is filtered through a 1 micron positive seal type after-filter before passing to the instrument air distribution system. The after-filter is provided with differential pressure indication, with a "high differential pressure" local alarm and a "common trouble alarm" in the control room.

The two air receivers provided act as "pulsation dampers" to eliminate the pressure pulses that reciprocating compressors generate. They also provide storage capacity to meet occasional high demands for compressed air.

Service Air System

The service air system, located outdoors east of Unit 2 transformer yard, has two rotary screw compressors: one rated at 650 scfm and the other at 1050 scfm. Power supply to these compressors is from the 12-kV underground distribution system. Normally, one of the rotary compressors operates as a base-loaded machine and the other rotary compressor is on automatic standby – start control. Rotary compressors automatically load and unload at pre-selected system supply header pressures. Each rotary compressor has its own control panel and local loading/unloading switch sensing pressure at the system supply header. Start and stop operation of the compressors is manual from its local control panels. Once started, these compressors load and unload automatically.

The system has three air dryers, two of which are “heatless” type and the other a “heat of compression” type. Like the instrument air system air dryers, these dryers are also designed to produce -40°F pressure dewpoint. The service air system pressure dewpoint is also maintained within the limits stated above for the instrument air system. Dry air leaving the air dryers is filtered through a micron positive seal type after-filter before passing to the service air distribution system. Additional temporary compressors are used, as appropriate, to supply supplemental service air.

9.3.1.2.2 Backup Air/Nitrogen Supply System

The backup air/nitrogen supply system supplies the motive force to operate certain pneumatic components in the event of a loss of the compressed gas system. In some cases the backup air/nitrogen supply system supplements the compressed gas system rather than serving as a backup air/nitrogen supply. The backup air/nitrogen supply system utilizes as its source instrument air supplied from the compressed air system, and high pressure nitrogen supplied from the nitrogen system, which is then stored for use in accumulators. In addition, high pressure bottled air, high pressure bottled nitrogen and low pressure nitrogen from the nitrogen system are utilized. Compressed gas from these sources is supplied to pneumatic components that normally use instrument air from the compressed air system or high-pressure nitrogen from the nitrogen supply system.

Pneumatically operated devices are identified in the piping schematics of the various systems in Section 3.2. Loss of the normal air supply from the compressed air system will result in a safe shutdown of the unit. Most pneumatically operated devices in the plant that have PG&E Design Class I functions are designed to maintain a safe position or to assume a safe position upon loss of air pressure. Movement to this safe position (or maintaining this safe position) is accomplished by means of spring-return actuators and compressed gas from the backup air/nitrogen supply system. All such pneumatically operated devices are designed to achieve this safe position in the required time under the most limiting conditions, including gradual loss of the normal air supply from the compressed air system. Tables 3.9-9 and 6.2-39 show how valves will fail on loss of air or electrical power and the desired condition for safe shutdown. The tabulations show that most air-operated valves fail in the safe shutdown position upon loss of power or air. The tabulations also show the air operated valves that are required to operate or be maintained in a certain position for safe shutdown after an assumed loss of the compressed air system and that they are supplied with compressed gas from the backup air/nitrogen supply system (refer to Section 9.3.1.2.2). Therefore, safe shutdown will not be compromised upon loss of power or the compressed air system.

The backup air/nitrogen supply system supplies compressed gas to the PG&E Design Class I pneumatic-operated components that are required to perform an active PG&E Design Class I function after the loss of the compressed air system. The backup air/nitrogen supply system also supplies air/nitrogen to selected components, pursuant to PG&E's commitments to the NRC to provide a seismically qualified backup air/nitrogen supply; hence these components are classified as PG&E Design Class I even though operability of the valves is not strictly a PG&E Design Class I function. There are a number of PG&E Design Class II components that have a backup supply of air/nitrogen in the event of loss of the compressed air system to prevent undesirable transients and/or to facilitate normal operation and shutdown. In addition, there are some components, or group of components, that when actuated require more air than can readily be supplied through the air/nitrogen supply connection. In these cases, local accumulators or receivers are provided that contain enough stored gas to allow these components to respond in the desired time. Design classifications are included in Table 3.2-3. This equipment is not part of a PG&E Design Class I pressure boundary, so it is not subject to ASME BPVC Section III or ANSI B31.7.

In general, accumulators and bottles are wall-mounted or column-mounted as close to the application as possible.

The main steam isolation valves (MSIVs) receive a close signal on indication of a main steam line rupture (low steam line pressure, above P-11 setpoint, or high steam line pressure rate, below P-11 setpoint, in any steam line), or high-high containment pressure. Locally mounted air reservoirs, protected against system failure by check valves, can hold open the MSIVs for a short duration of time to allow for recovery of the air system. However, if the air system cannot be recovered, the plant can still be safely shutdown with the MSIVs closed.

9.3.1.3 Safety Evaluation

9.3.1.3.1 General Design Criterion 2, 1967 – Performance Standards

Portions of the compressed air system are housed within the PG&E Design Class I auxiliary and containment buildings. These buildings, or applicable portions thereof, are designed to withstand the effects of winds and tornadoes (refer to Section 3.3), external floods and tsunamis (refer to Section 3.4), earthquakes (refer to Sections 3.7 and 3.8.3), and other natural phenomena.

The PG&E Design Class I compressed air components have been seismically qualified.

PG&E Design Class I portions of the compressed air system that are located outside are resistant to tornadoes. The capability of the system served by compressed air to support plant safe shutdown is maintained if compressed air service to the supported components is lost (refer to Section 3.3.2).

The compressed air system components required to perform their PG&E Design Class I functions are located at higher elevations such that they are not affected by external floods and tsunamis.

The PG&E Design Class I portion of the backup air/nitrogen supply system is installed in accordance with the quality assurance requirements for PG&E Instrument Class IA and PG&E Instrument Class ID equipment. The accumulators are fabricated in accordance with the quality assurance requirements for PG&E Design Class I.

The PG&E Design Class I tubing is installed and protected in accordance with the PG&E Design Class I air piping requirements. The equipment is seismically qualified for its location in the plant to the applicable seismic spectra as described in Section 3.7.

9.3.1.3.2 General Design Criterion 3, 1971 – Fire Protection

The compressed air system is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.3.1.3.3 General Design Criterion 4, 1967 – Sharing of Systems

Since the instrument air and service air systems are not required for proper operation of pneumatically operated devices, which have PG&E Design Class I functions, sharing one air system for both units does not affect plant safety. A major failure of the distribution system for one unit could result in a loss of air pressure for the second unit but would not affect the safety of either unit. Manual isolation valves between units and on all major distribution headers allow isolation of sections of the system without affecting the normal operation of the remainder of the system. Automatic containment isolation will prevent accident conditions from propagating through the air system. The PG&E Design Class I portions of the backup air/nitrogen system are not shared between Unit 1 and Unit 2.

9.3.1.3.4 General Design Criterion 11, 1967 – Control Room

The compressed air system is designed to support actions to maintain and control the safe operational status of the plant from the control room.

Alarms in the control room indicate:

(1) "Common trouble" instrument air

(2) "Common trouble" service air

(3) Status of the standby reciprocating air compressors

The "Common Trouble" Instrument Air alarms are comprised of various system and component abnormal operations, and are flashed at the local annunciator "PK-80" located near the compressors at the 85 feet elevation in the turbine building.

The "common trouble" service air alarms are comprised of:

(1) Service air high moisture

(2) Service air low pressure

Activation of the standby reciprocating air compressors to full operational status is displayed in the control room.

Instruments are provided to indicate operational status of the major components of the compressed air system.

Backup air for the 10 percent atmospheric dump valves is alarmed in the control room.

9.3.1.3.5 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the compressed air system variables within prescribed operating ranges.

Local indications for the service air system moisture and pressure are provided near Unit 2 containment access controls at elevation 140 feet of the turbine building.

9.3.1.3.6 General Design Criterion 40, 1967 – Missile Protection

The provisions taken to protect the containment isolation portion of the compressed air system from damage that might result from missiles and dynamic effects associated with equipment and high-energy pipe failures are discussed in Sections 3.5, 3.6, and 6.2.4.

9.3.1.3.7 General Design Criterion 49, 1967 - Containment Design Basis

The compressed air piping routed through containment penetrations is designed to withstand the pressures and temperatures that could result from a LOCA without exceeding the design leakage rates. Refer to Sections 3.8.2.1.1.3 and 6.2.4, and Table 6.2-39 for additional details.

9.3.1.3.8 General Design Criterion 54, 1971 - Piping Systems Penetrating Containment

The compressed air system CIVs are periodically tested for operability and leakage. Testing of components required for the CIS is discussed in Section 6.2.4. Test connections are provided in the penetration and in the piping to verify valve and penetration leakage within prescribed limits.

9.3.1.3.9 General Design Criterion 56, 1971 - Primary Containment Isolation

The compressed air penetrations that are part of the CIS include the instrument and service air penetrations, which comply with the requirements of GDC 56, 1971, as described in Section 6.2.4 and Table 6.2-39.

9.3.1.3.10 Compressed Air System Safety Function Requirements

(1) Protection from Missiles

The provisions taken to protect the PG&E Design Class I backup-air/nitrogen portion of the compressed air system are discussed in Section 3.5.

Equipment essential for a safe and maintained reactor shutdown is located near major components of the compressed air system. This equipment is separated from these components by a PG&E Design Class I structure. As discussed in Section 3.5.2, the air receiver tanks (refer to Table 3.9-11) are capable of withstanding the thrust developed by failure of the largest pipe connected to them. The stresses in the tank holddown structure relating from this thrust do not exceed yield strength. Thus there is no danger to safe shutdown from postulated missiles created by the compressed air system.

(2) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect the PG&E Design Class I backup-air/nitrogen portion of the compressed air system from damage that might result from dynamic effects associated with a postulated rupture of piping are discussed in Section 3.6.

(3) Protection from Moderate Energy Pipe Rupture Effects – Outside Containment

The provisions taken to provide protection of the PG&E Design Class I backup-air/nitrogen portion of the compressed air system located outside containment from the effects of moderate energy pipe failure are discussed in Section 3.6.

(4) Protection from Jet Impingement – Inside Containment

The provisions taken to provide protection of the PG&E Design Class I backup-air/nitrogen portion of the compressed air system located inside containment from the effects of jet impingement which may result from high energy pipe rupture are discussed in Section 3.6.

(5) Protection from Flooding Effects – Outside Containment

The provisions taken to provide protection of the PG&E Design Class I backup-air/nitrogen portion of the compressed air system from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

(6) Backup Air/Nitrogen Capacity

The backup air/nitrogen system is designed to have sufficient capacity to allow pneumatically operated devices to perform their safety functions.

The backup air/nitrogen supply system provides a backup supply of compressed gas to the air-operated valves that are required to take the plant to cold shutdown and for those pneumatic operated valves that require a backup supply of compressed gas for other functions.

To take the plant to cold shutdown from hot shutdown, compressed gas from the backup air/nitrogen supply system is provided to valves for charging/spray capability, steam dump capability, and RCS boration sample capability. In addition, the operator also will have available the pressurizer power-operated relief valves (PORVs) required for overpressure protection, the capacity of letdown by line isolation valves, and containment fire water isolation valves. Backup air is also supplied to CCW control valves on the outlet of the RHR heat exchanger and the ASW control valves on the inlet of the CCW heat exchanger to ensure that these valves may be operated or maintained in the required position for safe shutdown.

Backup air/nitrogen for shutdown is provided in four ways:

(1) RCS boration sample valves are equipped with air accumulators that are protected from back flow into the main system by check valves. These can be used because the number of cycles for each valve is small, the air required for each valve is small, and the valves do not consume air in the quiescent state. The containment fire water isolation valve, the RHR heat exchanger CCW outlet valves and CCW heat exchanger ASW inlet valves are also equipped with accumulators to ensure that the valves can be operated or maintained in the required position for safe shutdown in the event of a loss of offsite power that causes the loss of the compressed air system. The containment fire water isolation valve is equipped with a seismically qualified accumulator to help ensure the valve can remain open in the event of a fire after an earthquake so that manual fire fighting capabilities are available inside containment to the PG&E Design Class I equipment.

(2) Throttling valves (the SG 10 percent atmospheric dump valves) and charging pump valve (the discharge to regenerative heat exchanger) consume air in the quiescent state. These valves are supplied with

backup nitrogen supplies. The nitrogen system is capable of supplying motive gas as long as required since the requirements are small compared to the system capacity. The nitrogen system is not seismically qualified, so a second backup system is provided. It is composed of compressed air bottles for the 10 percent atmospheric dump valves which can supply varying amounts of air via solenoid valves controlled from the control room and a nitrogen accumulator for the charging pump valve. Should the compressed air system and the nitrogen backup systems both fail, the compressed air bottles will be enabled by the operator to allow the operator to position the 10 percent atmospheric dump valves at any position desired. The charging pump valve (the discharge to regenerative heat exchanger) needs only to be placed in the open or closed position, so its backup consists of a control-room-operated solenoid valve that can supply motive gas from the nitrogen accumulator bypassing the normal control system.

(3) Letdown line isolation valves are supplied with backup nitrogen from the nitrogen system.

(4) The pressurizer PORVs, charging pump valves (to loop 3 and loop 4 of the cold leg), and the pressurizer auxiliary spray valve are special cases of on-off valves. Due to the number of cycles and the size of the valves, normal air receivers would be huge. Therefore, a high pressure nitrogen accumulator for each valve is provided which is supplied by the nitrogen system at approximately 850 psig. These high-pressure accumulators are capable of providing sufficient motive gas to meet all the requirements on loss of both the compressed air and nitrogen systems. (It should be noted that the cycling requirements for the PORVs comes from overpressure protection for mitigation of a spurious SI event, not shutdown requirements [refer to Section 15.2.15.3]).

9.3.1.3.11 10 CFR 50.49 – Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

The instrument air components required to function in harsh environments under accident conditions are qualified to the applicable environmental conditions to ensure that they will continue to perform their safety functions. The affected equipment are instrument air control isolation valve FCV-584 components consisting of a solenoid valve and position indication require EQ that are listed on the EQ Master List.

9.3.1.3.12 10 CFR 50.55a(f) – Inservice Testing Requirements

The IST requirements for compressed air pumps and valves are contained in the IST Program Plan and comply with the ASME Code for Operation and Maintenance of Nuclear Power Plants.

9.3.1.3.13 10 CFR 50.55a(g) – Inservice Inspection Requirements

Compressed air system components are included in the ISI program in accordance with ASME BPVC, Section XI.

9.3.1.3.14 10 CFR 50.63 – Loss of All Alternating Current Power

The backup air bottles supplying the 10 percent atmospheric dump valves contain sufficient pressure to support decay heat removal to maintain hot standby during a SBO event.

9.3.1.3.15 Regulatory Guide 1.97, Revision 3, May 1983 - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

The compressed air system provides instrumentation in the control room to monitor the instrument air system CIV post-accident position indication.

9.3.1.3.16 Generic Letter 88-14, August 1988 – Instrument Air Supply Problems Affecting Safety-Related Equipment

In the response to Generic Letter 88-14, August 1988, PG&E committed to verify by test that actual instrument air quality supplied to PG&E Design Class I equipment was consistent with the manufacturer’s recommendations and that PG&E Design Class I equipment would function as intended on loss of instrument air.

In order to ensure that oil, water, or other impurities will not result in the failure of instrumentation or other equipment, the compressed air system is provided with oil-free compressor cylinders, prefilter, moisture separator, air dryers, and 1-micron after-filters.

9.3.1.4 Tests and Inspections

The compressed air system was tested and inspected prior to initial plant operation. Provisions are made for functional tests of the low air pressure alarm and containment isolation. Filters, air dryers, and air receivers are periodically inspected.

9.3.1.5 Instrumentation Applications

Instrumentation for the compressed air system is discussed in Sections 9.3.1.3.4 and 9.3.1.3.5.

9.3.2 SAMPLING SYSTEMS

The plant sampling systems provide a means for obtaining liquid and gas samples for laboratory analyses of chemical and radiochemical conditions of the designated reactor and secondary plant systems. The systems are designed to permit sampling during all modes of plant operation. The sampling systems provide the means for manual, grab type, sample collection, and where applicable, on-line monitoring of key chemistry parameters. Assurance of a representative sample will be by administrative procedures based on experience and good sampling techniques. These will include purging of sample lines prior to taking a sample and utilizing appropriate pre-cleaned sampling containers.

The DCPP Unit 1 and Unit 2 sampling systems are composed of the following subsystems:

1) NSSS sampling system (refer to Section 9.3.2.2.1)

2) Post-accident sampling system (PASS) (refer to Section 9.3.2.2.2)

3) Secondary sampling system (refer to Section 9.3.2.2.3)

4) Turbine steam analyzer system (refer to Section 9.3.2.2.4)

9.3.2.1 Design Bases

9.3.2.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the NSSS sampling system, containment atmosphere monitoring system, and secondary sampling system are designed to withstand the effects of, or are protected against, natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis, and other local site effects.

9.3.2.1.2 General Design Criterion 3, 1971 – Fire Protection

The PG&E Design Class I portion of the NSSS sampling system, containment atmosphere monitoring system, and secondary sampling system are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.3.2.1.3 General Design Criterion 9, 1967 – Reactor Coolant Pressure Boundary

The portion of the NSSS sampling system within the RCPB is designed and constructed to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

9.3.2.1.4 General Design Criterion 11, 1967 – Control Room

The NSSS sampling system, containment atmosphere monitoring system, and secondary sampling system are designed to or contain instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room.

9.3.2.1.5 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the NSSS sampling system, secondary sampling system, and PASS variables within prescribed operating ranges.

9.3.2.1.6 General Design Criterion 17, 1967 – Monitoring Radioactivity Releases

The sampling systems are designed to provide means for monitoring the containment atmosphere, the facility effluent discharge paths of the NSSS sampling system and containment atmosphere monitoring system, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions (refer to Section 12.2).

9.3.2.1.7 General Design Criterion 40, 1967 – Missile Protection

The containment isolation portion of the NSSS sampling, secondary sampling, and containment atmosphere monitoring systems that penetrate containment is designed to be protected against dynamic effects and missiles that might result from plant equipment failures.

9.3.2.1.8 General Design Criterion 49, 1967 – Containment Design Basis

The PG&E Design Class I portion of the NSSS sampling system, containment atmosphere monitoring system, and secondary sampling system are designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a LOCA, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of ECCSs.

9.3.2.1.9 General Design Criterion 54, 1971 – Piping Penetrating Containment

NSSS sampling system piping, containment atmosphere monitoring system piping, and secondary sampling system piping that penetrate containment are provided with leak detection, isolation, redundancy, reliability and performance capabilities that reflect the importance to safety of isolating the system. The piping is designed with the capability to periodically test the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

9.3.2.1.10 General Design Criterion 55, 1971 – Reactor Coolant Pressure Boundary Penetrating Containment

Each NSSS sampling system line piping that penetrates containment is provided with CIVs.

9.3.2.1.11 General Design Criterion 56, 1971 – Primary Containment Isolation

The containment atmosphere monitoring system contains valves in piping that penetrates containment and is connected directly to the containment atmosphere. One automatic isolation valve is provided inside containment and one automatic isolation valve outside containment to ensure containment integrity is maintained.

9.3.2.1.12 General Design Criterion 57, 1971 – Closed System Isolation Valves

The secondary sampling system is designed such that each line that penetrates primary reactor containment and is not part of the RCPB or connected directly to the containment atmosphere has at least one CIV which is automatic.

9.3.2.1.13 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The NSSS sampling system is designed for maintaining control over the plants radioactive gaseous effluents. Appropriate holdup capacity is provided for retention of gaseous effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment.

9.3.2.1.14 Sampling System Safety Function Requirement

(1) Post-accident Sampling

Contingency plans are required for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere.

(2) Protection from Jet Impingement – Inside Containment

The PG&E Design Class I portion of the NSSS, post-accident, and secondary sampling systems located inside containment is designed to be protected against the effects of jet impingement which may result from high energy pipe rupture.

9.3.2.1.15 10 CFR Part 20 – Standards for Protection Against Radiation

The NSSS sampling system supports the protection of personnel from radiation sources such that doses are maintained below the limits prescribed in 10 CFR Part 20.

9.3.2.1.16 10 CFR 50.49 – Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants

NSSS sampling system, containment atmosphere monitoring system, and secondary sampling system that require EQ are qualified to the requirements of 10 CFR 50.49.

9.3.2.1.17 10 CFR 50.55a(f) – Inservice Testing Requirements

The PG&E Design Class I sampling system valves comply with the ASME code for Operation and Maintenance of Nuclear Power Plants and are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

9.3.2.1.18 10 CFR 50.55a(g) – Inservice Inspection Requirements

The PG&E Design Class I sampling system components are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.3.2.1.19 Regulatory Guide 1.97, Revision 3, May 1983 – Instrumentation for Light-Water- Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

The NSSS sampling system provides instrumentation to monitor system variables during and following an accident.

9.3.2.1.20 NUREG-0737 (Item III.D.1.1), November 1980 – Clarification of TMI Action Plan Requirements

Item III.D.1.1 – Primary Coolant Outside Containment: A program to reduce leakage from the NSSS sampling system and PASS outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels.

9.3.2.1.21 Generic Letter 96-06, September 1996 – Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions

NSSS sampling system piping and secondary sampling system piping have been evaluated for the issue of thermal over-pressurization of isolated piping sections that could affect containment integrity during accident conditions, as described in Generic Letter 96-06, September 1996.

9.3.2.2 System Description

9.3.2.2.1 Nuclear Steam Supply System Sampling System

The NSSS sampling system is designed for manual operation on an intermittent basis, under conditions ranging from full power operation to cold shutdown. Pipe internal diameters are sized such that solids do not clog the lines.

Sampling system discharge flows are limited under normal and anticipated fault conditions (malfunctions or failure) to preclude any radioactivity release beyond the site boundary in excess of plant release limitations. Adequate safety features are provided to protect laboratory personnel and prevent the spread of contamination from the sampling room. The reactor coolant hot leg samples are routed through a sufficiently long length of tubing inside containment, and flow rates are controlled to permit decay of the short-lived N^{16} isotope to a level that permits normal access to the sampling room. Equipment required for sampling capability to confirm RCS boron concentration is seismically qualified to allow the plant to be taken to cold shutdown conditions following a design basis seismic event. Backup air or nitrogen is provided to the required pneumatic-operated valves as described in Section 9.3.1.2.2.

The sampling system(refer to Figure 3.2-11) provides the representative samples for laboratory analyses. The analyses show both chemical and radiochemical conditions and provide guidance in the operation of the RCS, RHR, CVCS. Typical information obtained includes reactor coolant boron and chloride concentrations, fission product radioactivity level, hydrogen, oxygen, and fission gas content, conductivity, pH, corrosion product concentration, chemical additive concentration, etc. The information is used in regulating boron concentration adjustments, evaluating fuel element integrity and CVCS mixed bed demineralizer performance, and regulating additions of corrosion-inhibiting chemicals to the systems.

Samples are drawn from the following locations:

(1) Inside Containment

   (a) The pressurizer steam space (RCS)

   (b) The pressurizer liquid space (RCS)

   (c) Hot legs of reactor coolant loops (2 points in the RCS)

   (d) Each accumulator (SI system)

(2) Outside Containment

   (a) The letdown line (2 points, upstream and downstream of the demineralizers) (CVCS)

(b) Each RHR heat exchanger outlet (RHR system)

(c) The volume control tank (VCT) gas and liquid space (CVCS)

Local sample connections are provided at various locations throughout the plant. These connections are not considered part of the sampling system. Samples originating from locations within the containment flow through lines to the sampling room in the auxiliary building. Each line is equipped with a manual isolation valve close to the sample source, a remote air-operated valve immediately downstream of the isolation valve, and containment boundary isolation valves located inside and outside the containment.

Manual valves are located inside the sampling room for component isolation, sample flow control, and routing. High-temperature sample lines also contain a sample heat exchanger.

The reactor coolant hot leg samples are routed through a sufficiently long length of tubing inside containment, and flow rates are controlled to permit decay of the short-lived N^{16} isotope to a level that permits normal access to the sampling room. This room has controlled ventilation and drainage to control radioactivity release.

All sample lines originating from locations outside the containment are provided with manual isolation valves. The RHR system sample lines and the VCT liquid sample lines have, in addition, a remote air-operated sampling valve. Manual valves are located in the sampling room for flow control and routing.

The sample sink, which is located in the sampling room, contains a drain line to the LRS. Local instrumentation is provided to permit manual control of sampling operations and to ensure that the samples are at suitable temperatures and pressures before diverting flow to the sample sink. All sample lines are provided with a sample valve located at the sample sink, except for the VCT gas sample. The sample sink has a hood that is connected to the building ventilation exhaust system.

9.3.2.2.1.1 Component Description

Component codes and classifications are given in Table 3.2-3 and component design parameters are listed in Table 9.3-2.

9.3.2.2.1.1.1 Sample Heat Exchangers

The sample heat exchangers are of the shell and coil tube type. Sample flow circulates through the tube side, while CCW circulates through the shell-side. The tube side connections have socket-welded joints for connections to the high-pressure sample lines.

9.3.2.2.1.1.2 Sample Pressure Vessels

The sample vessels are sized to provide sufficient gas volume to perform a radiochemical analysis on the VCT gas space constituents, or sufficient reactor coolant volume for dissolved hydrogen and fission gas analyses.

Integral isolation valves are furnished with the vessel. Quick disconnect couplings containing poppet-type check valves are connected to nipples extending from the valves on each end.

9.3.2.2.1.1.3 Sample Sink

The sample sink is located in a hooded enclosure connected to the auxiliary building ventilation exhaust system. The work area around the sink and the enclosure is large enough for sample collection and storage, as well as for the radiation monitoring equipment.

The enclosure is penetrated by the various sample lines from the reactor systems and by a demineralized water line, all of which discharge into the sink which drains to the LRS. The sink and work area material is stainless steel.

9.3.2.2.1.1.4 Delay Line

The high-pressure reactor coolant loop sample line has sufficient length to provide at least a 40-second sample transit time within the containment. Additional transit time from the reactor containment to the sampling hood is provided by the sampling line. This allows for decay of the short-lived isotope, N^{16}, to a level that permits normal access to the sampling room.

9.3.2.2.1.1.5 Piping

All liquid and gas sample lines are austenitic stainless steel tubing and are designed for high-pressure service. Lines are so located as to protect them from accidental damage during routine operation and maintenance.

9.3.2.2.1.1.6 Valves

Stop valves within the containment are remotely operated from the sampling room. They are used to isolate all sample points and to route sample flow. A remotely operated isolation valve is provided for samples originating from the RHR system so that the operator need not enter a possibly high radiation area following a LOCA. Two isolation valves are provided, one inside and one outside the containment, on all sample lines leaving the containment. The valves trip closed upon actuation of the containment isolation signal. All valves in the system are constructed of austenitic stainless steel or equivalent corrosion-resistant material.

9.3.2.2.1.1.7 Hot Sample Sub-System (Unit 1 only)

The RCS Hot Sample Sub-system provides the capability to collect samples filtered from a side stream of the hot-leg sample flow. Filters to collect solids from the flow are located both upstream and downstream of a heat exchanger. The sub-system includes temperature, flow, and pressure instruments as well as valves and tubing.

9.3.2.2.1.2 System Operation

9.3.2.2.1.2.1 Reactor Coolant Loop and Pressurizer Liquid Samples

Reactor coolant loop and pressurizer liquid samples are obtained by opening the remotely operated isolation valve of the selected sample point. The sample heat exchangers cool the liquid samples.

A valve downstream of each heat exchanger is manually throttled to obtain the correct liquid sample flow rate. The sample stream flows through the sampling system to either sample sink 1-1 and 2-1 or the VCT in the CVCS until sufficient volume is purged to ensure that a representative sample will be obtained. When purging is completed, flow is diverted, if necessary, into the sample sink, and a sample is collected in a suitable container.

To obtain a reactor coolant loop liquid or pressurizer liquid sample for dissolved gas content, flow is diverted by opening and closing the appropriate valves into the appropriate sample pressure vessel. A valve is adjusted to the required downstream pressure for obtaining the correct sample flow rate. After the sample pressure vessel is purged and the sample collected by closing the isolation valves, the sample vessel is removed by opening the quick disconnect couplings. Note that pressurized samples of reactor coolant for gas analysis are normally obtained from the CVCS demineralizer inlet sample lines as described below.

9.3.2.2.1.2.2 Pressurizer Steam Space Samples

Pressurizer steam space samples are obtained by opening the remotely operated isolation valves. The sample heat exchanger condenses and cools the steam and the pressure is reduced by a manual valve. The condensate flows through a sample pressure vessel and into either sample sink 1-1 and 2-1 or the VCT in the CVCS until sufficient purge flow has been passed. The sample pressure vessel is then isolated and removed.

9.3.2.2.1.2.3 Residual Heat Removal Samples

During plant shutdown operations, samples are withdrawn from the outlet line of the RHR heat exchangers at a maximum temperature of 350°F. The correct sample flow rate is obtained by adjusting the valve downstream of the hot leg sample heat exchanger.

The fluid temperature is reduced in the sample heat exchanger by manual regulation of the sample flow to the heat exchanger. After sufficient volume has been purged to either sample sink 1-1 and 2-1 or the VCT in the CVCS to ensure that a representative sample will be obtained, flow is diverted, if necessary, to the sample sink for collection. If the pressure and temperature is low in the RHR system, local samples off the pump discharge can be taken.

9.3.2.2.1.2.4 Letdown Line Samples

Samples are obtained from the letdown line at a point upstream of the mixed bed demineralizers and at a point upstream of the VCT. After throttling, the fluid is initially purged to either sample sink 1-1 and 2-1 or the VCT in the CVCS and then routed into the sample sink for collection.

9.3.2.2.1.2.5 Volume Control Tank Gas Samples

After the VCT gas sample line has been drained of condensate, the gas is sampled by collection with a pressurized sample vessel or laboratory apparatus.

9.3.2.2.1.2.6 Volume Control Tank Liquid Samples

VCT liquid samples are obtained by opening the remotely operated isolation valve and purging the line to the sample sink. Following sufficient purging, a sample may be taken at the sample sink.

9.3.2.2.1.2.7 Accumulator Samples

Accumulator samples are obtained by opening the remotely operated isolation valve of the selected accumulator and purging the line to sample sink 1-1 and 2-1. Following sufficient purging, a sample is drawn at the sample sink.

9.3.2.2.1.2.8 Reactor Coolant Solids Samples (Unit 1 only)

Samples of solids from the RCS are obtained by opening manual valves on the hot sample panel to direct a small flow through the sampling filters. The filter effluent is directed to the inlet of the CVCS demineralizers. After the sub-system is isolated, collected solids can be dried and examined in onsite or offsite laboratories.

9.3.2.2.1.2.9 Availability and Reliability

The sampling system is not required to function during an emergency but is utilized to classify an emergency condition and is PG&E Design Class II. In the event of a LOCA, the system is isolated at the containment boundary. RCS sampling capability is available following a design basis seismic event.

9.3.2.2.1.2.10 Leakage Provisions

Samples are collected under a hood provided with a vent to the building exhaust ventilation system. Liquid leakage in the sample sink is collected in the sink and drained to the LRS. If there is any leakage from the system inside the containment (e.g., valve stem leakage), it is collected in the containment sump.

9.3.2.2.1.2.11 Exposure Control

The sampling room is equipped with a ventilated sample hood to reduce the potential for airborne radioactivity exposure of operating personnel. Sufficient length and flow control is provided in the reactor coolant sample line to reduce personnel exposure from short-lived radionuclides. Shielding is provided, as necessary, to reduce personnel exposures. The operating procedures specify the precautions to be observed when purging and drawing samples. All sampling operations are conducted with strict adherence to plant health physics safety regulations.

9.3.2.2.1.2.12 Incident Control

The system is designed to be operated on an intermittent basis under administrative control. The system is normally closed with no flow, except for the pressurizer steam space sample, which may be left open to provide a continuous purge. The reactor coolant hot leg sample may be open for extended periods of time for operation of the hot sample panel sub-system (Unit 1 only).

Sample lines penetrating the containment are equipped with remotely operated isolation valves that close on receipt of a containment isolation signal. In addition, the isolation valve in the CVCS letdown line outside the containment will close on containment isolation signal, isolating the letdown line sample lines from the containment.

9.3.2.2.2 Post-Accident Sampling System

The PASS (refer to Figure 3.2-11) is composed of the reactor coolant and containment sump (RHR pumps discharge) liquid sampling system and the containment atmosphere monitoring system. The equipment location for this system is also shown in Figure 3.2-11.

License Amendment 149 (Unit 1) and 149 (Unit 2) removed PASS administrative controls from the Technical Specifications and removed PASS-specific requirements from the DCPP licensing bases. License Amendment 168 (Unit 1) and 169 (Unit 2) eliminated Technical Specification requirements associated with hydrogen recombiners and hydrogen monitors. However, contingency plans to obtain samples from the RCS, containment sump, and containment atmosphere are required.

Elimination of PASS requirements does not affect previous approval of elimination of the boron concentration monitoring system (BCMS). Although approval of the elimination of the BCMS was based, in part, on the availability of PASS, the use of contingency plans is an acceptable alternative to PASS to obtain post-accident RCS chemistry information.

The PASS, also referred to as the Post-LOCA Sampling System in other plant documentation, provides facilities for sampling and analysis of reactor coolant, containment sump (RHR pumps discharge), and containment atmosphere following an accident. This system may be used to obtain further information on accident conditions in the RCS and containment.

The system is designed and located such that plant personnel are able to obtain the necessary samples and analyses under accident conditions while limiting personnel radiation exposure.

The PASS system (excluding the portions involved with containment penetrations), the sample room enclosure, the electrical supply, and the HVAC system are all designated as PG&E Design Class II.

9.3.2.2.2.1 Personnel Protection

The shielding panels provided in the sampling system cabinets and the 2 foot thick concrete walls that enclose the post-accident sampling room and its access ways, provide the necessary shielding to allow occupancy of the sample room and operation of the sampling system following a LOCA.

The ventilation system for the sample room is designed to aid in the protection of plant personnel from radiological contamination. This system is discussed in Section 9.4.10.

An area radiation monitor with local annunciation is installed in the post-accident sampling room to warn personnel occupying the sampling room of high or increasing radiation. High radiation in the sampling room is also alarmed at the main control room annunciator.

An eyewash station is provided in the post-accident sampling room.

9.3.2.2.2.2 Post-Accident Sampling System Liquid Sampling System

The PASS liquid sampling system consists of the liquid sample panel and the sample coolers.

The following liquid samples are received at the liquid sample panel:

(1) Reactor coolant hot legs 1 and 4

(2) RHR pump discharge

All samples received at the liquid sample panel are taken from existing lines outside of containment.

The liquid samples flow from their sources through the sample cooler to the liquid sample panel.

The liquid sample panel provides the means to purge each sample through its sample line and the sample panels to ensure representative samples are obtained for analysis.

All sample lines can be flushed with makeup water following each sampling operation, thereby reducing the radiation level in the sampling system.

The sample coolers for the liquid sampling system are installed in a sample cooler rack. One sample cooler is provided for each liquid sample line.

The cooling water sides of the sample coolers are arranged into two cooling banks. Each bank consists of five sample coolers connected in series and is provided with the following:

(1) Cooling water over-temperature switch

(2) Cooling water under-pressure switch

(3) Cooling water low flow switch

(4) Cooling water relief valve sized to relieve a full bore flow from a broken sample coil

(5) Cooling water isolation valves for the inlet and discharge for each bank

Each sample line is provided with an inlet isolation valve for maintenance purposes.

9.3.2.2.2.3 Containment Atmosphere Monitoring System

The containment atmosphere monitoring system is designed to sample the containment atmosphere for isotopic analysis.

The containment atmosphere samples are collected in the containment and routed through a containment penetration, then through the sample panel, and returned to the containment atmosphere. A containment atmosphere dilution system provides the capability to obtain grab samples of diluted containment atmosphere. For particulate and iodine isotopic analysis, removable particulate and silver zeolite filters are provided in the sampling panel.

The sample flow to the containment atmosphere sample panel is established with a nitrogen gas flow through an eductor.

Special features are employed to prevent particulate and iodine plateout in the inlet line to the first air sample flask. These features include electrical heat tracing on the sample line, minimum radius bends in the sample lines, and the use of plug valves that avoid changes in the sample conditions.

The sampling panel design provides the means to purge or flush the sample system.

9.3.2.2.2.4 Control Provisions

Controls are provided on the control panel in the PASS room that allows the operator to route these selected samples to the PASS room. Controls are also provided on the control panel to position the electrically operated CIVs on the containment air monitoring system.

Under accident conditions, post-accident sampling waste is returned to the containment via a local valve and a remote valve with controls at the control panel in the post-accident sample room.

9.3.2.2.3 Secondary Sampling System

Samples from the secondary side of the SGs originate at several points in the turbine cycle. These samples are taken from the discharge of condensate pumps, condensate booster pumps, condensate demineralizers, condensate drain pump, feedwater heaters, main steam leads and SG blowdown lines. Samples may be obtained at various local panels or at the secondary process control room for analysis. The analysis may be used to determine action level, condenser leak detection, corrosion product transport, and/or annunciator alarm signals. The main condensers are equipped with tube sheet and condensate tray salt water leak detection systems. These systems identify sea water ingress to the condenser. The leak detection system consists of eight in-line tube sheet monitors and seven in-line condensate tray monitors. Each sample point is monitored separately to identify condenser in-leakage.

9.3.2.2.4 Turbine Steam Analyzer System

A turbine steam analyzer system, which was originally installed to provide continuous direct sampling and analysis of both the low and high pressure turbine steam on Unit 1 and only high pressure steam on Unit 2, is no longer being used at DCPP. Other chemistry methods for evaluating steam chemistry are used.

9.3.2.3 Safety Evaluation

9.3.2.3.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the NSSS sampling system, containment atmosphere monitoring system, and secondary sampling system are located in the containment and the auxiliary buildings that are PG&E Design Class I (refer to Section 3.8). These buildings or applicable portions thereof are designed to withstand the effects of, or are protected against, natural phenomena, such as earthquakes (refer to Section 3.7), flooding (refer to Section 3.4) and tsunamis (refer to Section 2.4), tornadoes and winds (refer to Section 3.3), and other local site effects.

Equipment required for sampling capability to confirm RCS boron concentration is PG&E Design Class II and seismically qualified to allow the plant to be taken to cold shutdown conditions following a design basis seismic event.

9.3.2.3.2 General Design Criterion 3, 1971 – Fire Protection

The sampling systems are designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.3.2.3.3 General Design Criterion 9, 1967 – Reactor Coolant Pressure Boundary

The portions of the NSSS sampling system within the RCPB are designed and constructed to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime. Piping upstream of and including the RCS hot leg sampling CIV outside containment is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation including all anticipated transients and to maintain the stress levels within applicable stress limits (refer to Section 5.2.3 for a discussion of the RCPB materials of construction).

9.3.2.3.4 General Design Criterion 11, 1967 – Control Room

The plant is provided with a centralized control room common to both units that contain the controls and instrumentation necessary for operation of both units under normal and accident conditions.

The RCS hot legs sampling, pressurizer liquid and steam space sampling, and accumulator sampling CIVs can be operated from the control room.

The sampling system may be operated from the post-accident sampling room. An area radiation monitor with local annunciation is installed in the post-accident sampling room to warn personnel occupying the sampling room of high or increasing radiation. High radiation in the sampling room is also alarmed at the main control room annunciator (refer to Section 9.3.2.2.2.1).

9.3.2.3.5 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Process control instrumentation within the NSSS sampling system, secondary sampling system, and PASS are provided to acquire data concerning key sampling system parameters (refer to Section 9.3.2.5).

9.3.2.3.6 General Design Criterion 17, 1967 – Monitoring Radioactivity Releases

The containment atmosphere monitoring, the plant vents, and the liquid and gaseous waste systems effluent discharge paths are monitored for radioactivity concentrations during all modes of operations. Sampling of effluents is described in Section 11.4.2.2. The offsite radiological monitoring program is described in Section 11.6. GDC 17, 1967 is discussed in Section 12.2 (refer to Section 9.3.2.2.3).

9.3.2.3.7 General Design Criterion 40, 1967 – Missile Protection

The provisions taken to protect the containment isolation portion of the NSSS sampling, secondary sampling, and containment atmosphere monitoring systems from damage that might result from missiles and dynamic effects associated with equipment and high-energy pipe failures are discussed in Sections 3.5, 3.6, and 6.2.4.

9.3.2.3.8 General Design Criterion 49, 1967 – Containment Design Basis

The NSSS sampling system, containment atmosphere monitoring system, and secondary sampling system penetrations are designed to withstand the pressures and temperatures that could result from a LOCA without exceeding the design leakage rates (refer to Section 3.8.2.1.1.3).

9.3.2.3.9 General Design Criterion 54, 1971 – Piping Penetrating Containment

The NSSS sampling system, containment atmosphere monitoring system, and secondary sampling system CIVs are periodically tested for operability and leakage. Testing of components required for the CIS is discussed in Section 6.2.4. Test connections are provided in the penetration and in the piping to verify valve and penetration leakage within prescribed limits.

9.3.2.3.10 General Design Criterion 55, 1971 – Reactor Coolant Pressure Boundary Penetrating Containment

The NSSS sampling system is designed such that each line that is part of the RCPB that penetrates containment is provided with CIVs that comply with GDC 55, 1971 (refer to Section 6.2.4 and Table 6.2-39 for penetration configuration details).

9.3.2.3.11 General Design Criterion 56, 1971 – Primary Containment Isolation

The containment atmosphere monitoring portion of the sampling system is designed such that each line that connects directly to the containment atmosphere and penetrates primary reactor containment is provided with CIVs in compliance with GDC 56, 1971 (refer to Section 6.2.4 and Table 6.2-39 for penetration configuration details).

9.3.2.3.12 General Design Criterion 57, 1971 – Closed System Isolation Valves

The secondary sampling system is designed such that each line that is not part of the RCPB that penetrates containment or connected directly to the containment atmosphere is provided with CIVs that comply with GDC 57, 1971 (refer to Section 6.2.4 and Table 6.2-39 for penetration configuration details).

9.3.2.3.13 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The sample sink, part of the NSSS sample system, located in the sample room, which is located in the auxiliary building, has a hood that is connected to the building ventilation and exhausts through the plant vent. Liquid leakage in the sample sink is collected in the sink and drained to the LRS (refer to Sections 9.4.10, 9.3.2.2.1 and 11.2).

9.3.2.3.14 Sampling System Safety Function Requirement

(1) Post-Accident Sampling

Contingency plans have been developed to obtain and analyze highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. Plant procedures provide instructions for operating PASS during post-accident conditions for obtaining reactor coolant, containment sump, and containment air hydrogen concentration and containment particulate, iodine, and noble gas samples during post-accident recovery (refer to Section 9.3.2.2.2).

(2) Protection from Jet Impingement – Inside Containment

The provisions taken to provide protection of the PG&E Design Class I portion of the NSSS, post-accident, and secondary sampling system located inside containment from the effects of jet impingement which may result from high energy pipe rupture are discussed in Section 3.6.

9.3.2.3.15 10 CFR Part 20 – Standards for Protection Against Radiation

The reactor coolant hot leg sample line is designed to provide a sufficient transit time in containment to allow decay of the short lived isotope N^{16}. This reduces doses to allow normal access to the sample panel (refer to Section 9.3.2.2.1.1.4).

9.3.2.3.16 10 CFR 50.49 – Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants

The PG&E Design Class I NSSS sampling system, containment atmosphere monitoring system, and secondary sampling system SSCs required to function in harsh environments under accident conditions are qualified to the applicable environmental conditions to ensure that they will continue to perform their safety functions. The affected equipment is CIV solenoid valves and position switches that are listed on the EQ Master List.

9.3.2.3.17 10 CFR 50.55a(f) – Inservice Testing Requirements

The NSSS sampling system, containment atmosphere monitoring system, and secondary sampling system valves are contained within the IST program. The IST requirements for these components are contained in the IST Program Plan and comply with the ASME code for Operations and Maintenance of Nuclear Power Plants.

9.3.2.3.18 10 CFR 50.55a(g) – Inservice Inspection Requirements

The NSSS sampling system piping and secondary sampling system piping have a periodic ISI program in accordance with the ASME BPVC, Section XI.

9.3.2.3.19 Regulatory Guide 1.97, Revision 3, May 1983 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

CIV position indication meets the requirements of Regulatory Guide 1.97, Revision 3 (refer to Section 7.5 and Table 7.5-6).

The elimination of the BCMS and the use of the PASS to meet the guidance in Regulatory Guide 1.97, Revision 3, is acceptable.

9.3.2.3.20 NUREG-0737 (Item III.D.1.1), November 1980 – Clarification of TMI Action Plan Requirements

Item III.D.1.1 – Primary Coolant Outside Containment: Plant procedures have been developed to reduce leakage by monitoring surveillance requirements of radioactive systems outside containment.

9.3.2.3.21 Generic Letter 96-06, September 1996 – Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

Generic Letter 96-06, September 1996, identified the potential of thermally induced overpressurization of isolated water-filled piping sections in containment that could jeopardize the ability of accident mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage.

In the evaluation to determine potential effects of piping sections, DCPP identified potentially affected sampling system piping sections. The evaluations concluded that all potentially overpressurized penetrations were operable. The sections included isolation valves (air operated globe valve) whose design prevents overpressurization.

9.3.2.4 Tests and Inspections

The sampling system is in use daily. Periodic visual inspection and preventive maintenance are conducted using normal industry practice.

9.3.2.5 Instrumentation Applications

The instrumentation provided for the sampling system is discussed below. All of the instrumentation gives local indication.

9.3.2.5.1 Temperature

Instrumentation is provided to measure the temperature of the sample flow in the outlet line of each sample heat exchanger.

9.3.2.5.2 Pressure

Instrumentation is provided to measure the pressure in the sample lines downstream of each of the three sample vessels: (a) pressurizer steam sample vessel, (b) pressurizer liquid sample vessel, and (c) hot leg sample vessel.

9.3.2.5.3 Flow

Instrumentation is provided to measure the sample purge flow of all liquid samples to the VCT in the CVCS and also the VCT gas sample purge to the vent header in the gaseous waste system.

9.3.2.5.4 Chemical

Instrumentation is provided to measure dissolved hydrogen and oxygen in the letdown line sample. These in-line instruments provide an alternate to manual collection.

9.3.3 EQUIPMENT AND FLOOR DRAINAGE SYSTEMS

The equipment and floor drainage systems collect and channel waste liquids to be either reprocessed or discharged from the plant, except for those originating in the turbine building. The equipment drainage system originates at tanks, heat exchangers, pumps, flanges, filters, and valves in the containment (refer to Section 9.3.3.2.1.1) and auxiliary building, including the fuel handling area (refer to Section 9.3.3.2.1.2), that may contain radioactive liquids, and terminates at the inlet to the reactor coolant drain tank (RCDT) or the miscellaneous equipment drain tank (MEDT). The containment building floor drainage system terminates at the inlet to either the reactor cavity sump or one of the two containment structure sumps. The auxiliary building floor drains for both Unit 1 and Unit 2 terminate at the entry to the auxiliary building sump.

The equipment and floor drainage systems discharge to the LRS, which is described in detail in Section 11.2. The LRS accepts the liquid wastes from the equipment and floor drainage systems for processing or discharge from the plant. Figure 3.2-19 shows detailed piping schematics of these systems.

9.3.3.1 Design Bases

9.3.3.1.1 General Design Criterion 3, 1971 – Fire Protection

The equipment and floor drainage systems are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.3.3.1.2 General Design Criterion 4, 1967 – Sharing of Systems

Portions of the equipment and floor drainage systems are shared by the DCPP units only where it is shown that safety is not impaired by the sharing.

9.3.3.1.3 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The equipment and floor drainage systems are designed to maintain control of radioactive liquids.

9.3.3.2 System Description

The equipment and floor drainage systems are designed to provide adequate drainage during normal operation and inadvertent actuation of a single fire water sprinkler head. Provisions are made for (a) multiple drainage of certain areas; (b) prevention of backflooding; and (c) visual inspection of flow in drain lines wherever space permits; i.e., most lines above elevation 70 feet. The floor drainage systems also provide a detection method in the event of flooding.

However, the water accumulation that would result from any postulated failure of the drain system would not preclude safe shutdown of the plant.

9.3.3.2.1 Equipment Drain or Closed Drain System

The closed drain system is so called because drains from equipment are connected directly to the drainage system. The system provides drainage for equipment located both inside and outside containment. Liquid waste is not exposed to the atmosphere once it leaves the equipment until it reaches its destination, which is either the RCDT in containment or the MEDT in the auxiliary building.

9.3.3.2.1.1 Containment Building Equipment Drain or Closed Drain System

Containment building closed drain wastes are collected in the RCDT. Sources include the following:

(1) Reactor coolant loop drains

(2) Pressurizer relief tank

(3) RCPs No. 2 seal leakoff

(4) Excess letdown line

(5) Accumulators

(6) Refueling canal

(7) RCPs seal water inlet line drain

(8) Reactor flange leakoff

(9) Excess letdown heat exchanger

(10) Regenerative heat exchanger outlet line drain

9.3.3.2.1.2 Auxiliary Building Equipment Drain or Closed Drain System

Auxiliary building closed drain wastes are collected in the MEDT. Sources include the following:

(1) CVCS

- (a) Deborating demineralizer drain
- (b) Mixed bed demineralizer drain
- (c) Cation bed demineralizer drain
- (d) Evaporator feed ion exchanger drains
- (e) Reactor coolant letdown filter
- (f) Seal water heat exchanger (tube side) and filter
- (g) VCT drain and sample line drain
- (h) Charging pumps, header, bypass, and seal injection filter

(i) Letdown heat exchanger (tube side)

(j) Gas stripper feed pumps drain

(k) LHUTs recirculation pumps, line, and relief valve discharge

(2) SI system

(a) Containment recirculation water chamber

(b) SI pumps seal and drip pocket drain

(c) Various valve steam leakoffs

(3) RHR system

(a) RHR heat exchanger tube side drains

(b) RHR pump and line drain

(c) Various valve steam leakoffs

(4) NSSS sampling system

(a) NSSS sampling sink drain

(b) NSSS sample line drain

(c) VCT sample line drain

(5) Containment spray system

(a) Containment spray line drain

(b) Containment spray pumps

(6) SFP cooling system

(a) SFP resin trap filter

(b) SFP filter

(c) SFP pumps

(d) RWP filter

(e) SFP skimmer filter, pump, strainer, and drain

(f) SFP heat exchanger

(g) SFP demineralizer drain

(7) CCW system

(a) Waste gas compressor seal water coolers

(8) LRS

(a) Equipment drain receivers overflow and pumps

(b) Processed waste receivers overflow and pumps

(c) Spent resin motive water pumps

(d) Spent resin storage tanks

(9) Gaseous radwaste system

(a) Gas decay tanks drain

(b) Waste gas compressor moisture separator

(c) Gaseous radwaste vent header drain

(d) Waste gas compressor surge tank

(10) Turbine steam supply system

(a) SG blowdown tank

(11) Gland steam sealing system

Note: Normally the following drains go to the miscellaneous condensate return tank and TBS. They can be routed to the MEDT if elevated levels of radioactivity are present.

(a) Gland steam condenser drains

(b) Steam jet air ejector

9.3.3.2.2 Floor Drain or Open Drain System

The open drain system, also known as the floor drainage system, drains potentially contaminated areas in the containment and auxiliary building, and collects liquids from equipment that is located in those areas and that normally does not handle reactor coolant. The piping systems or trenches used in this system permit exposure of contents to the atmosphere.

The auxiliary building has been divided into a number of drainage zones. Each equipment compartment or area within a zone is drained by several grated floor drains (refer to Figure 9.3-5). The piping connected to the individual floor drains is 2 inches in diameter and feed 4-inch diameter headers, which eventually drain to a common collection header leading to the auxiliary building sump. Each header is provided with a check valve and loop seal as it enters the sump to prevent backup of water.

9.3.3.2.2.1 Containment Building Floor Drain or Open Drain System

Inside containment floor drain wastes are collected in the containment sumps and the reactor cavity sump. Sources include the following:

(1) RCP seal No. 3 leakoff

(2) Excess letdown heat exchanger shell relief and drain

(3) RCP thermal barrier relief

(4) RCP upper bearing cooling relief

(5) Containment fan cooler drip pans and coils

(6) RV support cooler relief

(7) RCP lube oil spill collection tanks

9.3.3.2.2.2 Auxiliary Building Floor Drain or Open Drain System

Potentially contaminated auxiliary building floor drain wastes are collected in the auxiliary building sump.

Sources for the auxiliary building sump include the following:

(1) CVCS

(a) Charging pump base drains

(b) Boric acid tanks, filters, and transfer pump drains

(c) Boric acid reserve tanks and transfer pumps' drain

(d) Chemical mixing tank

(e) Batching tank

(2) NSSS sampling system

(a) Sample sinks (1-2 and 2-2)

(3) Containment spray system

(a) Spray additive tank

(4) SFP cooling system

(a) SFP sump

(5) CCW system

(a) CCW surge tank relief

(b) RHR heat exchanger shell side drains

(6) LRS

(a) Floor drain receivers and pumps

(b) Chemical drain tank overflow

(c) Laundry and hot shower tanks overflow

(d) Spent resin loadout area

(e) Spent resin transfer filters

(f) Laundry/distillate tanks and pumps drain and tank overflow

(7) Ventilation system

(a) Plant vent drains

(8) Auxiliary steam system

(a) Package boiler blowdown tempering tank

(b) Auxiliary steam drain receiver and pumps

(9) Elevation 140-foot roof drains discharge to LRS only if contamination is detected

9.3.3.3 Safety Evaluation

9.3.3.3.1 General Design Criterion 3, 1971 – Fire Protection

The equipment and floor drainage systems are designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.3.3.3.2 General Design Criterion 4, 1967 – Sharing of Systems

Sharing of drain system components by the two units does not adversely affect plant safety. The use of shared components provides additional operating flexibility and provides increased capacity to accommodate occurrences which result in large drain system flows.

9.3.3.3.3 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The equipment and floor drainage systems design includes piping, sumps, and trenches that control radioactive liquids (refer to Section 9.3.3.2).

An analysis of the consequences of leakage from the LRS, including normal operation and postulated accidental releases, is presented in Chapters 11 and 15.

9.3.3.4 Tests and Inspections

The drainage systems were tested and inspected prior to plant operation and are periodically monitored during plant operation.

9.3.3.5 Instrumentation Applications

Flow and level instruments are provided for control, indication, and alarm in the drainage systems piping, collection tanks, and sumps. These instruments are described in Section 11.2.

9.3.4 CHEMICAL AND VOLUME CONTROL SYSTEM

The CVCS(refer to Figure 3.2-8) provides the following services to the RCS:

(1) Control of water chemistry conditions, activity level, soluble chemical neutron absorber concentration, and makeup water

(2) Maintenance of required water inventory in the RCS

(3) Filling, draining, and pressure testing

(4) Maintenance of seal water injection flow to the RCPs

(5) Processing of effluent reactor coolant to effect recovery and reuse of soluble chemical neutron absorber and makeup water

(6) High-head SI for the ECCS (refer to Section 6.3)

The CVCS is required for safe shutdown of the plant (refer to Sections 7.4, 9.3.4.1.5, and 9.3.4.1.26).

Centrifugal charging pumps (CCPs) CCP1 and CCP2 in the CVCS serve as the high-head SI pumps in the ECCS. Other than CCP1 and CCP2 and associated piping and valves, the CVCS is not required to function during a LOCA. During a LOCA, the CVCS is isolated except for CCP1 and CCP2 and the piping in the SI and seal injection paths. Operation of CVCS components performing ECCS functions to mitigate the effects of accidents is addressed in Section 6.3. CVCS components required to maintain the RCPB are discussed in Section 5.2. The remaining functions of the CVCS are discussed within this section.

9.3.4.1 Design Bases

9.3.4.1.1 General Design Criterion 2, 1967 – Performance Standards

The CVCS is designed to withstand the effects of, or be protected against, natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis, and other local site effects.

9.3.4.1.2 General Design Criterion 3, 1971 – Fire Protection

The CVCS is designed and located to minimize, consistent with other safety requirements, the probability and effects of fires and explosions.

9.3.4.1.3 General Design Criterion 4, 1967 – Sharing of Systems

The CVCS is not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.3.4.1.4 General Design Criterion 9, 1967 – Reactor Coolant Pressure Boundary

The CVCS design includes provisions for the control of RCS chemistry such that the materials of construction of the pressure-retaining boundary of the RCS are protected from corrosion that might otherwise reduce the system structural integrity during its service lifetime.

9.3.4.1.5 General Design Criterion 11, 1967 – Control Room

The CVCS is designed to support actions for safe shutdown and to maintain and control the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.3.4.1.6 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided, as required, to monitor and maintain the CVCS variables within prescribed operating ranges.

9.3.4.1.7 General Design Criterion 13, 1967 – Fission Process Monitors and Controls

The CVCS design includes means for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as concentration of soluble reactivity control poisons.

9.3.4.1.8 General Design Criterion 21, 1967 – Single Failure Definition

The CVCS is designed to perform its function after sustaining a single failure. Multiple failures resulting from a single event are treated as a single failure.

9.3.4.1.9 General Design Criterion 26, 1971 – Reactivity Control System Redundancy and Capability

The CVCS design includes one of the two independent reactivity control systems which are based on different design principles. The CVCS reactivity control system is capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. The CVCS is capable of holding the reactor core subcritical under cold conditions.

9.3.4.1.10 General Design Criterion 28, 1967 – Reactivity Hot Shutdown Capability

The CVCS design includes one of the two independent reactivity control systems which are capable of independently making and holding the core subcritical from any hot standby or hot operating condition.

9.3.4.1.11 General Design Criterion 30, 1967 – Reactivity Holddown Capability

The CVCS design includes a reactivity control system capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

9.3.4.1.12 General Design Criterion 31, 1967 – Reactivity Control Systems Malfunction

The CVCS design includes reactivity control systems capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

9.3.4.1.13 General Design Criterion 32, 1967 – Maximum Reactivity Worth of Control Rods

Limits, which include considerable margin, are placed on the maximum rates at which reactivity can be increased by the CVCS to ensure that the potential effects of a sudden or large change of reactivity cannot: (a) rupture the RCPB, or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

9.3.4.1.14 General Design Criterion 40, 1967 – Missile Protection

The containment isolation portion of the CVCS is designed to be protected against dynamic effects and missiles that might result from plant equipment failures.

9.3.4.1.15 General Design Criterion 49, 1967 – Containment Design Basis

The CVCS is designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of ECCSs.

9.3.4.1.16 General Design Criterion 54, 1971 – Piping Systems Penetrating Containment

The piping that is part of the CVCS that penetrates containment is provided with leak detection, isolation, redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating this system. The piping is designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

9.3.4.1.17 General Design Criterion 55, 1971 – Reactor Coolant Pressure Boundary Penetrating Containment

Each CVCS line that penetrates the containment is provided with CIVs.

9.3.4.1.18 General Design Criterion 56, 1971 – Primary Containment Isolation

The CVCS contains valving in piping that penetrates containment and that is connected directly to the containment atmosphere. Normally closed isolation valves are provided outside containment and automatic (check) valves are provided inside containment to ensure containment integrity is maintained.

9.3.4.1.19 General Design Criterion 68, 1967 – Fuel and Waste Storage Radiation Shielding

Radiation shielding is provided in the design of CVCS waste storage tanks, as required, to meet the requirements of 10 CFR Part 20.

9.3.4.1.20 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The CVCS design includes those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity is provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. The design for radioactivity control is justified on the basis of 10 CFR Part 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur.

9.3.4.1.21 Chemical and Volume Control System Safety Function Requirements

(1) Reactor Coolant System Inventory and Pressure Control

The CVCS is designed to maintain proper coolant inventory in the RCS for all normal modes of operation, including startup from cold shutdown, full power operation, and plant cooldown.

The CVCS is also designed to provide a flowpath to the pressurizer auxiliary spray system for means of cooling the pressurizer vapor volume during plant cooldown conditions.

(2) Protection from Missiles

The PG&E Design Class I portions of the CVCS are designed to be protected against the effects of missiles which may result from plant equipment failure and from events and conditions outside the plant.

(3) Protection Against High Energy Pipe Rupture Effects

The PG&E Design Class I portions of the CVCS are designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(4) Protection from Moderate Energy Pipe Rupture Effects – Outside Containment

The PG&E Design Class I portions of the CVCS located outside containment are designed to be protected against the effects of moderate energy pipe failure.

(5) Protection from Jet Impingement – Inside Containment

The PG&E Design Class I portions of the CVCS located inside containment are designed to be protected against the effects of jet impingement which may result from high energy pipe rupture.

(6) Protection from Flooding Effects – Outside Containment

The PG&E Design Class I portions of the CVCS located outside containment are designed to be protected from the effects of internal flooding.

(7) Reactor Coolant Pump Seal Water Injection

The CVCS is designed to provide filtered seal water flow to each RCP seal to provide cooling of the RCP seals and bearings.

9.3.4.1.22 10 CFR 50.49 – Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

CVCS components that require EQ are qualified to the requirements of 10 CFR 50.49.

9.3.4.1.23 10 CFR 50.55a(f) – Inservice Testing Requirements

CVCS ASME Code components are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

9.3.4.1.24 10 CFR 50.55a(g) – Inservice Inspection Requirements

CVCS ASME Code components are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.3.4.1.25 10 CFR 50.63 – Loss of All Alternating Current Power

The CVCS seal water system (supplied by CCP1, powered from the Class 1E bus associated with the credited AAC power source) is required to provide RCP seal injection in the event of a SBO.

9.3.4.1.26 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

Fire Protection of the CVCS is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.3.4.1.27 Regulatory Guide 1.97, Revision 3, May 1983 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

The CVCS provides instrumentation to monitor system variables during and following an accident.

9.3.4.1.28 NUREG-0737 (Items I.C.1, II.B.2, II.K.1.5, III.D.1.1), November 1980 – Clarification of TMI Action Plan Requirements

Item I.C.1 – Guidance for the Evaluation and Development of Procedures for Transients and Accidents: NUREG-0737, Supplement 1, January 1983 provides the requirements for I.C.1 as follows:

Section 7.1(b) – Transients and accidents were reanalyzed for the purposes of preparing technical guidelines and upgrading emergency operating procedures.

Item II.B.2 – Design Review of Plant Shielding and Environmental Qualification of Equipment for Space/Systems Which May Be Used in Post-accident Operations: Plant shielding provides adequate access to, and occupancy of, the switchgear rooms for the purpose of manually operating CVCS in its normal charging and letdown mode from the HSP, if required, following an MSLB.

Item II.K.1.5 – Safety Related Valve Position: Refer to Section 9.3.4.1.33.

Item III.D.1.1 – Integrity of Systems Outside Containment Likely to Contain Radioactive Material for PWRs and Boiling-Water Reactors: Appropriate portions of the CVCS are periodically pressure leak tested and visually inspected for leakage into the building environment.

9.3.4.1.29 Generic Letter 80-21, March 1980 – Vacuum Condition Resulting in Damage to Chemical Volume Control System Holdup Tanks

CVCS low pressure tanks, that can be valved to contain RCS water, are protected against vacuum conditions that could result in tank damage.

9.3.4.1.30 Generic Letter 88-17, October 1988 – Loss of Decay Heat Removal

To meet the requirements of Generic Letter 88-17, October 1988, and associated Generic Letter 87-12, July 1987, the CVCS PG&E Design Class I CCPs serve as the high pressure injection pumps to provide one of the two available or operable means of adding inventory to the RCS, that are in addition to the normal decay heat removal systems, to serve as backup sources to control RCS inventory upon loss of decay heat removal during non-power operation.

9.3.4.1.31 Generic Letter 89-10, June 1989 – Safety-Related Motor-Operated Valve Testing and Surveillance

CVCS PG&E Design Class I and position changeable MOVs meet the requirements of Generic Letter 89-10, June 1989, and associated Generic Letter 96-05, September 1996.

9.3.4.1.32 Generic Letter 95-07, August 1995 – Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves

The CVCS PG&E Design Class I, power-operated gate valves are designed such that they are not susceptible to pressure locking or thermal binding.

9.3.4.1.33 IE Bulletin 79-06A (Positions 8 and 12), April 1979 – Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident

Position (8) (subsequently NUREG-0737, November 1980, Item II.K.1.5, Safety Related Valve Position): CVCS PG&E Design Class I valve positions, positioning requirements, and positive controls have been assured such that the valves remain positioned (open or closed) in a manner to ensure the proper operation of ESF to satisfy Position (8) of IE Bulletin 79-06A, April 1979.

Position (12): CVCS design features are provided and procedures are in place during operating modes to deal with hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

9.3.4.1.34 NRC Bulletin 88-04, May 1988 – Potential Safety-Related Pump Loss

The CVCS is designed such that PG&E Design Class I pumps that share a common minimum flow recirculation line will not be susceptible to the pump-to-pump interaction described in NRC Bulletin 88-04, May 1988.

9.3.4.1.35 NRC Bulletin 88-08, June 1988 – Thermal Stresses in Piping Connected to Reactor Coolant Systems

Unisolable CVCS piping sections connected to the RCS, which have the potential to be subjected to unacceptable thermal stresses due to temperature stratifications induced by leaking valves, have been identified. Means have been provided to ensure that the pressure upstream from block valves, which might leak, is monitored and controlled.

9.3.4.2 System Description

9.3.4.2.1 Reactivity Control

The CVCS regulates the concentration of chemical neutron absorber in the reactor coolant to control reactivity changes resulting from the change in reactor coolant temperature between cold shutdown and hot full power operation, burnup of fuel and burnable poisons, and xenon transients.

9.3.4.2.1.1 Reactor Makeup Control

(1) The CVCS is capable of borating the RCS through either one of two flowpaths and from either one of two boric acid sources.

(2) The amount of boric acid stored in the CVCS always exceeds that amount required to borate the RCS to cold shutdown concentration assuming that the RCCA with the highest reactivity worth is stuck in its fully withdrawn position. This amount of boric acid also exceeds the amount required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

(3) The CVCS is capable of counteracting inadvertent positive reactivity insertion caused by the maximum boron dilution accident.

9.3.4.2.2 Regulation of Reactor Coolant Inventory

The CVCS maintains the proper coolant inventory in the RCS for all normal modes of operation including startup from cold shutdown, full power operation, and plant cooldown. This system also has sufficient makeup capacity to maintain the minimum required inventory in the event of minor RCS leaks.

The CVCS flow rate is based on the requirement that it permits the RCS to be either heated to or cooled from hot standby condition at the design rate and maintain proper coolant level.

9.3.4.2.3 Reactor Coolant Purification

The CVCS removes fission products and corrosion products from the reactor coolant during operation of the reactor and maintains these within acceptable levels. The CVCS can also remove excess lithium from the reactor coolant, keeping the lithium ion concentration within the desired limits for pH control.

The CVCS is capable of removing fission and activation products, in ionic form or as particulates, from the reactor coolant to provide access to those process lines carrying reactor coolant during operation and to minimize activity released due to leakage.

9.3.4.2.4 Chemical Additions

The CVCS provides a means for adding chemicals to the RCS to control the pH of the coolant during initial startup and subsequent operation, scavenge oxygen from the coolant during startup, and control the oxygen level of the reactor coolant due to radiolysis during all operations subsequent to startup. The CVCS is capable of maintaining the oxygen content and pH of the reactor coolant within limits specified in Table 5.2-15. There is also a capability to add zinc acetate to inhibit primary water stress corrosion cracking in Alloy 600 RCS components.

9.3.4.2.5 Seal Water Injection

The CVCS is able to continuously supply filtered water to each RCP seal, as required by the RCP design.

9.3.4.2.6 Hydrostatic Testing of the Reactor Coolant System

The CVCS can pressurize the RCS to its maximum specified hydrostatic test pressure to verify the integrity and leaktightness of the RCS through the use of a temporary hydrostatic test pump. The pump is capable of producing a hydrostatic test pressure greater than that required. The hydrostatic test is performed prior to initial operation and as part of the periodic RCS ISI program.

9.3.4.2.7 Emergency Core Cooling

The CVCS components providing ECCS functions are discussed in Section 6.3.

9.3.4.2.8 Subsystems Description

The CVCS is shown in Figure 3.2-8 with system design parameters listed in Table 9.3-5.

The CVCS consists of several subsystems: the charging, letdown, and seal water system; the chemical control, purification and makeup system; and the boron recycle system.

9.3.4.2.8.1 Charging, Letdown, and Seal Water System

The charging and letdown functions of the CVCS are employed to maintain a programmed water level in the RCS pressurizer, thus maintaining proper reactor coolant inventory during all phases of normal plant operation. This is achieved by means of a continuous feed and bleed process during which the feed rate is automatically controlled based on pressurizer water level. The bleed rate can be chosen to suit various plant operational requirements by selecting the proper combination of letdown orifices in the letdown flowpath.

Reactor coolant is discharged to the CVCS from the reactor coolant loop piping between the RCP and the SG; it then flows through the shell-side of the regenerative heat exchanger where its temperature is reduced by heat transfer to the charging flow passing through the tubes. The coolant then experiences a large pressure reduction as it passes through a letdown orifice and flows through the tube side of the letdown heat exchanger where its temperature is further reduced to the operating temperature of the mixed bed demineralizers. Downstream of the letdown heat exchanger a second pressure reduction occurs. This second pressure reduction is performed by the low-pressure letdown valve, the function of which is to maintain upstream pressure, which prevents flashing downstream of the letdown orifices.

The coolant then flows through one of the two mixed bed demineralizers. The flow may then pass through the cation bed demineralizer, which is used intermittently when additional purification of the reactor coolant is required.

The coolant then flows through the reactor coolant letdown filter and into the VCT through a spray nozzle in the top of the tank. The gas space in the VCT is filled with hydrogen. The partial pressure of hydrogen in the VCT determines the concentration of hydrogen dissolved in the reactor coolant.

The CCPs normally take suction from the VCT and return the cooled, purified reactor coolant to the RCS through the charging line. The VCT is located above the CCPs to provide sufficient net positive suction head. Normal charging flow is handled by one of the three CCPs. The bulk of the charging flow is pumped back to the RCS through the tube side of the regenerative heat exchanger. The letdown flow in the shell-side of the regenerative heat exchanger raises the charging flow to a temperature approaching the reactor coolant temperature. The flow is then injected into a cold leg of the RCS. Two charging paths are provided from a point downstream of the regenerative heat exchanger. A flowpath is also provided from the regenerative heat exchanger outlet to the pressurizer spray line. An air-operated valve in the spray line is employed to provide auxiliary spray from the CCPs to the vapor space of the pressurizer during plant cooldown. This provides a means of cooling the pressurizer near the end of plant cooldown, when the RCPs are not operating.

A portion of the charging flow is directed to the RCPs through a seal water injection filter. It enters the pumps at a point between the labyrinth seals and the No. 1 seal. Here the flow splits and a portion enters the RCS through the labyrinth seals and thermal barrier. The remainder of the flow is directed up the pump shaft, cooling the lower bearing, and leaves the pump via the No. 1 seal. Most of the No. 1 seal flow discharges to a common manifold, exits the containment, and then passes through the seal water return filter and the seal water heat exchanger to the suction side of the CCPs, or by alternate path to the VCT. A very small portion of the seal flow leaks through to the No. 2 seal. Seal No. 3 is a double-dam seal, providing a final barrier to leakage to containment atmosphere.

An excess letdown path from the RCS is provided in the event that the normal letdown path is inoperable. Reactor coolant can be discharged from a cold leg and flows through the tube side of the excess letdown heat exchanger. Downstream of the heat exchanger a remote-manual control valve controls the excess letdown flow. The flow normally joins the No. 1 seal discharge manifold and passes through the seal water return filter and heat exchanger to the VCT. The excess letdown flow can also be directed to the RCDT. When the normal letdown line is not available, the normal purification path is also not in operation. Therefore, this alternative condition would allow continued power operation for limited periods of time dependent on RCS chemistry and activity. The excess letdown flowpath may also be used to provide additional letdown capability when needed. This capability may be needed during RCS heatup, as a result of coolant expansion. This path removes some of the excess reactor coolant due to expansion of the system as a result of the RCS temperature increase. In this case, the excess letdown is diverted to the RCDT.

Surges in RCS inventory due to load changes are accommodated for the most part in the pressurizer. The VCT provides surge capacity for reactor coolant expansion not accommodated by the pressurizer. If water level in the VCT exceeds the normal operating range, a proportional controller modulates a three-way valve downstream of the reactor coolant letdown filter to divert a portion of the letdown to the LHUTs in the boron recovery system. If the high-level limit in the VCT is reached, an alarm is actuated in the control room and the letdown is completely diverted to the LHUTs.

Liquid effluent in the LHUTs is processed as a batch operation. This liquid is pumped by the gas stripper feed pumps through the evaporator feed ion exchangers. It then flows through the ion exchanger filter, and then the liquid is drained to the LRS for processing.

Low level in the VCT initiates makeup from the reactor makeup control system (RMCS). If the RMCS does not supply sufficient makeup to keep the VCT level from falling to a lower level, an emergency low-level signal causes the suction of the CCPs to be transferred to the RWST.

All parts of the charging and letdown system are shielded as necessary to limit dose rates during operation with 1 percent fuel defects assumed. The regenerative heat exchanger, excess letdown heat exchanger, letdown orifices, and seal bypass orifices are located within the reactor containment. All other system equipment is located inside the auxiliary building.

9.3.4.2.8.2 Chemical Control, Purification, and Makeup System

9.3.4.2.8.2.1 pH Control

The pH control chemical employed is lithium hydroxide. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium alloy/Inconel systems. In addition, lithium is produced in the core region due to irradiation of the dissolved boron in the coolant.

The lithium hydroxide is introduced into the RCS via the charging flow. The solution is prepared in the laboratory and poured into the chemical mixing tank. Primary makeup water is then used to flush the solution to the suction manifold of the CCPs. The concentration of lithium hydroxide in the RCS is maintained in the range specified for pH control (refer to Table 5.2-15). If the concentration exceeds this range, as it may during the early stages of core life, the cation bed demineralizer is employed in the letdown line in series operation with a mixed bed demineralizer. Since the amount of lithium to be removed is small and its buildup can be readily calculated, the flow through the cation bed demineralizer is not required to be full letdown flow.

9.3.4.2.8.2.2 Oxygen Control

During reactor startup from the cold condition, hydrazine is employed as an oxygen-scavenging agent. The hydrazine solution is introduced into the RCS in the same manner as described above for the pH control agent. Hydrazine is not employed at any time other than startup from the cold shutdown state.

Dissolved hydrogen is employed to control and scavenge oxygen produced due to radiolysis of water in the core region. Sufficient partial pressure of hydrogen is maintained in the VCT such that the specified equilibrium concentration of hydrogen is maintained in the reactor coolant. A pressure control valve maintains a minimum pressure in the vapor space of the VCT. This valve can be adjusted to provide the correct equilibrium hydrogen concentration.

9.3.4.2.8.2.3 Reactor Coolant Purification

Mixed bed demineralizers are provided in the letdown line to provide cleanup of the letdown flow. The demineralizers remove ionic corrosion products and certain fission products. One demineralizer is usually in continuous service for normal letdown flow and can be supplemented intermittently by the cation bed demineralizer, if necessary, for additional purification. The cation resin removes principally cesium and lithium isotopes from the purification flow.

The maximum temperature that will be allowed for the mixed bed and cation bed demineralizers is approximately 140°F. If the temperature of the letdown stream approaches this level, the flow will be diverted automatically so as to bypass the demineralizers. If the letdown is not diverted and temperature increases, there would be a decrease in overall ion removal capacity; however, the resins do not lose their exchange capability immediately.

There would be no safety problem associated with overheating of the demineralizer resins. The only effect on reactor operating conditions would be the possibility of a slight increase in the reactor coolant activity level and RCS chemical contaminants.

The deborating demineralizers are located downstream of the mixed bed and cation bed demineralizers and can be used intermittently to remove boron from the reactor coolant near the end of the core life when boron concentration is low. When the deborating demineralizers are in operation, the letdown stream passes through the mixed bed demineralizers and then through the deborating demineralizers and into the VCT after passing through the reactor coolant letdown filter.

A further cleanup feature is provided for use during cold shutdown and RHR. A remotely operated valve admits a bypass flow from the RHR system into the letdown line upstream of the letdown heat exchanger. The flow passes through the heat exchanger, through a mixed bed demineralizer and the reactor coolant letdown filter to the VCT. The fluid is then returned to the RCS via the normal charging route.

Filters are provided at various locations to ensure filtration of particulate and resin fines and to protect the seals on the RCPs.

Fission gases are removed from the system by venting the VCT to the waste disposal system.

9.3.4.2.8.2.4 Chemical Shim and Reactor Coolant Makeup

The soluble neutron absorber (boric acid) concentration and the reactor coolant inventory are controlled by the RMCS. In addition, for emergency boration and makeup, the capability exists to provide refueling water or 4 weight percent boric acid (7,000 ppm boron) to the suction of the CCPs. All CCPs are capable of providing emergency boration flow; however, only CCP1 and CCP2 are credited with this function during power operation, startup, and hot standby. However, CCP3 can serve as a backup to CCP1 and CCP2 in the event they are lost due to a fire (refer to Section 9.5.1).

The boric acid is stored in two boric acid tanks (also known as boric acid storage tanks). Two boric acid transfer pumps are provided with one pump normally aligned with one boric acid tank running continuously at low speed to provide recirculation of the boric acid system. The second pump is aligned with the second boric acid tank and is considered as a standby pump, with service being transferred as operation requires. This second pump also circulates fluid, as needed, through the second boric acid tank. Manual or automatic initiation of the RMCS will activate the running pump to the higher speed to provide normal makeup of boric acid solution as required.

The primary makeup water pumps, taking suction from the primary makeup water storage tank, are employed for various makeup and flushing operations throughout the systems. One of these pumps also starts on demand from the RMCS and provides flow to the boric acid blender. The flow from the boric acid blender is directed to either the suction manifold of the CCPs or the VCT through the letdown line and spray nozzle.

During reactor operation, changes are made in the reactor coolant boron concentration for the following conditions:

(1) Reactor startup - boron concentration must be decreased from shutdown concentration to achieve criticality.

(2) Load follow - boron concentration must be either increased or decreased to compensate for the xenon transient following a change in load.

(3) Fuel burnup - boron concentration must be decreased to compensate for fuel burnup.

(4) Cold shutdown - boron concentration must be increased to the cold shutdown concentration.

The RMCS instruments provide a manually preselected makeup composition to the CCP suction header or the VCT. The makeup control functions are those of maintaining desired operating fluid inventory in the VCT and adjusting reactor coolant boron concentration for reactivity control.

Automatic Makeup of Reactor Makeup Control System

The automatic makeup mode of operation of the RMCS provides boric acid solution preset to match the boron concentration in the RCS. The automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the coolant boron concentration.

Under normal plant operating conditions, the RMCS mode and makeup stop valves are set in the "automatic makeup" position. A preset low-level signal from the VCT level controller causes the automatic makeup control action to start a primary makeup water pump, switch a boric acid transfer pump to high-speed operation, open the makeup stop valve to the CCP suction, and throttle the concentrated boric acid control valve and the primary makeup water control valve. The flow controllers then blend the makeup stream according to the preset concentration. Makeup addition to the CCP suction header causes the water level in the VCT to rise. At a preset high-level point, the makeup is stopped, the primary makeup water pump stops, the primary makeup water control valve closes, the boric acid transfer pump returns to low-speed operation, the concentrated boric acid control valve closes, and the makeup stop valve to CCP suction closes.

If the automatic makeup fails or is not aligned for operation and the tank level continues to decrease, a low-level alarm is actuated. Manual action may correct the situation or, if the level continues to decrease, an emergency low-level signal from both channels opens the stop valves in the refueling water supply line and closes the stop valves in the VCT outlet line.

Dilute

The dilute mode of operation permits the addition of a preselected quantity of primary makeup water at a preselected flow rate to the RCS. The operator sets the RMCS mode to "dilute," the primary makeup water flow setpoint to the desired flow rate, the primary makeup water batch to the desired quantity, and initiates system start. This opens the primary makeup water control valve to the VCT and starts a primary water makeup pump that will deliver water to the VCT. From here the water goes to the CCP suction header. Excessive rise of the VCT water level is prevented by automatic actuation (by the tank level controller) of a three-way diversion valve which routes the reactor coolant letdown flow to the LHUTs in the boron recovery system. When the preset quantity of water has been added, the RMCS causes the pump to stop and the control valve to close.

Alternate Dilute

The alternate dilute mode of operation is similar to the dilute mode except a portion of the dilution water flows directly to the CCP suction and a portion flows into the VCT via the spray nozzle and then flows to the CCP suction.

Borate

The borate mode of operation permits the addition of a preselected quantity of concentrated boric acid solution at a preselected flow rate to the RCS. The operator sets the RMCS mode to "borate," the concentrated boric acid flow setpoint to the desired flow rate, the concentrated boric acid batch to the desired quantity, and initiates system start. This opens the makeup stop valve to the CCPs’ suction and switches the boric acid transfer pump to high-speed operation, which delivers a 4 weight percent boric acid solution (7,000 ppm boron) to the CCP suction header. The total quantity added in most cases is so small that it has only a minor effect on the VCT level. When the preset quantity of concentrated boric acid solution is added, the RMCS returns the boric acid transfer pump to low-speed operation and closes the makeup stop valve to the suction of the CCPs.

Manual

The manual mode of operation may be used to provide a blend of boric acid solution, to borate, or to dilute to the RCS, to the RWST, or to the LHUTs in the boron recovery system. While in the manual mode of operation, automatic makeup to the RCS is precluded. The discharge flowpath may be manually configured.

The operator then sets the RMCS mode to "manual," the concentration of boric acid is set and the required batch gallons are set to the desired flow rates, the boric acid and primary makeup water batch to the desired quantities, and initiates system start. The system start actuates the boric acid flow control valve and the primary makeup water flow control valve to the boric acid blender, starts the preselected primary makeup water pump, and switches the boric acid transfer pump to high-speed operation.

When the preset quantities of boric acid and primary makeup water have been added, the primary makeup water pump stops, the boric acid transfer pump returns to low-speed operation, and the boric acid control valve and the primary makeup water flow control valve close. This operation may be stopped manually by initiating system stop.

If either batch setpoint is satisfied before the other has recorded its required total, the pump and valve associated with the setpoint which has been satisfied will terminate flow. The flow controlled by the other setpoint will continue until that setpoint is satisfied.

Alarm Functions

The RMCS is provided with alarm functions to call the operator's attention to the following conditions:

(1) Deviation of primary makeup water flow rate from the control setpoint.

(2) Deviation of concentrated boric acid flow rate from control setpoint.

(3) High level in the VCT. This alarm indicates that the level in the tank is approaching high level and a resulting 100 percent diversion of the letdown stream to the LHUTs in the boron recovery system.

(4) Low level in the VCT. This alarm indicates that the level in the tank is approaching emergency low level and resulting realignment of CCP suction to the RWST.

9.3.4.2.8.3 Boron Recovery System

The boron recovery system collects borated water that results from the following plant operations. In each of these operations, the excess reactor coolant is diverted from the letdown line to the LHUTs as a result of high VCT level.

(1) Dilution of reactor coolant to compensate for core burnup

(2) Load follow

(3) Hot shutdowns and startups

(4) Cold shutdowns and startups

(5) Refueling shutdowns and startups

Excess liquid effluents containing boric acid and flow from the RCS through the letdown line are collected in the LHUTs. As liquid enters the LHUTs, the nitrogen cover gas is displaced to the gas decay tanks in the waste disposal system through the waste vent header. The concentration of boric acid in the LHUTs varies through core life from the refueling concentration to near zero at the end of the core cycle. An LHUT recirculation pump is provided to transfer liquid from one LHUT to another.

Liquid effluent in the LHUTs is processed as a batch operation. This liquid is pumped by the gas stripper feed pumps through the evaporator feed ion exchangers. It then flows through the ion exchanger filter, and then the liquid is drained to the LRS for processing.

9.3.4.2.9 Component Description

A summary of principal CVCS component design parameters is given in Table 9.3-6. CVCS safety classifications and design codes are given in Table 3.2-3. All CVCS piping that handles radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal for maintenance and hydrostatic testing.

9.3.4.2.9.1 Centrifugal Charging Pumps

Three CCPs inject coolant into the RCS and are of the single-speed, horizontal, centrifugal type. All parts in contact with the reactor coolant are fabricated of austenitic stainless steel or other material of adequate corrosion-resistance. There is a minimum flow recirculation line on each CCP discharge header to protect them against a closed discharge valve condition.

Charging flow rate is determined from a pressurizer level signal. Charging flow control is accomplished by a modulating valve on the discharge side of the CCPs. CCP1 and CCP2 also serve as the high-head SI pumps in the ECCS with induction motors that are powered by the Class 1E 4.16-kV system.

9.3.4.2.9.2 Boric Acid Transfer Pumps

Two horizontal, centrifugal, two-speed pumps with mechanical seals are supplied. The pumps' motors are powered by the Class 1E 480-V system. Normally, one pump is aligned with one boric acid tank and runs continuously at low speed to provide recirculation of the boric acid system. The second pump is aligned with the second boric acid tank, then considered as a standby pump, with service being transferred as operation requires. This second pump also intermittently circulates fluid through the second tank. Manual or automatic initiation of the RMCS will activate the running pump to the higher speed to provide normal makeup of boric acid solution, as required. For emergency boration, supplying of boric acid solution to the suction of the CCPs can be accomplished by manually actuating either or both pumps. The transfer pumps also function to transfer boric acid solution from the batching tank to the boric acid tanks. In addition to the automatic actuation by the RMCS, and manual actuation from the main control board, these pumps may also be controlled locally at the HSP.

The pumps are heat-traced to prevent crystallization of the boric acid solution. All parts in contact with the solution are of austenitic stainless steel.

9.3.4.2.9.3 Gas Stripper Feed Pumps

The two gas stripper feed pumps supply feed through the evaporator feed ion exchangers from the LHUTs and route it to the LRS. The non-operating pump is a standby and is available for operation in the event the operating pump malfunctions. These centrifugal pumps are constructed of austenitic stainless steel.

9.3.4.2.9.4 Liquid Holdup Tank Recirculation Pump

The recirculation pump is used to mix the contents of an LHUT for sampling or to transfer the contents of an LHUT to another LHUT. The wetted surface of this pump is constructed of austenitic stainless steel.

9.3.4.2.9.5 Boric Acid Reserve Tank Pumps

The two boric acid reserve tank pumps discharge water from the boric acid reserve tanks to other portions of the CVCS. The pumps are constructed of austenitic stainless steel.

9.3.4.2.9.6 Boric Acid Reserve Tank Recirculation Pumps

Two boric acid reserve tank recirculation pumps are provided for each tank. Only one pump per tank is running at a time with the other on standby. These pumps are used to recirculate boric acid through installed piping, equipment, and an inline heater to maintain a fluid temperature above 80°F. These pumps are seal-less and the wetted surface is constructed of austenitic stainless steel.

9.3.4.2.9.7 Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover heat from the letdown flow by reheating the charging flow, which reduces thermal shock on the charging penetrations into the reactor coolant loop piping.

The letdown stream flows through the shell of the regenerative heat exchanger, and the charging stream flows through the tubes. The unit is made of austenitic stainless steel and is of an all-welded construction.

9.3.4.2.9.8 Letdown Heat Exchanger

The letdown heat exchanger cools the letdown stream to the operating temperature of the mixed bed demineralizers. Reactor coolant flows through the tube side of the exchanger while CCW flows through the shell-side. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel.

9.3.4.2.9.9 Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow, which is equivalent to the nominal seal injection flow which flows downward through the RCP labyrinth seals.

The excess letdown heat exchanger can be employed either when normal letdown is temporarily out of service to maintain the reactor in operation or it can be used to supplement maximum letdown during the final stages of heatup. The letdown flows through the tube side of the unit and CCW is circulated through the shell side. All surfaces in contact with reactor coolant are austenitic stainless steel, and the shell is carbon steel. All tube joints are welded.

9.3.4.2.9.10 Seal Water Heat Exchanger

The seal water heat exchanger is designed to cool fluid from three sources: RCP seal water returning to the CVCS, reactor coolant discharged from the excess letdown heat exchanger, and CCP bypass flow. Reactor coolant flows through the tube side of the heat exchanger, and CCW is circulated through the shell side. The design flow rate is equal to the sum of the excess letdown flow, maximum design RCP seal leakage, and bypass flow from the CCPs. The unit is designed to cool the above flow to the temperature normally maintained in the VCT. All surfaces in contact with reactor coolant are austenitic stainless steel, and the shell is carbon steel.

9.3.4.2.9.11 Volume Control Tank

The VCT provides surge capacity for part of the reactor coolant expansion volume not accommodated by the pressurizer. When the level in the tank reaches the high-level setpoint, the remainder of the expansion volume is accommodated by diversion of the letdown stream to the LHUTs. It also provides a means for introducing hydrogen into the coolant to maintain the required equilibrium concentration, is used for degassing the reactor coolant, and serves as a head tank for the suction of the CCPs.

A spray nozzle located inside the tank on the letdown line nozzle provides liquid-to-gas contact between the incoming fluid and the hydrogen atmosphere in the tank.

For degassing, the tank is provided with a remotely operated solenoid valve backed up by a pressure control valve, which ensures that the tank pressure does not fall below minimum operating pressure during degassing to the waste disposal system. Relief protection, gas space sampling, and nitrogen purge connections are also provided. The tank can also accept the seal water return flow from the RCPs, although this flow normally goes directly to the suction of the CCPs.

9.3.4.2.9.12 Boric Acid Tanks

The combined boric acid tank capacity is sized to store sufficient boric acid solution for a cold shutdown from full power operation immediately following refueling with the most reactive RCCA not inserted, plus operating margins.

The concentration of boric acid solution in storage is maintained between 4.0 and 4.4 percent by weight (7,000 to 7,700 ppm boron). Periodic manual sampling and corrective action, if necessary, ensure that these limits are maintained. As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the concentration.

Each of two electric heaters in each boric acid tank is designed to maintain the temperature of the boric acid solution at 165°F with ambient air temperature of 40°F, thus ensuring a temperature in excess of the solubility limit. Heater controls maintain the temperature of the boric acid solution at nominally between 110°F and 120°F.

The boric acid tanks can be filled from the batch tank.

9.3.4.2.9.13 Batching Tank

The batching tank is used for mixing a makeup supply of boric acid solution for transfer to the boric acid tanks or the boric acid reserve tanks. The tank may also be used for solution storage.

A local sampling point is provided for verifying the solution concentration prior to transferring it out of the tank. The tank is provided with an agitator to improve mixing during batching operations and a means for heating the boric acid solution.

9.3.4.2.9.14 Chemical Mixing Tank

The primary use of the chemical mixing tank is in the preparation of lithium hydroxide solutions for pH control and hydrazine for oxygen scavenging.

9.3.4.2.9.15 Liquid Holdup Tanks

A total of five LHUTs are provided for DCPP Unit 1 and Unit 2. Two of these tanks serve Unit 1, and two serve Unit 2. The fifth tank can be used with either Unit 1 or Unit

2. The LHUTs hold radioactive liquid, which enters from the letdown line. The liquid is released from the RCS during startup, shutdowns, load changes, and from boron dilution to compensate for burnup. The contents of one tank are normally being processed by the ion exchangers while the other tank is being filled. The tank shared by Unit 1 and Unit 2 is typically kept empty to provide additional storage capacity, when needed, and can be used to store supplemental refueling water.

The total liquid storage capacity of three LHUTs is approximately equal to two RCS volumes. The tanks are constructed of austenitic stainless steel.

9.3.4.2.9.16 Boric Acid Reserve Tanks

Two boric acid reserve tanks are provided for storage of boric acid to meet operational needs for a ready supply of boric acid solution. One tank is maintained on a short recirculation through an inline circulation heater to maintain the tank contents and the associated piping and equipment above 80°F. The other tank is maintained on long recirculation, which also includes the transfer piping in the recirculation loop. Recirculation is normally accomplished by using either one of the two installed recirculation pumps. A transfer pump is provided to send boric acid to the boric acid tank or to the batch tank.

Flush water can be provided from the MWS to flush the boric acid from the piping, through the flush bypass line, to the LHUTs. In addition, water from the boric acid reserve tank can also be pumped to the processed waste receiver or to the LHUTs via installed connections.

The tanks are provided with a Hypalon coated floating cover that will prevent absorption of oxygen by the boric acid solution. In addition, the annular space around the side of the cover between the fluid surface and the bladder attachment point is continuously purged with nitrogen to further reduce the absorption of oxygen by the boric acid. The nitrogen purge vents out vent holes in the bladder attachment bar inside the tank and then vents to the room atmosphere through the tank vent.

9.3.4.2.9.17 Mixed Bed Demineralizers

Two flushable mixed bed demineralizers assist in maintaining reactor coolant purity. A lithium-form cation resin and hydroxyl-form anion resin are charged into the demineralizers. Both forms of resin remove fission and corrosion products. The resin bed is designed to reduce the concentration of ionic isotopes in the purification stream, except for cesium, yttrium, and molybdenum, by a minimum factor of 10.

Each demineralizer nominally has sufficient capacity for approximately one core cycle with 1 percent defective fuel rods. One demineralizer serves as a standby unit for use if the operating demineralizer becomes exhausted during operation.

The normal resin volume is 30 cubic feet per demineralizer. The maximum resin volume is 39 cubic feet per demineralizer. Resin volumes greater than 30 cubic feet cannot be regenerated in the vessel and must be flushed when no longer needed. Resin volumes less than 30 cubic feet may be used for special resins or to meet various requirements.

9.3.4.2.9.18 Cation Bed Demineralizer

The flushable cation resin bed in the hydrogen form is located downstream of the mixed bed demineralizers and is used intermittently to control the concentration of Li^7 which builds up in the coolant from the B^{10} (n, α) Li^7. The demineralizer also has sufficient capacity to maintain the cesium-137 concentration in the coolant below 1 μCi/cc with 1 percent defective fuel. The resin bed is designed to reduce the concentration of ionic isotopes, particularly cesium, yttrium, and molybdenum, by a minimum factor of 10.

The cation bed demineralizer has sufficient capacity for approximately one core cycle with 1 percent defective fuel rods.

9.3.4.2.9.19 Deborating Demineralizers

When required, two anion demineralizers remove boric acid from the RCS fluid. The demineralizers are provided for use near the end of a core cycle, but can be used at any time when boron concentration is low. As an alternative, one of these demineralizers may be filled with a mixed bed and used for removal of radionuclides during forced oxygenation.

The normal resin volume is 30 cubic feet per demineralizer. The maximum resin volume is 39 cubic feet per demineralizer. Resin volumes greater than 30 cubic feet cannot be regenerated in the vessel and must be flushed when no longer needed.

Hydroxyl-based ion exchange resin is used to reduce RCS boron concentration by releasing a hydroxyl ion when a borate is absorbed. Facilities are provided for regeneration. When regeneration is no longer feasible, the resin is flushed to the spent resin storage tank.

The demineralizers are sized to remove approximately 100 ppm of boric acid from the RCS to maintain full power operation near the end of core life should the LHUTs be full.

9.3.4.2.9.20 Evaporator Feed Ion Exchangers

Two trains of ion exchangers purify the feed and routes it to the LRS. Each train consists of two demineralizer vessels in series. The resin beds in these demineralizers remove cationic impurities including cesium and molybdenum, and anionic impurities including chlorides, fluorides, and sulfur species. One train is in service during evaporator operation and the other is on standby.

9.3.4.2.9.21 Reactor Coolant Letdown Filter

The reactor coolant letdown filter is located on the letdown line upstream of the VCT. The filter collects resin fines and particulates from the letdown stream. The nominal flow capacity of the filter is equal to the maximum purification flow rate. A redundant reactor coolant letdown filter has been installed as a standby.

9.3.4.2.9.22 Seal Water Injection Filters

Two seal water injection filters are located in parallel in a common line to the RCP seals; they collect particulate matter that could be harmful to the seal faces. Each filter is sized to accept flow in excess of the normal seal water requirements. One filter is normally in operation and the other is in standby.

9.3.4.2.9.23 Seal Water Return Filter

The seal water return filter collects particulates from the RCP seal water return and from the excess letdown flow. The filter is designed to pass flow in excess of the sum of the excess letdown flow and the maximum design leakage from the RCP seals.

9.3.4.2.9.24 Boric Acid Filter

The boric acid filter collects particulates from the boric acid solution being pumped to the CCP suction line or boric acid blender. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously.

9.3.4.2.9.25 Ion Exchange Filter

This filter collects resin fines and particulates from the gas stripper feed pumps and routes the effluent to the LHUTs or the LRS.

9.3.4.2.9.26 Boric Acid Reserve Tank Recirculation Heaters

Two, 12-kW heaters are provided in the boric acid reserve tank recirculation paths (one heater per path) to heat the boric acid. This will in turn maintain the tank contents and the associated recirculation piping above 80°F. The heaters are controlled by a temperature controller that senses tank temperature and regulates the power supplied to the heaters accordingly. An over temperature controller is provided which senses heater temperature and cuts power to the heater when a high temperature is sensed at the heating element.

9.3.4.2.9.27 Boric Acid Blender

The boric acid blender promotes thorough mixing of boric acid solution and reactor makeup water for the reactor coolant makeup circuit. The blender consists of a conventional pipe tee fitted with a perforated tube insert. The blender decreases the pipe length required to homogenize the mixture for taking a representative local sample. A sample point is provided in the piping just downstream of the blender.

9.3.4.2.9.28 Letdown Orifices

The three letdown orifices are arranged in parallel and serve to reduce the pressure of the letdown stream to a value compatible with the letdown heat exchanger design. Two of the three are sized such that either can pass normal letdown flow; the third can pass less than the normal letdown flow. One or both standby orifices may be used with the normally operating orifice in order to increase letdown flow such as during reactor heatup operations. This arrangement also provides a full standby capacity for control of letdown flow.

9.3.4.2.9.29 Gas Stripper-Boric Acid Evaporator Package

Liquid effluent in the LHUTs is processed as a batch operation. This liquid is pumped by the gas stripper feed pumps through the evaporator feed ion exchangers. It then flows through the ion exchanger filter, and then the liquid is drained to the LRS for processing. The boric acid evaporator system is abandoned in place and no longer in use.

9.3.4.2.9.30 Electric Heat Tracing

Electric heat tracing is installed under the insulation on piping, valves, line-mounted instrumentation, and components normally containing concentrated boric acid solution. The heat tracing is designed to prevent boric acid precipitation due to cooling, by compensating for heat loss. Even though the heat tracing is not required to maintain the 4 percent boric acid solution above the 65°F precipitation temperature, it does provide added assurance against falling below this limit. The existing boric acid heat tracing provides two parallel PG&E Design Class II heater circuits in a bifilar arrangement. One circuit is used for normal operations and the second serves as a backup. The parallel circuits are supplied from different power sources to provide power supply redundancy. The size of the section heated by each pair of circuits is determined by the capacity of the heaters and their feeders, as well as the temperatures required.

There are no heat tracings on:

(1) Lines that may transport concentrated boric acid but are subsequently flushed with reactor coolant or other liquid of low boric acid concentration during normal operation

(2) The boric acid tanks, which are provided with immersion heaters

(3) The boric acid reserve tanks, which are provided with inline recirculation heaters

(4) The batching tank, which is provided with a steam jacket

Each circuit is controlled independently by a thermostat at a location having the most representative temperature of the circuit. The backup circuits are set to operate at a slightly lower temperature than the normal circuits. A thermocouple to detect metal temperature is installed under the thermal insulation near the thermostats. The thermocouple monitor has local indication and stored data memory that may be downloaded to commercial spreadsheet software and printed. The thermocouples are monitored, and both low and high temperatures are alarmed. The low-temperature alarm setpoint is within the operating range of the backup heater.

The boric acid tank sample lines to the sample sink are provided with a single circuit of self-regulating heat trace. Because the precipitation temperature of 4 percent boric acid solution is below normal room temperature, the boric acid line heat tracing functions as a precautionary measure, and is PG&E Design Class II. The circuit is monitored by an indicating light.

9.3.4.2.9.31 Valves

Valves, other than diaphragm valves, that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection. Valves are normally installed such that, when closed, pressure is not on the packing. Basic material of construction is stainless steel for all valves. Isolation valves are provided for all lines entering the reactor containment. These valves are discussed in detail in Section 6.2.4.

Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction. Relief valves with potentially contaminated effluent are collected and routed to tanks in order to minimize radioactive releases inside the auxiliary building.

<u>Charging Line Downstream of Regenerative Heat Exchanger</u>

If the charging side of the regenerative heat exchanger is isolated while the hot letdown flow continues at its maximum rate, the volumetric expansion of coolant on the charging side of the heat exchanger is relieved to the RCS through a spring-loaded check valve. The spring in the valve is designed to permit the check valve to open in the event that the differential pressure exceeds the design pressure differential of approximately 75 psi.

Letdown Line Downstream of Letdown Orifices

The pressure-relief valve downstream of the letdown orifices protects the low-pressure piping and the letdown heat exchanger from overpressure when the low-pressure piping is isolated. The capacity of the relief valve exceeds the maximum flow rate through all letdown orifices. The valve set pressure is equal to the design pressure of the letdown heat exchanger tube side.

Letdown Line Downstream of Low-pressure Letdown Valve

The pressure-relief valve downstream of the low-pressure letdown valve protects the low-pressure piping, demineralizers, and filter from overpressure when this section of the system is isolated. The overpressure may result from leakage through the low-pressure letdown valve. The capacity of the relief valve exceeds the maximum flow rate through all letdown orifices. The valve set pressure is equal to the design pressure of the demineralizers.

Volume Control Tank

The relief valve on the VCT permits the tank to be designed for a lower pressure than the upstream equipment. This valve has a capacity greater than the summation of the following items: maximum letdown, maximum seal water return, excess letdown, and nominal flow from one reactor makeup water pump. The valve set pressure equals the design pressure of the VCT.

Charging Pump Suction

A relief valve on the CCP suction header relieves pressure that may build up if the suction line isolation valves are closed or if the system is overpressurized. The valve set pressure is equal to the design pressure of the associated piping and equipment.

Seal Water Return Line (Inside Containment)

This relief valve is designed to relieve overpressurization in the seal water return piping inside the containment if the motor-operated isolation valve is closed. The valve is designed to relieve the total leakoff flow from the No. 1 seals of the RCPs plus the design excess letdown flow. The valve is set to relieve at the design pressure of the piping.

Seal Water Return Line (Charging Pumps Bypass Flow)

This relief valve protects the seal water heat exchanger and its associated piping from overpressurization. If either of the isolation valves for the heat exchanger is closed and if the bypass line is closed, the piping may be overpressurized by the bypass flow from the CCPs. It is assumed that all CCPs are running with full bypass flow. The valve is set to relieve at the design pressure of the heat exchanger.

9.3.4.2.9.32 Piping

All CVCS piping handling radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal for maintenance and hydrostatic testing.

9.3.4.2.9.33 Zinc Injection Sub-systems

The CVCS includes a skid-mounted zinc injection sub-system that provides the capability to inject a zinc acetate solution to a line leading to the VCT. The chemical solution is injected into the PG&E Design Class II portion of the CVCS.

This sub-system includes a storage and mixing tank, three chemical feed pumps with electrical controls, and the associated piping, tubing, valves, controls, and instrumentation. The tank has a divider that separates the chemical supply for Unit 1 and Unit 2. There is one chemical feed pump dedicated to each unit and a common pump that can serve either Unit 1 or Unit 2. The valves and pumps associated with this sub-system are all manually controlled from the skid. A nitrogen blanket is provided in the tank to exclude oxygen. Make-up water to prepare the zinc acetate solution is provided from the PWST. Power is provided from a non-Class 1E power source.

9.3.4.2.9.34 Argon Injection Sub-systems

The CVCS includes a wall mounted argon injection sub-system that provides the capability to inject high purity argon into the zinc acetate supply line. The zinc acetate injection sub-system is described in 9.3.4.2.9.33. The argon injection sub-system is PG&E Design Class II and connects to a PG&E Design Class II portion of the CVCS.

Argon injection aids in the identification of small primary to secondary SG tube leaks. Trace amounts of argon is injected to generate a short half-life isotope (Ar-41) that can easily be detected by the condenser off gas steam jet air ejector radiation monitor in the event of an SG tube leak.

This sub-system includes an argon bottle, regulator, valves and controls connected via tubing. A flow indicating controller and relief valve is installed downstream from the regulator to control the injection rate and provide over pressure protection. A check valve is located upstream of the zinc injection system tie-in. Other valves associated with this sub-system are all manually controlled from the skid. Power for the flow controller is provided from a non-Class 1E power source.

9.3.4.2.10 System Operation

9.3.4.2.10.1 Chemical and Volume Control System Operation During Reactor Startup

Reactor startup is defined as the operations which bring the reactor from cold shutdown to normal operating temperature and pressure. Reactor pressure vessel heatup and cooldown and compliance with ASME BPVC Section III–1972 Summer Addenda, Appendix G, is discussed in Section 5.2.

It is assumed that:

(1) Normal RHR is in progress.

(2) RCS boron concentration is at or above the cold shutdown concentration.

(3) RMCS is set to provide makeup at or above the cold shutdown concentration.

(4) RCS is either water-solid or drained to minimum level for the purpose of refueling or maintenance. If the RCS is water-solid, system pressure is controlled by letdown through the RHR system and through the letdown pressure control valve.

(5) The charging and letdown lines of the CVCS are filled with coolant at the cold shutdown boron concentration. The letdown orifice isolation valves are closed.

If the RCS requires filling and venting, the general process is as follows (depending on plant conditions, the startup process may vary):

(1) One CCP is started, which provides blended flow from the RMCS (or RWST) at the cold shutdown boron concentration.

(2) The vents on the head of the RV and pressurizer are vented.

(3) The various portions of the RCS are filled and the vents closed.

The CCP and the low-pressure letdown valve continue to pressurize the system. When the system pressure is adequate for operation of the RCPs, seal water flow to the pumps is established (if not already in service) and the pumps are sequentially operated and the RCS is vented until all gases are cleared from the system.

After the filling and venting operations are completed, pressurizer heaters are energized and the RCPs are operated to heat up the system. After the RCPs are started, the RHR pumps are stopped, but pressure control via the RHR system and the low-pressure letdown line is continued. At this point, steam formation in the pressurizer is accomplished by adjusting charging flow and the pressurizer pressure controller. When the pressurizer level reaches the no-load programmed setpoint, the pressurizer level is controlled to maintain the programmed level. The RHR system is then configured in its ECCS alignment.

The reactor coolant boron concentration is now reduced by operating the RMCS in the "dilute" mode. The reactor coolant boron concentration is adjusted to the point where the RCCAs may be withdrawn and criticality achieved. Power operation may then proceed with corresponding manual adjustment of the reactor coolant boron concentration to balance the temperature coefficient effects and maintain the RCCAs within their operating range. During operation, the appropriate combination of letdown orifices is used to provide necessary letdown flow.

Prior to and during the heatup process, the CVCS is employed to obtain the correct chemical properties in the RCS. The RMCS is operated on a continuing basis to ensure correct RCCA position. Chemicals are added through the chemical mixing tank, as required, to control reactor coolant chemistry such as pH and dissolved oxygen content. Hydrogen overpressure is established in the VCT to ensure the appropriate hydrogen concentration in the reactor coolant.

9.3.4.2.10.2 Power Generation and Hot Standby Operation

<u>Base Load</u>

At a constant power level, the rates of charging and letdown are dictated by the requirements for seal water to the RCPs and the normal purification of the RCS. Typically, one CCP is employed and charging flow is controlled automatically from pressurizer level. The only adjustments in boron concentration necessary are those to compensate for core burnup. Rapid variations in power demand are accommodated automatically by RCCA movement. If variations in power level occur, and the new power level is sustained for long periods, some adjustment in boron concentration may be necessary to maintain the RCCAs at their desired position.

During typical operation, normal letdown flow is maintained and one mixed bed demineralizer is in service. Reactor coolant samples are taken periodically to check boron concentration, water quality, pH, and activity level. The CCP flow to the RCS is controlled by the pressurizer level control signal through the discharge header flow control valve.

Load Follow (not normally performed)

A power reduction will initially cause a xenon buildup followed by xenon decay to a new, lower equilibrium value. The reverse occurs if the power level increases; initially, the xenon level decreases and then it increases to a new and higher equilibrium value associated with the amount of the power level change.

The RMCS is used to vary the reactor coolant boron concentration to compensate for xenon transients occurring when reactor power level is changed.

One indication available to the plant operator (enabling him to determine whether dilution or boration of the RCS is necessary) is the position of the RCCAs within the desired band. If, for example, the RCCAs are moving down into the core and are approaching the bottom of the desired band, the operator must borate the reactor coolant to bring the RCCAs outward. If not, the RCCAs may move into the core beyond the insertion limit. If, on the other hand, the RCCAs are moving out of the core, the operator dilutes the reactor coolant to keep the RCCAs from moving above the top of the desired band.

During periods of plant loading, the reactor coolant expands as its temperature rises. The pressurizer absorbs most of this expansion as the level controller raises the level setpoint to the increased level associated with the new power level. The remainder of the excess coolant is let down and may be accommodated in the VCT or LHUTs. During this period, the flow through the letdown orifice remains constant and the charging flow is reduced by the pressurizer level control signal, resulting in an increased temperature at the regenerative heat exchanger outlet. The temperature controller downstream from the letdown heat exchanger increases the CCW flow to maintain the desired letdown temperature.

During periods of plant unloading, the charging flow is increased to make up for the coolant contraction not accommodated by the programmed reduction in pressurizer level.

Hot Standby

If required for periods of maintenance or following reactor trips, the reactor can be held subcritical, but with the capability to return to full power within the period of time it takes to withdraw RCCAs. During this hot standby period, temperature is maintained at no-load T_{avg} by dumping steam to remove core residual heat.

Following shutdown, xenon buildup and decay results in a variation in the degree of shutdown. During this time, boration and dilution of the system are performed to counteract these xenon variations.

9.3.4.2.10.3 Reactor Shutdown

Reactor shutdown is defined as the operations that bring the reactor to cold shutdown for maintenance or refueling.

Before initiating a cold shutdown, the RCS hydrogen concentration is reduced by reduction of the VCT overpressure and venting the gases to the waste gas vent header.

Before cooldown and depressurization of the reactor plant is initiated, the reactor coolant boron concentration is increased to the value required for the corresponding target temperature. The operator uses the RMCS to add the volume of concentrated boric acid solution necessary to perform the boration. After the boration is completed, the operator uses the RMCS to maintain the desired reactor coolant boron concentration. Subsequent reactor coolant samples are taken to verify that the RCS boron concentration is correct.

Contraction of the coolant during cooldown of the RCS results in actuation of the pressurizer level control to maintain normal pressurizer water level. The charging flow is increased, relative to letdown flow, and results in a decreasing VCT level. The RMCS initiates makeup to maintain the inventory.

Coincident with plant cooldown, a portion of the reactor coolant flow may be diverted from the RHR system to the CVCS for cleanup. Demineralization of ionic radioactive impurities and stripping of fission gases reduce the reactor coolant activity level sufficiently to permit personnel access for refueling or maintenance operations.

9.3.4.3 Safety Evaluation

9.3.4.3.1 General Design Criterion 2, 1967 – Performance Standards

All CVCS components are located within the PG&E Design Class I auxiliary and containment buildings. These buildings, or applicable portions thereof, are designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), earthquakes (refer to Section 3.7), and other natural phenomena, to protect CVCS SSCs, ensuring their safety functions will be performed.

The CVCS is designed to perform its function of providing shutdown capability under DDE and HE loading. The seismic requirements are defined in Sections 3.7 and 3.10, and the provisions to protect the system from seismic damage are discussed in Sections 3.7, 3.9, and 3.10. CVCS components are designed to withstand the appropriate seismic loadings in accordance with their design class. CVCS components required for safe shutdown are PG&E Design Class I.

9.3.4.3.2 General Design Criterion 3, 1971 – Fire Protection

The CVCS is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.3.4.3.3 General Design Criterion 4, 1967 – Sharing of Systems

The CVCS components which are shared by both Unit 1 and Unit 2 are the boric acid batching tank, the boric acid reserve tanks, the boric acid reserve tank pumps, the boric acid reserve tank recirculation pumps, and the zinc injection tank. Other components which can be aligned to either Unit 1 or Unit 2 include the LHUTs (refer to Section 9.3.4.2.9.15), LHUT recirculation pumps, and the zinc injection pumps. Administrative controls are in place to ensure that PG&E Design Class I components are not aligned to both Unit 1 and Unit 2 at the same time through the use of closed isolation valves. These administrative controls ensure that CVCS safety functions are not affected by the sharing.

9.3.4.3.4 General Design Criterion 9, 1967 – Reactor Coolant Pressure Boundary

The CVCS provides a means for controlling the RCS water chemistry to minimize corrosion and protect the RCS pressure boundary (refer to Section 5.2.2.3.4). The CVCS is designed with provisions for pH control, oxygen control, and purification of the RCS. The CVCS provides a means for adding chemicals to the RCS to control the pH of the coolant during initial startup and subsequent operation (refer to Section 9.3.4.2.8.2.1), scavenge oxygen from the coolant during startup, and control the oxygen level of the reactor coolant due to radiolysis during all operations subsequent to startup (refer to Sections 9.3.4.2.4 and 9.3.4.2.8.2.2). The CVCS removes fission products and corrosion products from the reactor coolant during operation of the reactor and maintains these within acceptable levels (refer to Sections 9.3.4.2.3 and 9.3.4.2.8.2.3).

9.3.4.3.5 General Design Criterion 11, 1967 – Control Room

Instrumentation, alarms, and controls are provided in the control room for operators to monitor and maintain CVCS parameters. In the event that control room access is lost, CVCS components required for safe shutdown which can be manually controlled from the HSP are: CCP1 and CCP2, boric acid transfer pumps, emergency borate valve 8104, charging flow control valves FCV-128 and HCV-142, and the letdown isolation valves. Refer to Section 7.4.2.1 for additional discussion.

The HSP also provides indication for the following CVCS parameters: RCP seal No. 1 differential pressure, emergency boration flow, VCT level, letdown flow, and charging header pressure and flow.

Instrumentation and alarms for the CVCS is further discussed in Section 9.3.4.3.6.

9.3.4.3.6 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Process control instrumentation is provided to acquire data concerning key parameters about the CVCS. The location of the instrumentation is shown in Figure 3.2-8.

The instrumentation furnishes input signals for monitoring and/or alarming purposes. Indications and/or alarms are provided for the following parameters:

(1) Temperature

(2) Pressure

(3) Flow

(4) Water level

The instrumentation also supplies input signals for control purposes. Some specific control functions are:

(1) Letdown flow is diverted to the VCT upon high-temperature indication upstream of the mixed bed demineralizers

(2) Pressure downstream of the letdown heat exchangers is controlled to prevent flashing of the letdown liquid

(3) Charging flow rate is controlled during CCP operation(refer to Section 9.3.4.2.9.1)

(4) Water level is controlled in the VCT

(5) Temperature of the boric acid solution in the batching tank is maintained

(6) Reactor makeup is controlled

Refer to Section 9.3.4.2.8.2.4 for a discussion of alarm functions.

Reactor coolant samples are taken to verify that the RCS boron concentration is within administrative limits. RCS boron concentration is maintained using the RMCS.

9.3.4.3.6.1 Regenerative Heat Exchanger

The temperatures of both outlet streams from the heat exchanger are monitored with indication given in the control room. High alarm is given on the main control board if the temperature of the letdown stream exceeds desired limits.

Excessive pressure in the letdown line at the regenerative heat exchanger would be indicated in the control room by signals from:

(1) Pressure sensors located on Loop 4 of the RCS

(2) Pressure sensors located on the pressurizer

9.3.4.3.6.2 Letdown Heat Exchanger

The letdown temperature control indicates and controls the temperature of the letdown flow exiting from the letdown heat exchanger. The temperature sensor, which is part of the CVCS, provides input to the controller in the CCW system. The exit temperature is controlled by regulating the CCW flow through the letdown heat exchanger by using the control valve located in the CCW discharge line. Temperature indication is provided on the main control board. Abnormally high temperature on the letdown line downstream of the regenerative heat exchanger or the letdown heat exchanger is indicated by a high-temperature alarm.

Pressure in the letdown line at the letdown heat exchanger is indicated in the control room by signals from the pressure sensor in the letdown line downstream of the letdown heat exchanger. Excessive pressure could lift the relief valve located downstream of the letdown orifices. This would be indicated on the main control board by a temperature sensor located in the relief valve of the relief discharge line.

9.3.4.3.6.3 Excess Letdown Heat Exchanger

A temperature detector measures temperature of excess letdown downstream of the excess letdown heat exchanger. High-temperature alarm and indication are provided on the main control board.

A pressure sensor indicates the pressure of the excess letdown flow downstream of the excess letdown heat exchanger and excess letdown control valve. Pressure indication is provided on the main control board.

9.3.4.3.6.4 Volume Control Tank

VCT pressure and temperature are monitored with indication given in the control room. Alarm is given in the control room for high- and low-pressure conditions and for high temperature. Two level channels govern the water inventory in the VCT. These channels provide local and remote level indication, level alarms, level control, makeup control, and emergency makeup control.

If the VCT level rises above the normal operating range, one channel provides an analog signal to a proportional controller, which modulates the three-way valve downstream of the reactor coolant letdown filter to maintain the VCT level within the normal operating band. The three-way valve can split letdown flow so that a portion goes to the LHUTs and a portion to the VCT. The controller would operate in this fashion during a dilution operation when primary makeup water is being fed to the VCT from the RMCS.

If the modulating function of the channel fails and the VCT level continues to rise, then the high-level alarm will alert the operator to the malfunction and the letdown flow can be manually diverted to the LHUTs. If no action is taken by the operator and the tank level continues to rise, the letdown flow will be automatically diverted to protect the tank from an overpressure condition.

During normal power operation, a low level in the VCT initiates automatic makeup, which injects a preselected blend of boron and water into the CCP suction header. When the VCT is restored to normal, automatic makeup stops.

If the automatic makeup fails or is not aligned for operation and the tank level continues to decrease, a low-level alarm is actuated. Manual action may correct the situation or, if the level continues to decrease, an emergency low-level signal from both channels opens the stop valves in the refueling water supply line and closes the stop valves in the VCT outlet line.

9.3.4.3.6.5 Boric Acid Tanks

A temperature sensor provides temperature measurement of each tank's contents. Local temperature indication is provided as well as high- and low-temperature alarms which are indicated on the main control board. For boric acid heater controls, refer to Section 9.3.4.2.9.12.

A level detector indicates the level in each boric acid tank. Level indication with high, low, and low-low level alarms is provided on the main control board. The low alarm is set to indicate the minimum level of boric acid in the tank to ensure sufficient boric acid to provide for a cold shutdown with one stuck RCCA.

9.3.4.3.6.6 Boric Acid Reserve Tanks

Recirculating flow indication and a low flow alarm are provided for each tank. Also provided are a tank low temperature alarm and a high-low tank level alarm. For controls associated with the boric acid reserve tank recirculation heaters, refer to Section 9.3.4.2.9.26.

9.3.4.3.6.7 Mixed Bed Demineralizers

A temperature sensor measures temperature of the letdown flow downstream of the letdown heat exchanger and controls the letdown flow to the mixed bed demineralizers by means of a three-way valve. If the letdown temperature exceeds the allowable resin operating temperature, the flow is automatically bypassed around the demineralizers.

Temperature indication and high alarm are provided on the main control board. The air-operated three-way valve failure mode directs flow to the VCT.

9.3.4.3.6.8 Reactor Coolant Letdown Filter

Two local pressure indicators are provided to show the pressures upstream and downstream of the reactor coolant letdown filter and thus provide filter differential pressure.

9.3.4.3.6.9 Seal Water Injection Filters

A differential pressure indicator monitors the pressure drop across each seal water injection filter and gives local indication with high differential pressure alarm on the main control board.

9.3.4.3.6.10 Seal Water Return Filter

Two local pressure indicators are provided to show the pressures upstream and downstream of the filter and thus provide the differential pressure across the filter.

9.3.4.3.6.11 Boric Acid Filter

The condition of the filter can be ascertained using a local differential pressure indicator.

9.3.4.3.6.12 Ion Exchange Filter

Local pressure indicators indicate the pressure upstream and downstream of the filter and thus provide filter differential pressure.

9.3.4.3.6.13 Letdown Orifices

Letdown flow rate is controlled by orifices, which are placed in and taken out of service by remote-manual operation of their respective isolation valves. A flow monitor provides indication in the control room of the letdown flow rate and high alarm to indicate unusually high flow.

A low-pressure letdown controller controls the pressure downstream of the letdown heat exchanger to prevent flashing of the letdown liquid. Pressure indication and high-pressure alarm are provided on the main control board.

9.3.4.3.6.14 Electric Heat Tracing

For details of the electric heat tracing instrumentation and controls, refer to Section 9.3.4.2.9.30.

9.3.4.3.6.15 Zinc Injection and Argon Injection Sub-systems

For details of the instrumentation and controls associated with the Zinc Injection and Argon Injection Sub-systems, refer to Sections 9.3.4.2.9.33 and 9.3.4.2.9.34, respectively.

9.3.4.3.7 General Design Criterion 13, 1967 – Fission Process Monitors and Controls

Control over the fission process for each reactor will be maintained throughout the core life by the combination of RCCAs (refer to Chapter 4) and chemical shim (boration). Long-term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Periodic samples of boron concentration provide fission process information. Refer to Section 7.7.3.3 for additional discussion.

The boron system maintains the reactor in the cold shutdown state independent of the position of the RCCAs and can compensate for all xenon burnout transients, as discussed in Section 9.3.4.3.9.

9.3.4.3.8 General Design Criterion 21, 1967 – Single Failure Definition

Two separate and independent flowpaths are available for reactor coolant boration; i.e., the charging line and the RCP seal injection. A single failure does not result in the inability to borate the RCS. An alternate flowpath is always available for emergency boration of the reactor coolant. Refer to Section 9.3.4.3.26 for discussion of flowpaths required for safe shutdown.

As backup to the normal boric acid supply, the operator can align the RWST outlet to the suction of the CCPs or SI pumps when all the RV head bolts are fully detensioned. If an SI pump is used for boration, it is aligned to take suction from the RWST and discharge to the cold legs of the RCS, and the boundary valves from the CVCS to the SI system are closed. At least one flowpath is available for boron injection whenever fuel is in the reactor, and the capability of such injection is adequate to ensure that cold shutdown can be maintained. Redundant Class 1E power supplies are provided for CCP1 and CCP2, the SI pumps, and the valves in the charging injection flowpath from the RWST to the RCS cold legs. Refer to Section 9.3.4.3.11 for additional information.

Since inoperability of a single component does not impair ability to meet boron injection requirements, plant operating procedures allow components to be temporarily out of service for repairs. However, with an inoperable component, the ability to tolerate additional component failure is limited. Therefore, operating procedures require immediate action to effect repairs of an inoperable component, restrict permissible repair time, and require verification of the operability of the redundant component. Boron injection system operability requirements are administratively controlled.

The reactor will not be made critical unless redundant boration capability is available in quantity sufficient to ensure shutdown to cold conditions.

Flow to the RCPs' seals is ensured by the fact that there are three CCPs, any one of which is capable of supplying the normal charging line flow plus the nominal seal water flow.

9.3.4.3.9 General Design Criterion 26, 1971 – Reactivity Control System Redundancy and Capability

Two independent reactivity control systems are provided for each reactor. These are RCCAs (refer to Chapter 4) and chemical shim (boration) provided by CVCS. The CVCS boron system maintains the reactor in the cold shutdown state independent of the position of the RCCAs and can compensate for all xenon burnout transients.

The CVCS reactivity control system is capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout). During normal power operation, routine boration and dilution of the RCS are administratively controlled to assure acceptable fuel design limits are not exceeded.

Any time that the plant is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown, assuming that the RCCA of greatest worth is in its fully withdrawn position. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay. An adequate quantity of boric acid is also available in the RWST to achieve cold shutdown. Refer to Section 9.3.4.2 for additional detail on xenon transients, CCP net positive suction head, and boric acid solution concentration and temperatures.

9.3.4.3.10 General Design Criterion 28, 1967 – Reactivity Hot Shutdown Capability

The chemical shim control provided by CVCS is one of two independent systems capable of making and holding the core subcritical, but at a slower rate than the other system (RCCAs), and is not employed as a means of compensating for rapid reactivity transients. Refer to Chapter 4 for the details of the RCCA system which is used in protecting the core from fast transients. For additional discussion, refer to Sections 9.3.4.2.1.1 and 9.3.4.2.8.2.4.

An upper limit to the boric acid tank boron concentration and a lower limit to the temperature for the tank and for flowpaths from the tank are specified in order to ensure that solution solubility is maintained. Refer to Section 9.3.4.2.9.12 for additional detail on boric acid tank concentration and temperature.

9.3.4.3.11 General Design Criterion 30, 1967 – Reactivity Holddown Capability

The CVCS boron reactivity (chemical shim) control system is the reactivity control system capable of making and holding the core subcritical under any anticipated condition and with appropriate margin for contingencies. Normal reactivity shutdown capability is provided by rapid RCCA insertion (refer to Chapter 4). The chemical shim control system permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. For additional discussion, refer to Section 9.3.4.3.9.

When the reactor is subcritical; i.e., during cold or hot shutdown, refueling, and approach to criticality, the neutron source multiplication is continuously monitored and indicated. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start a corrective action (boron dilution stop and boration) to prevent the core from becoming critical. The rate of boration, with a single boric acid transfer pump operating, is sufficient to take the reactor from full power operation to 1 percent shutdown in the hot condition, with no RCCAs inserted, in 150 minutes. In an additional 3 hours, enough boric acid can be injected to compensate for xenon decay, although xenon decay below the full power equilibrium level will not begin until approximately 25 hours after shutdown. Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions. Boric acid can also be added by injection through the RCP seals at approximately 5 gpm per pump (20 gpm total). At this rate, enough boric acid solution to compensate for xenon decay is added in less than 5 hours. Refer to Sections 9.3.4.2.1.1 and 9.3.4.2.8.2.4 for additional detail on xenon transients.

In the event of loss of offsite power, the safety (boration) function of the CVCS would be maintained. Power to CCP1 and CCP2 or SI pumps and associated valves would be available from the diesel generators. CCP1 or CCP2 or an SI pump is sufficient to meet boron injection requirements for shutdown. Since each CCP (CCP1 and CCP2) or SI pump is loaded on a separate diesel generator, a single failure of any one diesel generator will not impair the safety function of the pumps. A natural circulation test program was conducted during startup testing to demonstrate that the boron mixing and cooldown functions associated with taking the plant to cold shutdown can be accomplished under natural circulation.

9.3.4.3.12 General Design Criterion 31, 1967 – Reactivity Control Systems Malfunction

For postulated boron dilution during refueling, startup, or with the reactor in manual or automatic operation at power, the operator has ample time to determine the cause of dilution, terminate the source of dilution, and initiate boration before the shutdown margin is lost. The facility reactivity control systems are discussed further in Chapter 7, and analyses of the effects of the other possible malfunctions are discussed in Chapter 15. For additional discussion on boron dilution, refer to Sections 9.3.4.2.1.1 and 9.3.4.2.8.2.4.

9.3.4.3.13 General Design Criterion 32, 1967 – Maximum Reactivity Worth of Control Rods

The maximum rates of reactivity insertion employing boron removal are limited to values that could not cause rupture of the RCS boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The appropriate reactivity insertion rate for the dilution of the boric acid in the RCS is determined by safety analyses for the facility. The data on reactivity insertion rates and dilution, together with RCCA withdrawal limits, are discussed in Section 4.3. The CVCS has the capability to avoid an inadvertent excessive rate of boron dilution. The core and RCS are not adversely affected by the maximum rate of reactivity increase due to boron dilution, which is less than that assumed for RCCA withdrawal analyses (refer to Section 15.2). For additional discussion on boron dilution, refer to Sections 9.3.4.2.1.1 and 9.3.4.2.8.2.4. The reactivity insertion rates due to uncontrolled boron dilution and the evaluation of plant safety is discussed in Section 15.2.4.

9.3.4.3.14 General Design Criterion 40, 1967 – Missile Protection

The provisions taken to protect the containment isolation portion of the CVCS from damage that might result from missiles and dynamic effects associated with equipment and high-energy pipe failures are discussed in Sections 3.5, 3.6, and 6.2.4.

9.3.4.3.15 General Design Criterion 49, 1967 – Containment Design Basis

The CVCS penetrations are designed to withstand the pressures and temperatures that could result from a LOCA without exceeding the design leakage rates. Refer to Section 3.8.2.1.1.3 for additional details.

9.3.4.3.16 General Design Criterion 54, 1971 – Piping Systems Penetrating Containment

The CVCS CIVs are periodically tested for operability and leakage. Testing of the components required for the CIS is discussed in Section 6.2.4. Test connections are provided in the penetration and in the piping to verify valve leakage and penetration leakage are within prescribed limits.

9.3.4.3.17 General Design Criterion 55, 1971 – Reactor Coolant Pressure Boundary Penetrating Containment

The CVCS penetrations that are part of the CIS include the regenerative heat exchanger to the letdown heat exchanger, normal charging to the regenerative heat exchanger, and the RCP seal water supply and return lines, which comply with the requirements of GDC 55, 1971, as described in Section 6.2.4 and Table 6.2-39.

9.3.4.3.18 General Design Criterion 56, 1971 – Primary Containment Isolation

The CVCS penetrations that are part of the CIS include the relief valve header lines, which comply with the requirements of GDC 56, 1971, as described in Section 6.2.4 and Table 6.2-39.

9.3.4.3.19 General Design Criterion 68, 1967 – Fuel and Waste Storage Radiation Shielding

Radiation shielding is provided, as required to meet the requirements of 10 CFR Part 20, for the LHUTs which store radioactive liquid until the batch processing for each tank begins. The radiation shielding is provided by the concrete walls, ceiling, and ground in the auxiliary building. Refer to Section 12.1.2.5 and Table 12.1-2 for details of radiation shielding for CVCS components, including the LHUTs.

9.3.4.3.20 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The CVCS is designed with provisions for maintaining control over radioactive liquid and gaseous effluents. Relief valves with potentially contaminated effluent are collected and routed to tanks in order to minimize radioactive releases inside the auxiliary building. The CVCS is designed with tanks (VCT and LHUTs) that allow for processing and/or appropriate confinement of radioactivity associated with the liquid and gaseous effluents from normal operation. Fission gases are removed from the system by venting the VCT to the waste disposal system. When the level in the VCT reaches the high-level setpoint, the remainder of the expansion volume is accommodated by diversion of the letdown stream to the LHUTs. As liquid enters the LHUTs, the nitrogen cover gas is displaced to the gas decay tanks in the waste disposal system through the waste vent header. Liquid effluent in the LHUTs is processed as a batch operation in the CVCS and then the liquid is drained to the LRS for processing.

The CVCS is capable of reducing the concentration of ionic isotopes in the purification stream, as required in the design basis, to minimize activity released due to leakage. This is accomplished by passing the letdown flow through the mixed bed demineralizers that remove ionic isotopes, except those of cesium, molybdenum, and yttrium, with a minimum decontamination factor of 10. Through occasional use of the cation bed demineralizer, the concentration of cesium can be maintained below 1 μCi/cc, assuming 1 percent of the power is being produced by defective fuel. The cation bed demineralizer is capable of passing the normal letdown flow, though only a portion of this capacity is normally utilized. Each mixed bed demineralizer is capable of processing the maximum letdown flow rate. If the normally operating mixed bed demineralizer's resin has become exhausted, the second demineralizer can be placed in service. Each demineralizer is designed, however, to operate for one core cycle with 1 percent defective fuel.

Administrative and engineering controls are in place for the control of radioactive effluents, as required to meet the requirements of 10 CFR Part 20 (refer to Sections 9.3.4.3.19, 9.4.2.3.13, and 12.2). The LHUTs and VCT are located in containment vaults in the auxiliary building that are sized to contain the full volume of liquid effluents for its respective tank. The VCT is designed with a spray nozzle inside the tank to strip part of the noble gases from the incoming liquid and retain these gases in the VCT vapor space. Gaseous effluents from an LHUT or VCT rupture will be released through the plant vent via the auxiliary building ventilation system (ABVS) (refer to Section 9.4.2.3.13). The radiological consequences of LHUT and VCT ruptures are analyzed in Sections 15.5.25 and 15.5.26, respectively. For additional discussion on CVCS leakage, refer to Section 9.3.4.3.28 (Item III.D.1.1).

9.3.4.3.21 Chemical and Volume Control System Safety Function Requirements

(1) Reactor Coolant System Inventory and Pressure Control

The CVCS maintains the proper coolant inventory in the RCS for all normal modes of operation including startup from cold shutdown, full power operation, and plant cooldown. The VCT provides surge capacity for reactor coolant expansion not accommodated by the pressurizer. For additional discussion, refer to Sections 9.3.4.2.2, 9.3.4.2.8.1, and 9.3.4.2.8.2.4. Pressurizer level control is discussed in Section 9.3.4.2.10.2.

The CVCS, coincident with RHR system operation during plant startup and shutdown, controls the RCS pressure. The CVCS regulates the flow rate to the low-pressure letdown line and the charging flow to control RCS pressure during these modes of operation. Refer to Section 5.5.6.2 for additional details.

The CVCS design includes a flowpath from the regenerative heat exchanger outlet to the pressurizer spray line. An air-operated valve in the spray line is employed to provide auxiliary spray from the CCPs to the vapor space of the pressurizer during plant cooldown. This provides a means of cooling the pressurizer near the end of plant cooldown, when the RCPs are not operating. Refer to Section 5.5.9.3.4 for additional details.

(2) Protection from Missiles

The provisions taken to protect the system from missiles are discussed in Section 3.5. Missile protection for portions of the CVCS which also serve ECCS functions are addressed in Section 6.3.3.8.

(3) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect the PG&E Design Class I portion of the CVCS system from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Section 3.6.

(4) Protection from Moderate Energy Pipe Rupture Effects – Outside Containment

The provisions taken to provide protection of the PG&E Design Class I portion of the CVCS system located outside containment from the effects of moderate energy pipe failure are discussed in Section 3.6.

(5) Protection from Jet Impingement – Inside Containment

The provisions taken to provide protection of the PG&E Design Class I portion of the CVCS system located inside containment from the effects of jet impingement which may result from high energy pipe rupture are discussed in Section 3.6.

(6) Protection from Flooding Effects

The provisions taken to provide protection of the PG&E Design Class I portion of the CVCS system from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

(7) Reactor Coolant Pump Seal Water Injection

The CVCS is able to continuously supply filtered water to each RCP seal, as required by the RCP design. Flow to the RCPs' seals is ensured by the fact that there are three CCPs, any one of which is capable of supplying the normal charging line flow plus the nominal seal water flow. Refer to Section 9.3.4.2.8.1 for a detailed discussion on the seal water system.

9.3.4.3.22 10 CFR 50.49 – Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

The CVCS SSCs required to function in harsh environments under accident conditions are qualified to the applicable environmental conditions to ensure that they will continue to perform their safety functions. Section 3.11 describes the DCPP EQ Program and the requirements for the environmental design of electrical and related mechanical equipment. The affected equipment list includes: pump motors, valve motors, valve operators, solenoid valves, heaters, switches, transmitters, local starters, and indicators and are listed in the EQ Master List.

9.3.4.3.23 10 CFR 50.55a(f) – Inservice Testing Requirements

The IST requirements for the CVCS are contained in the DCPP IST Program Plan.

9.3.4.3.24 10 CFR 50.55a(g) – Inservice Inspection Requirements

The ISI requirements for the CVCS are contained in the DCPP ISI Program Plan and comply with ASME BPVC Section XI-2001 through 2003 Addenda.

9.3.4.3.25 10 CFR 50.63 – Loss of All Alternating Current Power

The CVCS provides RCP seal injection in support of safe shutdown of the plant (Mode 3) following a SBO. With seal injection flow, there is no loss of reactor coolant from the RCP seals. Also, in conjunction with RCP seal cooling provided by the CCW system (refer to Section 9.2.2.3.14). RCP seal cooling provided by seal injection protects the RCP seals from overheating in the event of an SBO event. CCP1 is available on the Class 1E Bus F associated with the credited AAC power source during the SBO event.

Refer to Section 8.3.1.6 for further discussion of station blackout.

9.3.4.3.26 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805 Fire Protection of the CVCS is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition (refer to Section 9.5.1).

9.3.4.3.27 Regulatory Guide 1.97, Revision 3, May 1983 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

CVCS post-accident instrumentation for meeting Regulatory Guide 1.97, Revision 3, requirements consist of CCP1 and CCP2 injection header flow indication, RCS makeup flow indication, RCS letdown flow indication, VCT level indication, and CIV position indication (refer to Table 7.5-6).

9.3.4.3.28 NUREG-0737 (Items I.C.1, II.B.2, II.K.1.5, III.D.1.1), November 1980 – Clarification of TMI Action Plan Requirements

Item I.C.1 – Guidance for the Evaluation and Development of Procedures for Transients and Accidents: NUREG-0737, Supplement 1, January 1983 provides the requirements for I.C.1 as follows:

Section 7.1(b) – Upgraded emergency operating procedures, which include guidance for initiating normal charging and emergency boration, have been implemented in accordance with the Westinghouse Owners Group developed generic emergency response guidelines.

Item II.B.2 – Design Review of Plant Shielding and Environmental Qualification of Equipment for Space/Systems Which May Be Used in Post-accident Operations: The switchgear rooms are sufficiently shielded from external sources of radiation such that personnel access and occupancy would not be unduly limited by the radiation environment caused by a degraded core accident.

Item II.K.1.5 – Safety Related Valve Position: Refer to Section 9.3.4.3.33.

Item III.D.1.1 – Integrity of Systems Outside Containment Likely to Contain Radioactive Material for PWRs and Boiling-Water Reactors: CVCS components, valves, and piping that see radioactive service are designed to permit essentially zero leakage to the atmosphere. The components are provided with welded connections except where flanged connections are provided to permit removal for maintenance.

The VCT in the CVCS provides an inferential measurement of leakage from the CVCS as well as the RCS. Low level in the VCT actuates makeup at the prevailing reactor coolant boron concentration. The amount of leakage can be inferred from the amount of makeup added by the RMCS.

Pressure containing portions of the CVCS are tested periodically to check for leakage. This testing includes the portions of the system flowpath that would circulate radioactive water from the RCS.

The requirements for a leakage reduction program from reactor coolant sources outside containment are included in the Technical Specifications (Reference 1). Inservice valve leak testing requirements are specified in the DCPP IST Program Plan. Refer to Section 6.2.4 for additional information on the CIS.

9.3.4.3.29 Generic Letter 80-21, March 1980 – Vacuum Condition Resulting in Damage to Chemical Volume Control System Holdup Tanks

The CVCS low pressure tanks that can be valved to contain RCS water are the VCT and the LHUTs. Both tanks are protected against failure under vacuum conditions. The VCT is structurally designed to withstand 15 psig external pressure. The LHUTs rely on at least one of several design features functioning to protect the tanks from vacuum conditions. These features include an LHUT discharge pump trip on negative tank pressure, control valves on the independent cover-gas supplies, and a vacuum relief valve on each tank set to open prior to reaching the LHUT allowable external pressure. This meets the requirements of Generic Letter 80-21, March 1980, which was issued as IE Bulletin 80-05, March 1980.

9.3.4.3.30 Generic Letter 88-17, October 1988 – Loss of Decay Heat Removal

The CVCS PG&E Design Class I CCPs are one of the two required available or operable means of adding inventory to the RCS, that are in addition to the normal decay heat removal systems, to meet the requirements of Generic Letter 88-17, October 1988. Makeup to the RCS is provided by a CCP, gravity feed from the RWST, and an SI pump which can quickly be put in service should RHR be lost. Each of these backup sources can provide enough water for decay heat removal in addition to continued core coverage. PG&E updated procedures to ensure these pumps are available prior to entering RHR mid-loop operation.

9.3.4.3.31 Generic Letter 89-10, June 1989 – Safety-Related Motor-Operated Valve Testing and Surveillance

CVCS MOVs are subject to the requirements of Generic Letter 89-10, June 1989, and associated Generic Letter 96-05, September 1996, and meet the requirements of the DCPP MOV Program Plan.

9.3.4.3.32 Generic Letter 95-07, August 1995 – Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves

PG&E Design Class I power-operated gate valves in the CVCS were evaluated and determined to not be susceptible to pressure locking or thermal binding.

9.3.4.3.33 IE Bulletin 79-06A (Positions 8 and 12), April 1979 – Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident

Position (8) (subsequently NUREG-0737, November 1980, Item II.K.1.5, Safety Related Valve Position): Critical manual valves in the CVCS are sealed in position and a check list is maintained for inspection on a typical audit basis. All CVCS PG&E Design Class I valves which are operated remotely and whose purpose is to open or close (rather than throttle flow) have position indicating lights on the main control board.

Position (12): During normal operating modes, hydrogen can be removed from the RCS by the letdown line and stripped in the VCT where it enters the waste gas system.

9.3.4.3.34 NRC Bulletin 88-04, May 1988 – Potential Safety-Related Pump Loss

The CVCS is designed such that PG&E Design Class I pumps that share a common minimum flow recirculation line are not susceptible to the pump-to-pump interaction described in NRC Bulletin 88-04, May 1988, and the existing minimum flow rates for CVCS PG&E Design Class I pumps are adequate.

9.3.4.3.35 NRC Bulletin 88-08, June 1988 – Thermal Stresses in Piping Connected to Reactor Coolant Systems

The unisolable CVCS lines connected to the RCS which are subject to thermal stresses are the charging injection headers, for both DCPP Unit 1 and Unit 2. A recirculation line was added to the charging injection header that provides continuous vent capability of the charging injection header back to the CCP suction. The charging injection header recirculation line is sized such that 1) during normal operation (charging injection header not in service), this passive recirculation line continuously vents valve seepage and accompanying pressure build-up in the charging injection header, and 2) during ECCS operation, charging injection header recirculation line flow is low enough such that ECCS flows are maintained within their required ranges. For additional discussion, refer to Section 6.3.3.40.

9.3.4.4 Tests and Inspections

As part of plant operation, periodic tests, surveillance inspections, and instrument calibrations are made to monitor equipment condition and performance.

Most components are in use regularly; therefore, assurance of the availability and performance of the systems and equipment is provided by control room and/or local indication. Refer to Sections 9.3.4.3.5 and 9.3.4.3.6 for details regarding local and remote indication.

Technical specifications have been established concerning calibration, checking, and sampling of the CVCS.

Refer to Sections 9.3.4.3.16, 9.3.4.3.23, 9.3.4.3.24, and 9.3.4.3.28 (Item III.D.1.1) for details regarding tests and inspections of the CVCS.

Refer to Section 9.3.4.2.6 for details of hydrostatic testing of the RCS.

9.3.4.5 Instrumentation Applications

Refer to Section 9.3.4.3.6 for the instrumentation applications related to CVCS.

9.3.5 FAILED FUEL DETECTION

Failed fuel detection is provided by analyzing reactor coolant grab samples via the nuclear steam supply sampling system.

9.3.6 NITROGEN AND HYDROGEN SYSTEMS

The nitrogen and hydrogen systems supply gases required for cover gases, accumulator fill, certain instrumentation operations, degasification purging, layup of SGs and feedwater heaters, and generator cooling. The nitrogen and hydrogen systems are not required for reactor protection, containment isolation, or ESFs. The design classifications for these systems are given in Table 3.2-3. The only PG&E Design Class I portions of the nitrogen and hydrogen systems are the containment penetrations for the nitrogen system. The supply sources of nitrogen and hydrogen are shared by both units but delivered by separate supply headers. The backup air/nitrogen supply system is described in Section 9.3.1.2.2.

9.3.6.1 Design Bases

9.3.6.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the nitrogen system is designed to withstand the effects of, or is protected against, natural phenomena such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.3.6.1.2 General Design Criterion 3, 1971 – Fire Protection

The nitrogen and hydrogen systems are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.3.6.1.3 General Design Criterion 4, 1967 – Sharing of Systems

The nitrogen and hydrogen systems are not shared by the DCPP units unless it is shown that safety is not impaired by the sharing.

9.3.6.1.4 General Design Criterion 11, 1967 – Control Room

The nitrogen system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.3.6.1.5 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the nitrogen and hydrogen system variables within prescribed operating ranges.

9.3.6.1.6 General Design Criterion 40, 1967 – Missile Protection

The ESF containment isolation portion of the nitrogen system is designed to be protected against dynamic effects and missiles that might result from plant equipment failures.

9.3.6.1.7 General Design Criterion 49, 1967 - Containment Design Basis

The PG&E Design Class I portion of the nitrogen system is designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of ECCSs.

9.3.6.1.8 General Design Criterion 54, 1971 - Piping Systems Penetrating Containment

The PG&E Design Class I portion of the nitrogen system piping penetrating containment is provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. This piping system is designed with a capability to test periodically that operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

9.3.6.1.9 General Design Criterion 55, 1971 – Reactor Coolant Pressure Boundary Penetrating Containment

The PG&E Design Class I portion of the nitrogen system that penetrates the containment is provided with CIVs.

9.3.6.1.10 General Design Criterion 56, 1971 - Primary Containment Isolation

The PG&E Design Class I portion of the nitrogen system contains valves in piping that penetrate containment. One automatic isolation valve is provided outside containment and one check valve inside containment to ensure containment integrity is maintained.

9.3.6.1.11 Nitrogen System Safety Function Requirements

(1) <u>Protection from Jet Impingement – Inside Containment</u>

The PG&E Design Class I containment isolation portion of the nitrogen system located inside containment is designed to be protected against the effects of jet impingement which may result from high energy pipe rupture.

9.3.6.1.12 10 CFR 50.55a(f) – Inservice Testing Requirements

The PG&E Design Class I nitrogen system ASME Code pumps and valves are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

9.3.6.1.13 10 CFR 50.55a(g) – Inservice Inspection Requirements

The PG&E Design Class I nitrogen system ASME BPVC Section XI components are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.3.6.2 System Description

9.3.6.2.1 Nitrogen System

The nitrogen system consists of a liquid nitrogen storage facility, which is the nitrogen source for both the low pressure gaseous nitrogen header and a series of high pressure gaseous nitrogen storage bottles. Nitrogen from the liquid supply tank is (a) gasified and supplied directly to the low pressure header or (b) compressed, then gasified, and supplied to the high pressure storage bottles.

The nitrogen system is capable of delivering nitrogen gas for various purposes. Pressure regulators are capable of reducing the pressure down to 1 psig. This is accomplished through a series of regulators from the supply source to the required equipment. Relief valves are set appropriately to prevent overpressure on the equipment. Refer to Table 9.3-7 for the list of equipment requiring nitrogen and their supply pressures and flows. The pressure and flow data envelope the actual operating conditions.

All nitrogen system piping penetrating the containment not isolated with sealed closed valves is isolated automatically by a containment isolation signal.

9.3.6.2.2 Hydrogen System

The hydrogen system is capable of delivering hydrogen gas for various purposes at a supply pressure of 2200 psig and reduced by pressure regulators in series to the required pressures. Relief valves are set at 2400 psig on the supply header. Other relief valves downstream of pressure regulators are set appropriately to prevent overpressure on the equipment. An excess flow check valve, FCV-39/40, is provided in the hydrogen supply header at each unit. The excess flow check valve (flow fuse) is a flow control device that limits the flow of hydrogen to the plant loads. Refer to Table 9.3-8 for the list of equipment requiring hydrogen and their supply pressures and flows.

9.3.6.3 Safety Evaluation

9.3.6.3.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portions of the nitrogen system are housed within the PG&E Design Class I auxiliary and containment buildings. These buildings, or applicable portions thereof, are designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), earthquakes (refer to Section 3.7), and other natural phenomena.

The PG&E Design Class I nitrogen system components have been seismically qualified.

Portions of the nitrogen system are located outside and are partially resistant to tornadoes. However, the capability of the system to support plant safe shutdown is maintained (refer to Section 3.3.2).

The nitrogen system components required to perform their PG&E Design Class I functions are located at higher elevations such that they are not affected by floods and tsunamis.

9.3.6.3.2 General Design Criterion 3, 1971 – Fire Protection

The nitrogen and hydrogen systems are designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

In fire zones containing equipment required for safe shutdown, the hydrogen system is enclosed in guard pipes and enclosures such that if leaks occur, the leakage is vented to areas that do not contain equipment required for safe shutdown. Enclosures containing valves and instrumentation have flanges or doors that make the instrumentation and valves accessible for operation and maintenance.

9.3.6.3.3 General Design Criterion 4, 1967 – Sharing of Systems

The nitrogen and hydrogen systems are shared by the DCPP units. The PG&E Design Class I portion of the nitrogen system consists of containment penetration piping and isolation valves in Unit 1 and Unit 2 that are not shared between the DCPP units. Only the PG&E Design Class II nitrogen and hydrogen storage facilities and PG&E Design Class II piping are shared between the DCPP units, and this sharing does not affect plant safety.

9.3.6.3.4 General Design Criterion 11, 1967 – Control Room

The nitrogen system is designed to support actions to maintain and control the safe operational status of the plant from the control room and from a remote location.

Low nitrogen header pressure is alarmed in the control room of each unit.

9.3.6.3.5 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the nitrogen and hydrogen system variables within prescribed operating ranges.

Local temperature indicators measure the temperature of the nitrogen and hydrogen supply.

Local pressure indicators measure the pressures in the various lines of the nitrogen and hydrogen systems.

Pressure transmitters located on the nitrogen and hydrogen supply headers are used to transmit the pressure to indicators on the auxiliary building control board. Pressure switches also located on the headers give a low-pressure alarm on the auxiliary building control board.

9.3.6.3.6 General Design Criterion 40, 1967 – Missile Protection

The provisions taken to protect the PG&E Design Class I containment isolation portion of the nitrogen system from damage that might result from missiles and dynamic effects associated with equipment and high-energy pipe failures are discussed in Sections 3.5, 3.6, and 6.2.4.

9.3.6.3.7 General Design Criterion 49, 1967 - Containment Design Basis

The nitrogen piping routed through containment penetrations is designed to withstand the pressures and temperatures that could result from a LOCA without exceeding the design leakage rates. Refer to Sections 3.8.2.1.1.3 and 6.2.4, and Table 6.2-39 for additional details.

9.3.6.3.8 General Design Criterion 54, 1971 - Piping Systems Penetrating Containment

The nitrogen system CIVs are periodically tested for operability and leakage. Testing of components required for the CIS is discussed in Section 6.2.4. Test connections are provided in the penetration and in the piping to verify valve and penetration leakage within prescribed limits.

CIVs associated with penetration 52C (nitrogen supply to the SGs) are exempted from IST.

No piping associated with the hydrogen system penetrates the containment.

9.3.6.3.9 General Design Criterion 55, 1971 – Reactor Coolant Pressure Boundary Penetrating Containment

The nitrogen system penetrations that are part of the CIS include the nitrogen supply lines for the pressurizer relief tanks, which comply with the requirements of GDC 55, 1971, as described in Section 6.2.4 and Table 6.2-39.

9.3.6.3.10 General Design Criterion 56, 1971 - Primary Containment Isolation

The nitrogen system penetrations that are part of the CIS include the nitrogen supply lines for the SI accumulators, the SGs, and the RCDT, which comply with the requirements of GDC 56, 1971, as described in Section 6.2.4 and Table 6.2-39. That portion of the nitrogen system piping penetrating the containment is PG&E Design Class I and meets the single failure criterion required for containment isolation as described in Section 6.2.4 and Table 6.2-39. The remainder of the hydrogen and nitrogen systems is PG&E Design Class II.

9.3.6.3.11 Nitrogen System Safety Function Requirements

(1) <u>Protection from Jet Impingement – Inside Containment</u>

The provisions taken to provide protection of the PG&E Design Class I containment isolation portion of the nitrogen system located inside containment from the effects of jet impingement which may result from high energy pipe rupture are discussed in Section 3.6.

9.3.6.3.12 10 CFR 50.55a(f) – Inservice Testing Requirements

The IST requirements for nitrogen system components are contained in the IST Program Plan and comply with the ASME Code for Operation and Maintenance of Nuclear Power Plants.

9.3.6.3.13 10 CFR 50.55a(g) – Inservice Inspection Requirements

Nitrogen system components are included in the ISI program in accordance with ASME BPVC, Section XI.

9.3.6.4 Tests and Inspections

Periodic inspections are made on the nitrogen and hydrogen systems. Periodic visual inspection and preventive maintenance are made using normal industry practice. Functional and leakage tests are performed on the CIVs.

9.3.6.5 Instrumentation Applications

The instrumentation for the nitrogen and hydrogen systems is discussed in Sections 9.3.6.3.4 and 9.3.6.3.5.

9.3.7 MISCELLANEOUS PROCESS AUXILIARIES

The process auxiliary systems not associated with the reactor process system but necessary for plant operation are:

(1) Auxiliary steam system

(2) OWS and TBS system

The PG&E Design Class I portion of the auxiliary steam system extends from the upstream side of the outboard CIV to the downstream side of the inboard CIV. The remainder of the auxiliary steam system is classified as PG&E Design Class II. The OWS and TBS system is classified as PG&E Design Class II (refer to Section 3.2).

The requirements for the temperature sensors, alarm, and switch associated with the pipe break isolation system (PBIS), used for manual isolation of an auxiliary steam system pipe break, are discussed in Section 7.6.

9.3.7.1 Auxiliary Steam System

Main steam is supplied to the auxiliary steam system through a branch line off of the 35 percent atmospheric dump header through a pressure reducing valve. Each unit can supply steam to the auxiliary steam system either separately or simultaneously.

9.3.7.1.1 Design Bases

9.3.7.1.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the auxiliary steam system is designed to withstand the effects of, or be protected against, natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.3.7.1.1.2 General Design Criterion 3, 1971 - Fire Protection

The PG&E Design Class I portion of the auxiliary steam system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.3.7.1.1.3 General Design Criterion 4, 1967 – Sharing of Systems

The auxiliary steam system is not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.3.7.1.1.4 General Design Criterion 40, 1967 – Missile Protection

The PG&E Design Class I containment isolation portion of the auxiliary steam system is designed to be protected against dynamic effects and missiles that might result from plant equipment failures.

9.3.7.1.1.5 General Design Criterion 49, 1967 - Containment Design Basis

The PG&E Design Class I portion of the auxiliary steam system is designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of ECCSs.

9.3.7.1.1.6 General Design Criterion 54, 1971 – Piping Systems Penetrating Containment

The PG&E Design Class I portion of the auxiliary steam system that penetrates containment is provided with leak detection, isolation, redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating this system. The piping is designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

9.3.7.1.1.7 General Design Criterion 56, 1971 - Primary Containment Isolation

The PG&E Design Class I portion of the auxiliary steam system contains valving in piping that penetrates containment and that is connected directly to the containment atmosphere. A sealed closed manual isolation valve is provided outside containment and an automatic (check) valve is provided inside containment to ensure containment integrity is maintained.

9.3.7.1.1.8 Auxiliary Steam System Safety Function Requirement

(1) Protection Against High-Energy Pipe Rupture Effects

The dynamic effects from a rupture of the PG&E Design Class II portion of the auxiliary steam system will not impact PG&E Design Class I essential SSCs required for safe (cold) shutdown of the plant.

(2) Protection from Jet Impingement – Inside Containment

The PG&E Design Class I containment isolation portion of the auxiliary steam system located inside containment is designed to be protected against the effects of jet impingement which may result from high energy pipe rupture.

9.3.7.1.1.9 10 CFR 50.55a(f) - Inservice Testing Requirements

PG&E Design Class I auxiliary steam system ASME Code components are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

9.3.7.1.1.10 10 CFR 50.55a(g) - Inservice Inspection Requirements

PG&E Design Class I auxiliary steam system ASME Code components are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.3.7.1.1.11 Generic Letter 89-08, May 1989 - Erosion/Corrosion-Induced Pipe Wall Thinning

DCPP has implemented formalized procedures and administrative controls to assure long-term implementation of its erosion/corrosion monitoring program for the auxiliary steam system.

9.3.7.1.2 System Description

This system is required to supply steam to certain pieces of equipment and plant locations. Steam is required for the following:

(1) Cask decontamination area

(2) Caustic storage tank

(3) Boric acid batching tank and water preheater

(4) Gland steam supply for the main turbine and main feedwater pump drive turbines

(5) Steam jet air ejector (main condenser)

(6) Containment atmosphere

(7) Steam for service cleaning and equipment maintenance inside containment

(8) Building heating reboiler:

   (a) Containment purge air

   (b) Fuel handling area

   (c) Machine shop

(9) Main condenser deaeration steam

(10) Caustic regeneration system

(11) Carbon dioxide (CO_2) vaporizer

During shutdown, it will be necessary to supply steam to the boric acid batch tank. All pressure parts and accessories are designed, constructed, inspected, and stamped in accordance with ASME BPVC Section I–1980 through 1980 Summer Addenda.

9.3.7.1.3 Safety Evaluation

9.3.7.1.3.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the auxiliary steam system is contained within the auxiliary building and containment building. These structures are PG&E Design Class I (refer to Section 3.8). These buildings or applicable portions thereof are designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), earthquakes (refer to Section 3.7), and other natural phenomena. This design protects auxiliary steam system components, ensuring their design function will be performed.

All valves, piping, and equipment that are considered to be isolation barriers are designed to PG&E Design Class I requirements ensuring their design function will be performed.

9.3.7.1.3.2 General Design Criterion 3, 1971 - Fire Protection

The PG&E Design Class I portion of the auxiliary steam system is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.3.7.1.3.3 General Design Criterion 4, 1967 – Sharing of Systems

The auxiliary steam system is designed such that either unit can supply the other unit with auxiliary steam. The interface between the auxiliary steam system and the main steam system is downstream of the MSIVs in the PG&E Design Class II portion of the system. Check valves in the supply lines prevent reverse flow from auxiliary steam to the main steam system. DCPP Unit 1 and Unit 2 also share auxiliary boiler 0-2, which is classified as PG&E Design Class II. Therefore, safety is not impaired by sharing the auxiliary steam system.

9.3.7.1.3.4 General Design Criterion 40, 1967 – Missile Protection

The provisions taken to protect the PG&E Design Class I containment isolation portion of the auxiliary steam system from damage that might result from missiles and dynamic effects associated with equipment and high-energy pipe failures are discussed in Sections 3.5, 3.6, and 6.2.4.

9.3.7.1.3.5 General Design Criterion 49, 1967 - Containment Design Basis

The auxiliary steam containment penetrations are designed and analyzed to withstand the pressures and temperatures that could result from a LOCA without exceeding the design leakage rates (refer to Section 3.8.2.1.1.3).

9.3.7.1.3.6 General Design Criterion 54, 1971 – Piping Systems Penetrating Containment

Isolation valves that are required for containment closure are periodically tested for operability and leakage and are contained in the PG&E Design Class I portion of the auxiliary steam system. Testing of the components required for the CIS is discussed in Section 6.2.4. Test connections are provided in the piping to verify valve leakages are within prescribed limits.

9.3.7.1.3.7 General Design Criterion 56, 1971 - Primary Containment Isolation

The PG&E Design Class I containment penetrations for the auxiliary steam system are penetration group E (refer to Section 6.2.4). A description of the isolation valves and piping for each penetration is provided in Table 6.2-39. Group E piping complies with the requirements of GDC 56, 1971 (refer to Section 6.2.4).

9.3.7.1.3.8 Auxiliary Steam System Safety Function Requirement

(1) <u>Protection Against High-Energy Pipe Rupture Effects</u>

Open crack breaks are postulated in the auxiliary steam system outside containment as it contains piping at a temperature that exceeds 200 ºF and pressure less than or equal to 275 psig. High energy crack breaks in the auxiliary steam system have been evaluated for effects on essential equipment to assure that safe (cold) shutdown of the plant can be attained (refer to Sections 3.6.2.1 and 3.6.2.1.2).

The PBIS provides alarm and control for any HELB in the auxiliary steam piping (refer to Section 7.6.3.2).

(2) <u>Protection from Jet Impingement – Inside Containment</u>

The provisions taken to provide protection of the PG&E Design Class I containment isolation portion of the auxiliary steam system located inside containment from the effects of jet impingement which may result from high energy pipe rupture are discussed in Section 3.6.

9.3.7.1.3.9 10 CFR 50.55a(f) - Inservice Testing Requirements

The IST requirements for the PG&E Design Class I components of the auxiliary steam system are contained in the IST Program Plan.

9.3.7.1.3.10 10 CFR 50.55a(g) - Inservice Inspection Requirements

The ISI requirements for the PG&E Design Class I portion of the auxiliary steam system are contained in the DCPP ISI Program Plan and comply with ASME BPVC Section XI-2001 through 2003 Addenda.

9.3.7.1.3.11 Generic Letter 89-08, May 1989 - Erosion/Corrosion-Induced Pipe Wall Thinning

DCPP procedures identify the interdepartmental responsibilities and interfaces for the monitoring program. Portions of the auxiliary steam system are considered in the DCPP Flow Accelerated Corrosion program for erosion/corrosion monitoring.

9.3.7.1.4 Tests and Inspections

Tests and inspections of the auxiliary steam system are done in accordance with plant procedures.

9.3.7.1.5 Instrumentation Applications

Instrumentation is provided to monitor and control the operation of the auxiliary steam system PBIS (refer to Section 7.6.3.2).

9.3.7.2 Oily Water Separator and Turbine Building Sump System

The OWS, common to both Unit 1 and Unit 2, is designed to separate oil and floating material from drains originating from the Unit 1 and Unit 2 TBSs.

The OWS system monitors the effluents from the TBS through the OWS retention tank discharge. Upon detection of radioactive concentrations in excess of pre-determined alarm setpoint limits (refer to Section 11.2.3.13.2.1), this channel will provide alarm annunciation.

9.3.7.2.1 Design Bases

9.3.7.2.1.1 General Design Criterion 3, 1971 - Fire Protection

The OWS/TBS system is designed and located to minimize, consistent with other safety requirements, the probability and effects of fires and explosions.

9.3.7.2.1.2 General Design Criterion 4, 1967 - Sharing of Systems

The OWS system is not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.3.7.2.1.3 General Design Criterion 11, 1967 - Control Room

The OWS system is designed to support actions to maintain and control the safe operational status of the plant from the control room.

9.3.7.2.1.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls are provided, as required, to monitor and maintain the OWS variables within prescribed operating ranges.

9.3.7.2.1.5 General Design Criterion 17, 1967 - Monitoring Radioactivity Releases

The OWS/TBS system is designed to provide means for monitoring the effluent discharge paths and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

9.3.7.2.2 System Description

The OWS, common to both Unit 1 and Unit 2, is designed to separate oil and floating material from drains originating from the Unit 1 and Unit 2 TBS. The clear water effluent normally is discharged to the condenser circulating water discharge tunnel. If required, it can be routed to the auxiliary building floor drain receivers. The radioactive content of liquids discharged from the turbine building is monitored by a radiation monitor and flow element in the process lines to the OWS (refer to Section 11.2.3.13.2.1).

9.3.7.2.3 Safety Evaluation

9.3.7.2.3.1 General Design Criterion 3, 1971 - Fire Protection

The OWS/TBS system is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.3.7.2.3.2 General Design Criterion 4, 1967 - Sharing of Systems

The OWS is shared by both Unit 1 and Unit 2 and is classified as PG&E Design Class II. The interface between the OWS and each unit is the PG&E Design Class II TBSs. Therefore, safety is not impaired by the sharing because neither the OWS nor the TBSs are essential to safe shutdown and isolation of the reactor; and failure of these structures and components would not result in the release of substantial amounts of radioactivity.

9.3.7.2.3.3 General Design Criterion 11, 1967 - Control Room

Instrumentation and alarms are provided in the control room for the OWS radiation monitor. Should radioactivity be detected above alarm setpoints, the discharge would be manually directed to the LRS (refer to Section 11.2.3.13.2.1).

9.3.7.2.3.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls for the OWS radiation monitor are provided to monitor and maintain applicable variables within prescribed operating ranges.

9.3.7.2.3.5 General Design Criterion 17, 1967 - Monitoring Radioactivity Releases

The OWS is provided with means for monitoring the release of radioactivity in facility effluent discharge paths (refer to Section 11.2.3.13.2.1).

9.3.7.2.4 Tests and Inspections

Tests and inspections of the OWS/TBS are done in accordance with plant procedures.

9.3.7.2.5 Instrumentation Applications

Instrumentation is provided to monitor and annunciate in the control room (refer to Section 9.3.7.2.3.3).

9.3.8 REFERENCES

1. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended.

2. NUREG-0737, Clarification of TMI Action Plan Requirements, U. S. Nuclear Regulatory Commission, November 1980.

9.4 HEATING, VENTILATION, AND AIR-CONDITIONING SYSTEMS

The design outdoor ambient air temperature for the HVAC systems described in this section is 76°F.

DCPP design outdoor ambient temperature is based on Recommended Outdoor Design Temperatures, Southern California, Arizona, and Nevada, Third Edition, Southern California Chapter, American Society of Heating, Refrigeration and Air-Conditioning Engineers, Inc., March 1964 (Reference 5). The most conservative temperature probability considered in that report is 1.0%, which are the temperature values that will be equaled or exceeded for 88 hours over a full year (8760 hours). PG&E has reviewed 40 years of onsite hourly ambient outdoor temperature data for the period May 1973 through April 2014 and determined the 1.0% probability of temperature occurrence is 70.5°F. Rounded up to 71°F and adding a 5°F margin to accommodate missing data and potential data uncertainties results in a design value of 76°F.

With a design value of 76°F and an allowable 26°F rise, the ambient indoor air temperatures will be maintained at or below 104°F at all times, except for areas as noted in subsequent sections. Outdoor ambient temperatures could exceed the design temperature for 88 hours per year; however, the 88 hours would not be continuous and would occur for a small portion of any given day. In fact, the above room temperature conditions are unlikely to be reached every year. The Class 1E electrical equipment is capable of operating for short periods at temperatures in excess of 117°F. This will have an insignificant effect on the aging of the electrical insulation.

In calculating the performance of ventilating and air conditioning systems, no credit was taken for wind cooling of buildings and structures and heat absorption by equipment at elevated temperatures. The maximum solar load was checked for selected areas that housed safety-related equipment and was found to have negligible effect on the calculated indoor temperatures in these areas.

The design classifications for the various HVAC systems are given in Table 3.2-3.

9.4.1 CONTROL ROOM

The control room ventilation system (CRVS) functions during all design accident conditions. The system permits continuous occupancy of the control room under normal and design basis accident conditions.

9.4.1.1 Design Bases

9.4.1.1.1 General Design Criterion 2, 1967 - Performance Standards

The CRVS is designed to withstand the effects of or is protected against natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.4.1.1.2 General Design Criterion 3, 1971 - Fire Protection

The CRVS is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.4.1.1.3 General Design Criterion 4, 1967 - Sharing of Systems

The CRVS components are not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.4.1.1.4 General Design Criterion 11, 1967 - Control Room

The control room is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.4.1.1.5 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the CRVS variables within prescribed operating ranges.

9.4.1.1.6 General Design Criterion 17, 1967 - Monitoring Radioactivity Releases

The control room is designed to provide means for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

9.4.1.1.7 General Design Criterion 19, 1999 - Control Room

The control room is designed to permit access and occupancy under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of a design basis accident.

9.4.1.1.8 General Design Criterion 21, 1967 Single Failure

The PG&E Design Class I CRVS is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event shall be treated as a single failure.

9.4.1.1.9 General Design Criterion 37, 1967 - Engineered Safety Features Basis for Design

The CRVS is designed to provide backup to the safety provided by the core design, the RCPB, and their protection systems.

9.4.1.1.10 General Design Criterion 38, 1967 - Reliability and Testability of Engineered Safety Features

The CRVS is designed to provide high functional reliability and ready testability.

9.4.1.1.11 General Design Criterion 40, 1967 - Missile Protection

The CRVS is designed to be protected against dynamic effects and missiles that might result from plant equipment failures.

9.4.1.1.12 General Design Criterion 41, 1967 - Engineered Safety Features Performance Capability

The CRVS is designed to provide sufficient performance capability to accommodate a partial loss of installed capacity, such as a failure of a single active component, and still perform its required safety function.

9.4.1.1.13 Control Room Ventilation System Safety Function Requirement

(1) Cooling of PG&E Design Class I Equipment

The CRVS is designed to provide cooling to PG&E Design Class I equipment.

(2) Protection from Moderate Energy Pipe Rupture Effects

The CRVS is protected against the effects of moderate energy pipe failure events or alternate means for accomplishing and maintaining safe (cold) shutdown condition of the plant is afforded.

(3) Protection from Flooding Effects

The CRVS is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe (cold) shutdown condition of the reactor can be accomplished and maintained.

9.4.1.1.14 10 CFR 50.63 - Loss of All Alternating Current Power

The CRVS is designed to continue to support control room habitability to support systems that assure core cooling and containment integrity is maintained following a SBO event.

9.4.1.1.15 Regulatory Guide 1.52, June 1973 - Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The CRVS design and maintenance comply with applicable Regulatory Guide 1.52, June 1973 requirements with exceptions noted in Table 9.4-2.

9.4.1.1.16 Regulatory Guide 1.52, Revision 2, March 1978 - Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The CRVS filter testing program complies with applicable Regulatory Guide 1.52, Revision 2, Regulatory Position C.5.a, C.5.c, C.5.d and C.6.a requirements as described in Table 9.4-2. The CRVS filter testing program also meets the requirements of Generic Letter 83-13, March 1983.

9.4.1.1.17 NUREG-0737 (Item II.B.2), November 1980 - Clarification of TMI Action Plan Requirements

Item II.B.2 – Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-accident Operations: Adequate access to the control room is provided by design changes, increased permanent or temporary shielding, or post-accident procedural controls.

9.4.1.2 System Description

The CRVS is composed of three systems:

(1) Control room heating, ventilation, and air conditioning (CRHVAC) system

(2) Control room pressurization system (CRPS)

(3) Plant process computer room air conditioning system

The CRVS has four operating modes (Modes 1-4, described below). There are two CRVS trains, both of which are operating during normal operations. These serve the common control room, one associated with the Unit 1 area and one associated with the Unit 2 area. Each train utilizes a single filter bank for removal of particulates and toxic or radioactive gases (Modes 3 or 4) and two redundant subtrains of air handling components. Each subtrain consists of a CRPS supply fan and electrical humidity control heater (Mode 4); CRHVAC supply fan (all modes) and filter booster fan (Modes 3 and 4); and an air conditioning unit (ACU) (all modes). Ducting, valves or dampers, and monitoring instrumentation are included in the system (refer to Table 9.4-1). The arrangement is shown in Figure 9.4-1. The isolation dampers are bubble-tight-type and are PG&E Design Class I. The ductwork, ventilation fans, and filter units are PG&E Design Class I. The associated design classifications are given in Table 3.2-3. The design data for the system components are given in the itemized list in Table 9.4-1. Design codes and standards are given in Table 9.4-8.

The Unit 1 area served by the system includes one computer room, one instrument safeguards room, one records storage room, one office, one kitchen area, one control room area, and one mechanical equipment area. The Unit 2 area served by the system includes one computer room, one instrument safeguards room, one toilet room, one control room area, one office, and one mechanical equipment area. CRPS intakes are located at the southwest and northwest ends of the turbine building.

A small PG&E Design Class II exhaust fan (also itemized in Table 9.4-1) and a PG&E Design Class I electric duct heater (the duct heater housing is PG&E Design Class I, the coil may be PG&E Design Class II) per unit are provided. The exhaust fan provides control room exhaust during normal operations (Mode 1) and for smoke removal (Mode 2).

The system is designed to provide a normal ambient temperature of 75°F with one set of redundant air conditioning equipment of each train operating. Inside air temperature will be less than 85°F except in the computer room, which is designed to have a temperature of 72 ±4°F. The cooling capacity of each HVAC train is 31 tons based on normal operating conditions. The estimated cooling load for each unit area, including a portion of the computer room load, is 19.0 tons under normal operating conditions. The air flow into each unit area is 7475 cfm which provides nine control room and fifteen computer room air changes per hour. The system also provides 325 cfm to the HVAC equipment room of each unit to maintain the ambient temperature below 104°F. Electric duct heaters reheat or temper the air as necessary to maximize personnel comfort control. The heaters are not redundant and are provided only to help balance the cooling system. In the event of heater malfunction, no vital control room air normalization functions are adversely affected.

Additionally, the computer room in each unit is provided with a supplemental PG&E Design Class II air conditioning system. The system consists of three ACUs, air cooled condensing units, and interconnecting refrigeration piping. The ACUs are staged by associated room thermostats. These units provide a suitable environment for the PG&E Design Class II computers. All redundant equipment receives power from Class 1E buses separated to meet single failure criteria.

The four modes of operation for the CRVS are:

Mode 1: Conditioned air is supplied and returned through ducts to the designated service area of each unit. Approximately 27 percent of the return air is normally exhausted to the atmosphere and 73 percent of the return air is normally recirculated. The recirculated air is mixed with outdoor makeup air and filtered through roughing filters, cooled (or heated), and supplied to the control room.

Estimated control room area heat loads for this mode of operation are listed in Table 9.4-9.

Mode 2: In the event of a fire in the control room, provisions are made for once through, 100 percent outdoor air operation. This mode exhausts the smoke from the room, thereby making it habitable. Roughing filters are used for filtering the outdoor air. The mode is manually initiated.

Mode 3: In the event of toxic gas outside the control room, provisions are made for manual zone isolation, 100 percent recirculated air with 27 percent passing through the HEPA filters and charcoal banks. Human detection (odor/smell) is used to identify the potential hazard and the mode is manually initiated.

Mode 4: This mode of operation is used in the event of airborne radioactivity and the requirement of long-term occupancy of the control room. Mode 4 isolates and pressurizes the control room and mechanical equipment room through the HEPA and charcoal filters with air from a low activity region to reduce local infiltration. The opposite control room HVAC train operates concurrently in Mode 3 recirculation. The flow rate of return air recirculated through the HEPA and charcoal filters is controlled by technical specifications (Reference 10). In the event an accident occurs in one unit, the system automatically selects the pressurization intake train of the opposite unit. With radiation detected at both pressurization intakes, one of the trains will start. However, the operator manually switches to the intake with lower airborne radioactivity.

For each unit, two manual mode selector switches (one for each subtrain) are used to place the CRVS in the desired operating mode and two manual bus selector switches (one for each subtrain) are used to place the desired subtrain in service. In addition, manual transfer switches are provided for the control, logic and power circuits that enable operation of the CRVS from either Unit 1 or Unit 2 power in the event that one unit is not operable.

The influx of airborne contaminants through the normal supply duct is limited by monitoring devices, which, upon detection of radioactive contaminants, automatically initiate CRVS Mode 4 operation. Closure of isolation dampers for CRVS Modes 3 and 4 (refer to Figure 9.4-1) is required within 10 seconds after manual initiation. A common warning light is provided on the control room panel to alert the operator if a ventilation damper is out of position. The position of each damper is indicated on a panel in the ventilation equipment room. Flow characteristics of the dampers are such that the average flow over the closure time is less than 60 percent of full flow.

Both the Unit 1 and Unit 2 CRVSs are designed to initiate Mode 4 operation, and Mode 3 operation on the opposite unit, automatically upon any one of the following signals:

(1) Containment phase A isolation from SI Unit 1

(2) Containment phase A isolation from SI Unit 2

(3) High outside air activity from Unit 1 normal air intake radiation monitor

(4) High outside air activity from Unit 2 normal air intake radiation monitor

The control room area is provided with minimum leakage dampers, weather-stripped doors, door vestibules, and absence of outdoor windows. Administrative controls ensure that all control room entranceways are normally closed.

Further protective options are provided by self-contained breathing apparatus located in the control room as stated in Section 6.4.1.

9.4.1.3 Safety Evaluation

9.4.1.3.1 General Design Criterion 2, 1967 - Performance Standards

The CRVS components are contained in the auxiliary and turbine buildings. These structures are PG&E Design Class I and PG&E Design Class II respectively (refer to Section 3.8). These buildings or applicable portions thereof are designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunami (refer to Section 3.4), external missiles (refer to Section 3.5), earthquakes (refer to Section 3.7), and to protect control room habitability SSCs from damage that may result from these events; ensuring their design function will be performed.

The portions of the CRVS that have the potential of being damaged by a tornado missile are the normal intake and exhaust louvers and the CRPS fans, ductwork, dampers and associated controls located on the turbine building operating deck and the auxiliary building roof. The ability of the normal intake and exhaust louvers to sustain tornado wind and missile damage is detailed in Section 3.3.2.5.2.1.

The CRVS, a PG&E Design Class I system, is designed to perform its safety functions under the effects of earthquakes (refer to Section 3.10.3.20).

9.4.1.3.2 General Design Criterion 3, 1971 - Fire Protection

The CRVS is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.4.1.3.3 General Design Criterion 4, 1967 - Sharing of Systems

The control room is common to DCPP Unit 1 and Unit 2, and therefore requires sharing of SSCs between units. Mechanically interlocked transfer switches allow for the transfer of power between Unit 1 and Unit 2 480-V, Class 1E switchgear buses. Refer to Section 8.3.1.1.4.3.3 for the sharing of electrical power and equipment. Safety of the CRVS is not impaired by the sharing because the two-train CRVS meets redundancy and single failure criteria for the common control room.

In addition, the CRPS is shared between the control room and the TSC. This sharing does not impair safety functions because each of the two pressurization trains independently have the capability to provide makeup air to the entire common control room area as well as to the TSC.

9.4.1.3.4 General Design Criterion 11, 1967 - Control Room

Under normal conditions (Mode 1), the Unit 1 and Unit 2 CRHVAC system supplies conditioned air consisting of recirculated air mixed with outdoor makeup air. In the event of a fire in the control room envelope, a smoke detector located in the return air duct is capable of detecting combustion products before visible smoke is present. The presence of combustion products would be alarmed in the control room where the operator would take corrective action. In the event of a large amount of smoke, the operator has the option to switch the CRVS into the 100 percent outdoor air mode of operation (Mode 2). This mode would purge the room of smoke. The control room is equipped with an adequate number of proper types of fire extinguishers and is manned by an operator(s) trained in fire-fighting procedures.

In the event that either damper 7 or 8 (refer to Figure 9.4-1) failed to open during Mode 2 ventilation, the rate of air removal from the control room would be reduced. If damper 7 failed to open, the air removal rate would be approximately 2100 cfm. If damper 8 failed to open, the air removal rate would be approximately 5375 cfm. Reduced air removal rates would result in slower removal of smoke.

In the event of toxic gas outside the control room, 100 percent of the air would be recirculated air with a portion passing through the HEPA filters and charcoal banks (Mode 3). Mode 3 operation of the CRVS may be manually initiated on human detection of toxic gas by the control room operators. Intake closure occurs within 10 seconds after manual actuation. Infiltration of the toxic gas from outdoors and other areas of the auxiliary building would be limited by minimum leakage dampers, zero leakage penetrations, weather-stripped doors, vestibules, and the absence of outside windows. Administrative controls ensure that all control room entranceways are normally closed. The plant complies with Regulatory Guide 1.78 (Refer to Sections 2.2.3.3 and 6.4.1.3.13).

With complete recirculation of the ventilation air, the CO_2 buildup is not expected to exceed an acceptable concentration of 1 percent by volume during 800 manhours of occupancy in the control room complex (e.g., 20 persons for 40 hours). Mode 4 operation maintains acceptable CO_2 levels inside the control room area due to the influx of filtered, outside air and exfiltration.

Refer to Section 9.4.1.2 for a general discussion of CRVS modes of operation. Mode 4 is governed by GDC 19, 1999 and is evaluated in Section 9.4.1.3.7.

9.4.1.3.5 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Monitoring devices for control room HVAC systems are capable of detecting low levels of airborne contaminants:

(1) Smoke detectors have sensitivity to detect trace amounts of combustion products.

(2) The two outside air activity area monitors per intake that monitor air entering the pressurization system duct are Geiger Muller tube-type general purpose monitors with a 10^{-2} to 10^4 mR/hr range. Each monitor has a control room readout module with instrument failure alarm. A high radiation signal provides control room annunciation, shuts down the operating CRPS train, and automatically starts the opposite unit's CRPS.

(3) One area monitor is mounted on the radiation monitoring racks in the control room. This monitor is a Geiger Muller tube-type general-purpose monitor with a 10^{-1} to 10^4 mR/hr range. This monitor has a control room readout with instrument failure alarm, local alarm, and control room annunciation.

(4) The chlorine monitors at the pressurization outside air duct are abandoned in place as there is no bulk chlorine on site.

High smoke or airborne radioactivity is annunciated in the control room so that the operator can take appropriate action.

9.4.1.3.6 General Design Criterion 17, 1967 - Monitoring Radioactivity Releases

Monitoring devices for control room HVAC systems are capable of detecting low levels of airborne radioactivity. Two outside air activity monitors per intake monitor the air entering the control room supply duct. Two outside air activity area monitors per intake monitor the air entering the pressurization system duct (refer to Section 9.4.1.3.5). One area monitor is mounted on the radiation monitoring racks in the control room (refer to Section 9.4.1.3.5).

9.4.1.3.7 General Design Criterion 19, 1999 - Control Room

CRVS Mode 4 operation is automatically initiated on a containment phase A isolation (SI) or normal air intake radiation monitor signal. Initiation of Mode 4 operation on a CRVS train initiates Mode 3 operation on the opposite train. Intake closure is designed to occur within 10 seconds or less after initiation of closure signal. Infiltration of activity from outdoors and other areas of the auxiliary building is limited by positive pressure, minimum leakage dampers, zero leakage penetrations, weather-stripped doors, door vestibules, and the absence of outside windows. Administrative controls ensure that all control room entranceways are normally closed.

The CRVS, in conjunction with shielding and administrative controls, is designed to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the most severe design basis accident. An evaluation of post-accident control room radiological exposures is presented in Section 15.5.

9.4.1.3.8 General Design Criterion 21, 1967 Single Failure

The PG&E Design Class I CRVS is designed to remain operable after sustaining a single failure (refer to Section 9.4.1.3.12).

9.4.1.3.9 General Design Criterion 37, 1967 - Engineered Safety Features Basis for Design

The CRVS is an ESF system designed to provide control room habitability in order to support systems that provide backup to the safety provided by the core design, the RCPB, and their protection systems. Refer to discussion of ESF design bases under Sections 9.4.1.3.4 and 9.4.1.3.7.

9.4.1.3.10 General Design Criterion 38, 1967 - Reliability and Testability of Engineered Safety Features

The CRVS has been designed with provisions for periodic tests, inspections, and surveillance to ensure that the systems will be dependable and effective when called upon to function. Refer to Section 9.4.1.4 for a discussion of inspection and testing requirements.

9.4.1.3.11 General Design Criterion 40, 1967 - Missile Protection

PG&E Design Class I structures and equipment outside the containment were reviewed to determine those that could be affected by potential missiles. For postulated missile sources, it was determined that release of a missile would not endanger the nuclear safety of the DCPP Unit 1 or Unit 2 CRVS (refer to Section 3.5.1.2).

DCPP is designed so that a postulated piping failure will not cause the loss of needed functions of PG&E Design Class I SSCs, so that the plant can be safely shut down in the event of such failure (refer to Section 3.6)..

9.4.1.3.12 General Design Criterion 41, 1967 - Engineered Safety Features Performance Capability

The CRHVAC system consists of two trains. Each train comprises two redundant sets of full capacity active components. The two pressurization fans for each of the air intakes associated with the CRPS and the charcoal adsorber and HEPA filter units are common to Unit 1 and Unit 2, and are fully redundant. Two independent and redundant control room HVAC trains; i.e., one train for each unit, are provided to ensure that at least one is available if a single active failure disables the other train. The source of the electrical power for each of the redundant active components is from a Class 1E bus, and active components are powered from separate Class 1E buses.

Pneumatic dampers have a designated position that they will assume upon loss of the control air supply. This position would be the same for Mode 3 or Mode 4 operation. Electric dampers that must assume a position in any mode of operation are supplied with power from Class 1E buses. Separation of the electrical supply from the Class 1E buses has been followed throughout the installation.

9.4.1.3.13 Control Room Ventilation System Safety Function Requirement

(1) <u>Cooling of PG&E Design Class I Equipment</u>

The design indoor ambient temperatures are:

(a) Control room and safeguard rooms	85°F
(b) HVAC equipment rooms	104°F

The upper limit of temperature environment for the control room instrumentation is 120°F. Below this point, degradation of the equipment will not be an important factor. The system is designed to meet single failure criteria, so that one train of cooling is available at all times. With only one train of cooling equipment operating, the calculated temperature in the control room area is approximately 89°F and in the instrument safeguard room of the unit without air conditioning is approximately 116°F. Local hot areas within the equipment cabinets will be identified under operating conditions and provisions made for ventilation.

(2) <u>Protection from Moderate Energy Pipe Rupture Effects</u>

The CRVS is not required to be protected from postulated moderate energy line break effects because the primary instrumentation and control functions required for shutdown are located on the hot shutdown panel (refer to Section 7.4), in addition to being available in the control room, and are provided for the purpose of achieving and maintaining a safe operational status in the event that an evacuation of the control room is required (refer to Section 3.6.2.2.3)

(3) Protection from Flooding Effects

The provisions taken to provide protection of the PG&E Design Class I essential systems, including CRVS from internal flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6. The CRVS however is not required to be protected from all postulated moderate energy line break effects because the primary instrumentation and control functions required for shutdown are located on the hot shutdown panel (refer to Section 7.4), in addition to being available in the control room, and are provided for the purpose of achieving and maintaining a safe operational status in the event that an evacuation of the control room is required (refer to Section 3.6.2.2.3)

9.4.1.3.14 10 CFR 50.63 - Loss of All Alternating Current Power

Following an SBO, 100 percent cooling of the control room is provided for the 4 hour coping period, as follows: one subtrain of the CRVS is powered from the operating EDG (AAC source) in the blacked-out unit and one subtrain of the CRVS is powered from an operating EDG in the non-blacked-out unit.

Refer to Section 8.3.1.6 for further discussion of station blackout.

9.4.1.3.15 Regulatory Guide 1.52, June 1973 - Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The CRVS provides the capability to control airborne radioactive material that could enter the control room atmosphere during a design basis event to acceptable levels. The CRVS is an ESF habitability system. These systems are analyzed to the positions in Regulatory Guide 1.52, June 1973 (Reference 11) as discussed in Table 9.4-2, with exceptions noted for specific items. Requirements for the performance of ventilation filter testing are stated in DCPP Technical Specifications.

Individual HEPA filters have been tested and are specified by the manufacturer to be able to remove 99.97 percent of particles 0.3 microns and larger, based on dioctyl phathalate particles in a standard test procedure. The overall efficiency of the filter bank is dependent on the initial efficiency of the individual filters, the care with which the filters have been stored and installed, and the sealing effectiveness of the filter frames with the supporting members of the bank. A penetration and leakage test was performed in place prior to putting the system in operation. Plant procedures for receiving, storing, and handling HEPA filters are employed to ensure factory performance of all HEPA filters.

Individual charcoal filters have been tested and are specified by the manufacturer to be able to remove 99 percent of radioactive iodine in the form of elemental iodine and 95 percent of radioactive iodine in the form of methyl iodide. A bypass leakage test was performed in place prior to putting the system in operation.

9.4.1.3.16 Regulatory Guide 1.52, Revision 2, March 1978 - Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The DCPP ventilation filter testing program (VFTP) provides the requirements for the testing of charcoal samples from the CRVS charcoal adsorbers in accordance with ASTM D 3803-1989, which complies with Regulatory Guide 1.52, Revision 2 and Generic Letter 83-13, March 1983 for laboratory testing of charcoal samples and in-place penetration and bypass leakage testing in accordance with ANSI N510-1980. The VFTP complies with Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.a requirements as described in Table 9.4-2, with exceptions noted.

9.4.1.3.17 NUREG-0737 (Item II.B.2), November 1980 - Clarification of TMI Action Plan Requirements

Item II.B.2 **-** Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-accident Operations: Plant shielding has been evaluated with regard to both radiation doses at equipment locations and vital area access/occupancy requirements for post-accident plant operations. Specifically for the CRVS, dose analyses were performed to determine gamma doses for an operator during access and occupancy of the HVAC mechanical room. Occupancy for reading the differential pressure gauge is limited by the exposure from radioactivity buildup on the ESF HVAC filter. Total dose for this activity (the only anticipated activity) is acceptable in consideration of the overall GDC 19, 1999, dose limits for duration of the accident.

9.4.1.4 Tests and Inspections

9.4.1.4.1 Initial System Inspection and Tests

Initial checks of the motors, dampers, compressors, controls, monitors, etc., are made at the time of installation. A system air balance test and adjustment to design conditions are conducted. The final tests performed prior to actual operation of the CRVS are the in-place tests of the HEPA and charcoal filter banks. Refer to Table 9.4-2 for additional testing requirements.

9.4.1.4.2 Routine System Tests

The CRVS is functionally tested periodically as required by the DCPP Technical Specifications (Reference 10) for proper operation of the modes of operation. This testing includes checking the function of the dampers in Modes 1, 3, and 4 and measuring air flow through the filter.

Control room radiation monitors are periodically tested and calibrated as required by the DCPP Technical Specifications and Equipment Control Guidelines.

The control room ventilation smoke detectors are tested and demonstrated "operable" as required by the Equipment Control Guidelines.

An in-place test of the HEPA and charcoal filter banks is performed periodically as required by the DCPP Technical Specifications. Refer to Table 9.4-2 for additional testing requirements.

9.4.1.5 Instrumentation Applications

Required controls for CRVS components for system operation and instrumentation for monitoring CRVS parameters during normal operations and accident conditions are provided (refer to Sections 9.4.1.2 and 9.4.1.3.5).

9.4.2 AUXILIARY BUILDING

The following rooms/areas of the auxiliary building are provided with separate ventilation systems and are described in the referenced sections below:

(1) Control room – refer to Section 9.4.1

(2) 125-Vdc and 480-V switchgear rooms – refer to Section 9.4.9

(3) Battery charger and inverter rooms – refer to Section 9.4.9

(4) HSP area – refer to Section 9.4.9

(5) Cable spreading rooms (CSRs) – refer to Section 9.4.9

The PG&E Design Class I ABVS described in this section includes the following rooms/areas:

(1) ESFs and PG&E Design Class I electrical equipment areas: CCW pump area, SI pumps room, containment spray pumps area, RHR pump room, boric acid transfer pump area, and CCP1 and CCP2 rooms

(2) Radwaste areas

The ABVS provides the following functions:

(1) The ESF function to reduce the amount of volatile radioactive materials that could be released to the atmosphere in the event of leakage from the RHR circulation loop following a LOCA (refer to Sections 6.1.2, 6.3.3.5.3.2, and 15.5)

(4) The PG&E Design Class I function of maintaining the temperature of the ESF pump motors within acceptable limits during their operation

(5) Ventilation to the auxiliary building

(6) A filtered flowpath which serves as a portion of one train of the containment hydrogen purge system and the containment excess pressure relief system

9.4.2.1 Design Bases

9.4.2.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the ABVS is designed to withstand the effects of, or is protected against natural phenomena such as earthquakes, tornadoes, flooding, winds, tsunamis, and other local site effects.

9.4.2.1.2 General Design Criterion 3, 1971 – Fire Protection

The ABVS is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.4.2.1.3 General Design Criterion 4, 1967 – Sharing of Systems

The ABVS is not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.4.2.1.4 General Design Criterion 11, 1967 – Control Room

The PG&E Design Class I portion of the ABVS is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.4.2.1.5 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the ABVS variables within prescribed operating ranges.

9.4.2.1.6 General Design Criterion 17, 1967 – Monitoring Radioactivity Releases

The ABVS is designed to provide means for monitoring the facility effluent discharge paths and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

9.4.2.1.7 General Design Criterion 21, 1967 – Single Failure Definition

The PG&E Design Class I portion of the ABVS is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event are treated as a single failure.

9.4.2.1.8 General Design Criterion 37, 1967 – Engineered Safety Features Basis for Design

The ESF portion of the ABVS is designed to provide backup to the safety provided by the core design, the RCPB, and their protection systems.

9.4.2.1.9 General Design Criterion 38, 1967 – Reliability and Testability of Engineered Safety Features

The ESF portion of the ABVS is designed to provide high functional reliability and ready testability.

9.4.2.1.10 General Design Criterion 40, 1967 – Missile Protection

The ESF portion of the ABVS is designed to be protected against dynamic effects and missiles that might result from plant equipment failures.

9.4.2.1.11 General Design Criterion 41, 1967 – Engineered Safety Features Performance Capability

The ESF portion of the ABVS is designed to provide sufficient performance capability to accommodate a partial loss of installed capacity, including a single failure of an active component, and still perform its required safety function.

9.4.2.1.12 General Design Criterion 42, 1967 – Engineered Safety Features Components Capability

The ESF portion of the ABVS is designed so that capability of each component and system to perform its required function is not impaired by the effects of a LOCA.

9.4.2.1.13 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The ABVS is designed with provisions for maintaining control over the plant's radioactive gaseous effluents. Appropriate holdup capacity is provided for retention of gaseous effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment.

9.4.2.1.14 Auxiliary Building Ventilation System Safety Function Requirements

(1) Protection from Missiles

The PG&E Design Class I non-ESF portion of the ABVS is designed to be protected against internal missiles generated outside containment that might result from plant equipment failure and from events and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(2) Protection Against High Energy Pipe Rupture Effects

The PG&E Design Class I non-ESF ABVS is designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(3) Protection from Moderate Energy Pipe Rupture Effects

The PG&E Design Class I ABVS is designed to be protected against the effects of moderate energy pipe failure to the extent necessary in order to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(4) Protection from Flooding Effects

The PG&E Design Class I ABVS is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(5) Cooling of PG&E Design Class I Equipment

The ABVS is designed to provide heat removal to ESF equipment to maintain the environment within design conditions.

9.4.2.1.15 10 CFR 50.49 – Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants

The ABVS components that require EQ are qualified to the requirements of 10 CFR 50.49.

9.4.2.1.16 Regulatory Guide 1.52, June 1973 – Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The ABVS design and maintenance conforms with the guidance provided in Regulatory Guide 1.52, June 1973 with exceptions noted in Table 9.4-2.

9.4.2.1.17 Regulatory Guide 1.52, Revision 2, March 1978 – Design, Testing, and Maintenance Criteria for Post-Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The VFTP conforms with the guidance provided in Regulatory Guide 1.52, Revision 2, Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.a as described in Table 9.4-2. The VFTP also conforms with the guidance provided in Generic Letter 83-13, March 1983.

9.4.2.1.18 Regulatory Guide 1.97, Revision 3, May 1983 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

The ABVS provides instrumentation to monitor system variables during and following an accident.

9.4.2.1.19 NUREG-0737 (Item II.F.1), November 1980 – Clarification of TMI Action Plan Requirements

Item II.F.1 – Additional Accident Monitoring Instrumentation:

Position (1) – The ABVS is designed with noble gas effluent monitors that are installed with an extended range designed to function during accident conditions.

Position (2) – The ABVS is designed with provisions for sampling of plant effluents for post-accident releases of radioactive iodines and particulates and onsite laboratory capabilities.

9.4.2.2 System Description

The auxiliary building ventilating system provides all outside air supply for the auxiliary building under all operating conditions. It also serves its primary purpose of providing cooling air for the ESF pump motors.

The supply system consists of two full capacity fans, roughing filters, and duct work for distribution. The exhaust system consists of two full capacity fans, two full capacity combined roughing and HEPA filter banks, one full capacity combined electric heater, roughing, HEPA and charcoal (may also be referred to as carbon) filter bank, and exhaust duct network (refer to Figures 9.4-2 and 9.4-3). The major equipment for the system is listed in Table 9.4-5. The containment hydrogen purge line connects to the inlet plenum for the filter bank containing the charcoal filter (refer to Section 6.2.5 and Figures 6.2-20, 9.4-3, and 9.4-3A). Manually controlled valves in this line allow flow to leave the containment and exhaust to the plant vent through the charcoal filter. The containment excess pressure relief line and the alternate path for the containment hydrogen purge discharge into the plant vent through the ABVS charcoal filter.

Air Distribution

The air flow pattern is arranged so that the air flows from areas of lower potential contamination to areas of higher potential contamination, and is then discharged to the plant vent. This is accomplished by supplying air to the occupied areas then exhausting the air from each equipment area separately. The system is balanced so that the building is normally under a slight negative pressure. The exhaust ducts from potentially high activity areas are so routed as to minimize exposure to normally occupied areas.

All ductwork serving ESF equipment has been designed to withstand any internal pressure that may be generated by any fan in the system, and braced according to PG&E Design Class I criteria to prevent earthquake damage to the ducts (refer to Section 9.4.2.3.1). All static ventilation components (i.e., volume dampers and air diffusers) have locking devices to prevent accidental closing. No duct liner or other insulating substance, which might sag or fall down, thereby blocking the duct, has been installed in any ductwork.

Electrical Power

The fans for the system receive power from the Class 1E 480-V buses. The supply system consists of two full capacity fans (each powered from a separate Class 1E bus). The exhaust system consists of two full capacity fans (each powered from a separate Class 1E bus).

Modes of Operation

The ABVS has three modes of operation:

Building Only Mode: During normal plant operation the system is designed to be in the "Building Only" mode. In this mode, one of two 100 percent capacity supply fans and one of two 100 percent capacity exhaust fans operates. Outside air is drawn through intake louvers, roughing filters, and ducted to the building space through the building only ventilation mode supply duct system. In the building only ventilation operating mode, exhaust air passes through the roughing and HEPA filter train. In any mode of operation the exhaust air is discharged through the plant vent.

Building and Safeguards Mode: The system automatically shifts to Building and Safeguards Mode when the control logic receives:

(1) A SI signal

(2) An indication that any one of the motors for CCP1 or CCP2, RHR pumps, SI pumps, or containment spray pumps has started

Building and Safeguards Mode may also be manually selected from the control panel in the control room for test or for operation. In this mode, the redundant set of supply and exhaust fans are designed to operate so that two supply and two exhaust fans are operating simultaneously. As in Building Only Mode, outside air is drawn through intake louvers, roughing filters and ducted to the building via the building only ventilation mode supply duct system and also through the ESF supply duct system. Depending on the presence of a SI signal, the exhaust air is handled as follows:

(1) If the system receives a SI signal, the exhaust air from the ESF pump rooms and RHR heat exchanger rooms passes through the manually actuated electric heater, roughing, HEPA, and charcoal filters, and the exhaust air from all other areas passes through the redundant roughing and HEPA filter trains.

(2) In the absence of a SI signal, the exhaust air from all areas passes through the roughing and HEPA filter trains.

Safeguards Only Mode: This mode is automatically actuated if a supply or an exhaust fan fails while the system is in Building and Safeguards Mode. The system supplies air only through the ESF supply duct system utilizing one exhaust fan and one supply fan. Depending on the presence of a SI signal, the exhaust is handled as follows:

(1) If the system receives a SI signal, the exhaust air from the ESF pump rooms and RHR heat exchanger rooms passes through the manually actuated electric heater, roughing, HEPA, and charcoal filters.

(2) In the absence of a SI signal, the exhaust air from all areas passes through the roughing and HEPA filters trains.

Design values for the ABVS are listed in Table 9.4-10.

The design classifications for the ABVS are given in Table 3.2-3. The design codes and standards are given in Table 9.4-8.

ESF Air Flow Relative Humidity

In the unlikely event of a LOCA accompanied by RHR loop leakage in the auxiliary building, flashing of leakage water would increase the relative humidity of the auxiliary building ESF exhaust air flow. To enhance the efficient performance of the charcoal filters, a manually actuated, 54 kW electric heater located upstream of the auxiliary building charcoal filters is sized to reduce the relative humidity of the ESF air flow. Operation of this heater is not required for this system to perform its safety function.

This heater is sized on the following assumptions:

(1) Outside air conditions of 45°F and 100 percent relative humidity. These conditions represent a reasonable bound on expected conditions that maximizes the relative humidity of ESF air flow entering the auxiliary building charcoal filters.

(2) Auxiliary building heat loads that result in a temperature rise of 16.5°F due to an increase in sensible heat of the 73,500 cfm ESF air flow. The only heat loads included are those associated with equipment that is required to operate following a LOCA and that would be expected to be operating during the period of postulated RHR loop leakage. These heat loads include motor losses and heat losses from piping and equipment associated with ECCS and ESF ventilation system operation. Two trains of ECCS pumps, one containment spray system train, and two CCW pumps are assumed to be operating.

(3) Adiabatic mixing of 738 lb/hr steam (3.1 percent flashing of 50 gpm leakage, refer to Section 15.5) with that portion of the ESF air flow which passes through the RHR pump rooms. Condensation is assumed that results in 100 percent relative humidity prior to mixing this portion of the ESF airflow with the remainder of the ESF air.

Using these assumptions, and assuming operation of the 54 kW electric heater, results in less than 70 percent relative humidity for ESF air flow entering the auxiliary building charcoal filters. However, the charcoal used in the filters is tested to 95 percent relative humidity and the electric heater is not required for filter operability (refer to Section 9.4.2.3.17 and Table 9.4-2).

Differential Pressures

Regarding the differential pressures, the total auxiliary building was considered as an open building. However, the exhaust system will remove approximately 9 percent more air than the supply system will provide. The extra air quantity must be made up by infiltration. Each equipment compartment is exhausted separately or through another compartment of higher potential contamination. This flow distribution, combined with the system air flow balance, will establish a slight negative pressure within the compartments. No provision other than this has been made to maintain or monitor this negative pressure.

Containment Hydrogen Purge and Pressure Relief

The ABVS provides a filtered flow path, to the plant vent, which serves as a portion of one train of the containment hydrogen purge and the containment excess pressure relief systems (refer to Sections 6.2.5 and 9.4.5, respectively).

Containment hydrogen purging is manually controlled and, when required, operates only intermittently. No single failure can accidentally initiate flow.

Containment hydrogen purge is required for plant safety but is not required until several weeks following a LOCA (refer to Section 6.2.5.2.3). By this time, most of the ESF pumps are no longer in operation and the heat load on the ventilation system is well below its maximum. In addition, purging would be conducted only about 2 hours a day, at times desired by the operator. This further guarantees the ability to avoid peak load periods on the ventilation system. In any case, the maximum expected purge flow of only 300 cfm would not have a significant effect on the ventilation system.

9.4.2.3 Safety Evaluation

9.4.2.3.1 General Design Criterion 2, 1967 – Performance Standards

The ABVS is designed to perform its safety functions under the effects of earthquakes (refer to Section 3.10.3.33), winds and tornadoes (refer to Section 3.3.2.5.2.1), and external missiles (refer to Section 3.5.1.2). The ABVS may be vulnerable to tornado effects but does not need to be operational while the plant is brought to a safe operational status.

Various ventilation systems exhaust to the plant vent; a PG&E Design Class I structure. The plant vent is designed to perform its safety functions under the effects of earthquakes (refer to Section 3.8.2.1.1), winds and tornadoes, and external missiles. The plant vent may also be vulnerable to tornado effects; however the plant can be safely shutdown without additional radiation exposure to the public even if the plant vent and its radiation monitors are rendered inoperable by a tornado. Refer to Section 3.3.2.5.2.5 for further discussion of wind, tornado, and external missile effects on the plant vent.

ABVS equipment is located within the auxiliary and FHBs which are PG&E Design Class I structures (refer to Section 3.8.2.3). The auxiliary building, including the relevant portions of the FHB, is designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), and earthquakes (refer to Section 3.7). This design protects the ABVS, ensuring its design functions can be performed during these events.

9.4.2.3.2 General Design Criterion 3, 1971 – Fire Protection

The ABVS is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.4.2.3.3 General Design Criterion 4, 1967 – Sharing of Systems

No ABVS equipment is shared between DCPP Unit 1 and Unit 2. However, common auxiliary building areas are ventilated by both the Unit 1 and Unit 2 ABVSs.

The Building and Safeguards Mode of operation provides sufficient cooling to PG&E Design Class I equipment in the auxiliary building at all times (refer to Section 9.4.2.2). In the event of failure of a supply or exhaust fan while the system is in Building and Safeguards Mode, Safeguards Only Mode is automatically actuated to continue to provide sufficient cooling to ESF equipment (refer to Section 9.4.2.2).

Each unit's ABVS is designed for single failure protection ensuring at least one train per unit is operational at all times providing full ventilation flow to all areas of the auxiliary building (refer to Section 9.4.2.3.7).

Therefore safety is not impaired by the shared ventilation.

9.4.2.3.4 General Design Criterion 11, 1967 – Control Room

The ABVS is normally controlled from the control room. Controls are provided on the main control board which allow for system startup and shutdown, fan selection, and for the selection of system operating configuration and modes (refer to Section 9.4.2.2).

Indication is provided on the main control board to indicate fan failure or damper out of position. Annunciators in the control room alert operators to abnormal conditions and system or equipment failures (refer to Section 9.4.2.3.5).

The ABVS can be operated remotely from the ventilation control logic and relay cabinet in the CSR.

9.4.2.3.5 General Design Criterion 12, 1967 – Instrumentation and Control Systems

The operation of the system is initiated from the ventilation control board in the control room. The logic control will position the dampers and start the system for normal operation. The supply fan and exhaust fan are interlocked in a manner as to ensure the building will not be subjected to an appreciable amount of positive pressure. High-temperature and fan-failure alarms and damper position are provided to the main annunciator from the supply fan room.

The logic control devices for both Unit 1 and Unit 2 control system are solid-state units providing a full selection of logic functions designed for binary system operation. The control system is based on a programmable logic controller (PLC). The basis for qualification of the Triconex PLC and associated software follows the guidance of Branch Technical Positions 7-14 and 7-18 (References 21 and 22). The logic control system has output to solenoids and dry contact relays for the control functions. Both the logic controls and relays are redundant and have power sources from Class 1E buses.

The ambient air temperatures in areas containing Class 1E electrical equipment will be monitored continuously where excessive temperatures could possibly occur.

Additional information about the temperature monitoring system is provided in Sections 3.11 and 9.4.2.3.14. Additional information about radiation monitoring is provided in Section 9.4.2.3.6.

There are three ABVS modes of operation: Building Only, Building and Safeguards, and Safeguards Only. These modes are described in Section 9.4.2.2.

9.4.2.3.6 General Design Criterion 17, 1967 – Monitoring Radioactivity Releases

The air flow from the auxiliary building is monitored for abnormal radiation levels of both particulate and gaseous nature at the plant vent monitors.

The plant vent is a large duct installed on the side of the containment structure discharging to the atmosphere at the 328.5 foot elevation. The discharges from the vent are monitored for radiation from both particulate and gaseous material by the plant vent air particulate and noble gas effluent monitors, RM-28 (or RM-28R) and RM-14 (or RM-14R), respectively. Additionally, the plant vent is equipped with iodine monitors RM-24 (or RM-24R) and an extended range noble gas effluent monitor (RM-87).

In addition, effluent from the RHR equipment compartments is monitored for abnormal radiation levels (RM-13). The sample for this monitor may be taken from either RHR compartment exhaust ducts so that an evaluation of any detected leakage may be made.

The radiological monitoring system is described in Section 11.4 and Table 11.4-1. Post-accident radiation monitoring is discussed in Section 9.4.2.3.19.

9.4.2.3.7 General Design Criterion 21, 1967 - Single Failure Definition

The ABVS has redundancy for the initiation circuitry and all non-static components, except for the electric heater (which is not required for filter operability). The fans are each full capacity, sized to handle the ESF pump motor cooling. Each ABVS train consists of two full capacity supply and exhaust fans, each powered from a separate Class 1E bus. In the event of the loss of an exhaust fan coincident with a SI signal, the second exhaust fan will start and the dampers will position themselves to route the air through the ESF ducts to the combined HEPA and charcoal filter banks. All dampers fail in the positions required for emergency conditions.

The ABVS is provided with a single, full capacity roughing, HEPA, and charcoal filter bank (including an electric heater) to limit the offsite exposures from a post-LOCA RHR pump seal failure. A single charcoal filter train is provided because the failure of the charcoal filter train in conjunction with a LOCA and RHR pump seal would constitute a second failure as discussed in Section 15.5.

The power for the fans and the initiating (logic) circuitry is taken from Class 1E buses with separation of redundant components. The dampers are positioned by pneumatic actuators supplied from the plant compressed air supply. The dampers are designed to assume the position required for emergency conditions on the failure of the air supply. If a damper's fail position is open, two dampers are mounted in parallel. Conversely, if the damper's fail position is closed, two dampers are mounted in series. The initiating (logic) circuitry is redundant, including relays and solenoids required to actuate the system. Each control train serves similar control functions, in addition to switching over circuits in the event of a failure to the other train. Separation has been maintained for the electrical circuits from each Class 1E bus.

The ventilation system is designed to meet single failure criteria by having the non-static elements installed in redundancy. All dampers that must fail in the open position are redundant. Failure in the open position maintains cooling air flow through the ESF pump motor compartments.

The radwaste area is primarily that area served by the system that is not included in the "safeguards only" mode of operation. Two conditions will interrupt the air flow to the radwaste area. The first condition involves a double failure, the loss of either an exhaust or a supply fan and a coincident SI signal or the start of an ESF pump motor. The second condition is the loss of control air supply to the building exhaust duct dampers 4A and 4B (refer to Figure 9.4-3). Either case is an abnormal condition that is indicated to the control room operator who then takes action to minimize the leakage of radioactive material to the compartments and/or to the ventilation system.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.

Experience at the Humboldt Bay reactor has shown that normal leakage or vents from the radwaste system are not a large contributor to airborne radioactivity.

9.4.2.3.8 General Design Criterion 37, 1967 – Engineered Safety Features Basis for Design

The PG&E Design Class I ABVS provides the ESF function to reduce the amount of volatile radioactive materials that could be released to the atmosphere in the event of leakage from the RHR circulation loop following a LOCA (refer to Sections 6.1.2, 6.3.3.5.3.2, and 15.5).

The ventilation system serves the auxiliary building (including the radwaste area, excluding the fuel handling area). The general flow pattern is from areas of lower potential contamination to areas of higher potential contamination. The building is under a slight negative pressure. This concept has been followed throughout. This flow pattern will be maintained under the first two modes of operation. However, when the system is operating under the "safeguards only" mode (refer to Section 9.4.2.2), the total air flow will be exhausted through the ESF motor compartments only.

The exhaust air from the auxiliary building is normally routed through roughing and HEPA filter banks for the purpose of removing particulates that may be in the ventilation air. When the system is in the "building and safeguards" or "safeguards only" modes of operation without SI signal, air that is exhausted from the ESF pump motor areas is routed through roughing and HEPA filter banks. In the presence of a SI signal in conjunction with the system in one of these two modes, the exhaust air from the ESF pump motor area passes through the combined roughing, HEPA, and charcoal filters. The specifications for the exhaust filters are given in Table 9.4-5. Refer to Section 15.5 for a discussion on dose consequences analyses.

9.4.2.3.9 General Design Criterion 38, 1967 – Reliability and Testability of Engineered Safety Features

The ESF portion of the ABVS is designed with provisions for periodic tests, inspections and surveillance to ensure that the systems will be dependable and effective when called upon to function.

The system is operationally tested in accordance with the Technical Specifications. These tests include checking the function of the dampers and controls under each mode of operation and determination of total air flow.

A functional test of the HEPA and/or charcoal filter banks is performed whenever a filter element is replaced. A charcoal sample is removed in accordance with the VFTP to determine the effectiveness of the charcoal bank (refer to Sections 9.4.2.3.16 and 9.4.2.3.17 and Table 9.4-2).

9.4.2.3.10 General Design Criterion 40, 1967 – Missile Protection

There are no credible missiles outside of containment resulting from plant equipment failures that would prevent the ESF portion of the ABVS from performing its design functions (refer to Section 3.5.1.2).

Dynamic effects as a result of plant equipment failure will not prevent the PG&E Design Class I, ESF portion of the ABVS from performing its design functions (refer to Section 3.6).

9.4.2.3.11 General Design Criterion 41, 1967 – Engineered Safety Features Performance Capability

Each full capacity fan of a redundant set receives power from a separate Class 1E bus. The dampers for the system are redundant, pneumatically operated, and designed to fail in the positions required for emergency conditions. Dampers are arranged in parallel if the fail position is open, and in series if the fail position is closed (refer to Section 9.4.2.3.7).

9.4.2.3.12 General Design Criterion 42, 1967 – Engineered Safety Features Components Capability

Following a LOCA, the ABVS can tolerate a 50 gpm RHR system leak during the recirculation phase in the auxiliary building (refer to Sections 6.1.2, 6.3.3.5.3.2, and 15.5).

9.4.2.3.13 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The normal building exhaust air is filtered through HEPA filters, and upon a SI signal, the emergency cooling exhausts air through both HEPA filters and charcoal filter banks.

The occurrence of complete failure of the auxiliary building ventilation flow would result in a slow rise in airborne activity in the building only if a significant amount of leakage existed from equipment carrying radioactive fluids prior to the failure of ventilation flow. If concentrations should rise to significant levels, personnel can be kept out of the area by administrative controls, or wear protective respiratory equipment during any required maintenance work. Specific limits on plant operation and requirements for shutdown under conditions of reduced or lost ventilation flow are contained in the Technical Specifications. The characteristics of the monitors and sampling procedures for auxiliary building areas are given in Section 11.4. There are no public safety implications of failure of the auxiliary building ventilation during normal plant operation, since reduced exhaust flow would only result in increased holdup and decay of airborne activity before release from the plant. With regard to simultaneous coincident major failures in both the ventilation system and a major radwaste component, an analysis of the most severe consequences of such events is contained in Sections 15.5.24 through

15.5.26. Assessments of long-term exposures from small equipment leakages are included in Sections 11.3.2.5 and 12.2.6.

Radioactive iodines and particulates, for the removal of which the ABVS is potentially effective, are generally stored at low pressure in the auxiliary building. Failure of a low-pressure storage vessel would be extremely unlikely and would not generate a pressure pulse sufficient to damage the ABVS.

A larger pressure pulse would result from the rupture of a gas decay tank (GDT) or the volume control tank (VCT) as discussed in Sections 11.3.3.9 and 11.2.3.12, respectively. While some momentary release of radioactivity to these areas could result from a pressure pulse, any damage to the ductwork is likely to be confined to either (a) the branch serving the gas decay tanks and waste gas compressors, or (b) the branch serving the VCT and the sample heat exchangers.

After the initial pulse of pressure and release of radioactive materials, any airborne activity in areas served by the ABVS will be drawn toward the high efficiency filters of the system. Any local damage to the duct should affect only the ventilation in the immediate area. The ABVS design will prevent contaminants from being delivered to other areas of the auxiliary building.

In case of the spread of radioactive materials to other auxiliary building areas, the operating restrictions, described above and within Section 9.4.2.3.7, can be instituted if necessary. As described in Sections 11.3.3.9 and 11.2.3.12, no credit is taken for radioactivity removal by the ABVS in calculating offsite doses resulting from postulated failure of the GDT or VCT.

9.4.2.3.14 Auxiliary Building Ventilation System Safety Function Requirements

(1) <u>Protection from Missiles</u>

There are no credible missiles outside of containment resulting from plant equipment failure that would prevent the non-ESF portions of the PG&E Design Class I ABVS from performing its design functions to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained (refer to Section 3.5.1.2).

(2) <u>Protection Against High Energy Pipe Rupture Effects</u>

The provisions taken to protect the non-ESF portions of the PG&E Design Class I ABVS from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained as discussed in Section 3.6.

(3) Protection from Moderate Energy Pipe Rupture Effects

The provisions taken to provide protection to the PG&E Design Class I essential systems located outside containment from moderate energy pipe rupture effects (water spray) are discussed in Section 3.6.

(4) Protection from Flooding Effects

The provisions taken to provide protection to the PG&E Design Class I essential systems from flooding that might result from the effects associated with a postulated rupture of high and moderate energy piping are discussed in Section 3.6.

(5) Cooling of PG&E Design Class I Equipment

The ventilation system has been designed to maintain a maximum design room temperature of 104°F in the auxiliary building with a design outdoor ambient air temperature of 76°F (refer to Section 9.4) with the following exceptions:

(1) The LHUT rooms shall have a maximum design room temperature of 130°F.

(2) The ambient temperature in the area of the boric acid transfer pumps may reach a maximum temperature of approximately 124°F after one hour following the postulated event of one train of the ABVS being inoperable post LOCA. The boric acid transfer pump may be required to operate within this period post LOCA. During any other operating conditions, the maximum design room temperature in this area shall be 104°F.

The airflow requirements have been based on the ESF motor cooling load. The allowable ambient air temperature for continuous operation of the ESF motors is 104°F.

Although not an ESF function, the ABVS supplies cooling air for the ESF pump motors. Motors are designed for continuous operation at an ambient air temperature of 104°F. The supply fans and their associated dampers need to be in operation under both normal and accident conditions to maintain the EQ of the ESF pump motors because the cooling function of the system is important for their long-term operation.

9.4.2.3.15 10 CFR 50.49 – Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants

The ABVS components required to function in harsh environments under accident conditions are qualified to the applicable environmental conditions to ensure that they will continue to perform their safety functions. Section 3.11 describes the DCPP EQ program and the requirements for the environmental design of the electrical and related mechanical equipment. The affected components are listed on the EQ Master List (refer to Section 3.11.2).

9.4.2.3.16 Regulatory Guide 1.52, June 1973 – Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The ABVS provides the capability to control the volatile radioactive material that could exit to the atmosphere. The system is analyzed to the positions in Regulatory Guide 1.52, June 1973 (Reference 11) as shown in Table 9.4-2 with exceptions noted for specific items. Requirements for the performance of ventilation filter testing are stated in the DCPP VFTP.

9.4.2.3.17 Regulatory Guide 1.52, Revision 2, March 1978 – Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The VFTP provides the requirements for the testing of charcoal samples from the ABVS charcoal adsorbers in accordance with ASTM D 3803-1989, which complies with Regulatory Guide 1.52, Revision 2, and Generic Letter 83-13, March 1983 for laboratory testing of charcoal samples and in-place penetration and bypass leakage testing in accordance with ANSI N510-1980. The VFTP complies with Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.a requirements as described in Table 9.4-2, with exceptions noted.

9.4.2.3.18 Regulatory Guide 1.97, Revision 3, May 1983 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

Damper position indication in the control room provides status of the Unit 1 and Unit 2 ABVS fan suction and discharge motorized dampers in response to an ESF actuation (refer to Table 7.5-6 for a summary of compliance to Regulatory Guide 1.97, Revision 3).

9.4.2.3.19 NUREG-0737 (Item II.F.1), November 1980 – Clarification of TMI Action Plan Requirements

Item II.F.1 – Additional Accident Monitoring Instrumentation:

Position (1) – Extended range noble gas effluent monitoring is installed in the plant vent and is designed to function during accident conditions (refer to Sections 7.5.2.3 and 11.4.2.1.2.1).

Position (2) – Installed capability is provided in the plant to obtain samples of the particulate and iodine radioactivity concentrations which may be present in the gaseous effluent being discharged to the environment from the plant under accident and post-accident conditions. The TSC laboratory is available for onsite testing of the containment air samples.

9.4.2.4 Tests and Inspections

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.

An initial checkout of the motors, dampers, fans, controls, etc., was made at the time of installation. A system air balance test and adjustment to design conditions was conducted. The final tests performed prior to actual operation of the ventilation system were the functional tests of the HEPA and charcoal filter banks. These tests determined the overall effectiveness of the filter banks.

Refer to Section 9.4.2.3.9 for additional information on testing requirements.

9.4.2.5 Instrumentation Applications

Refer to Sections 9.4.2.3.4 and 9.4.2.3.5 for a complete discussion of the instrumentation requirements and controls for the ABVS.

9.4.3 TURBINE BUILDING

The turbine building ventilation system provides for personnel comfort and is a PG&E Design Class II system (refer to Table 3.2-3).

Although the onsite TSC is adjacent to the turbine building, the TSC is provided with its own ventilation system (refer to Section 9.4.11). Separate PG&E Design Class I ventilation systems in the turbine building are provided for the diesel generator rooms (refer to Section 9.4.7), and the 4.16-kV switchgear and 4.16-kV CSRs, (refer to Section 9.4.8).

9.4.3.1 Design Bases

9.4.3.1.1 General Design Criterion 3, 1971 – Fire Protection

The turbine building ventilation system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.4.3.2 System Description

The turbine building ventilation system has been designed for the purpose of providing personnel comfort in the operation of the turbine-generator equipment. This system is generally not necessary for airborne radioactivity control because of the low potential for contamination in the steam cycle. The overall design objectives of the ventilation systems regarding control of airborne radioactive materials are discussed in Section 12.2.

Because significant airborne activity is not expected on a continuous basis in the turbine building, continuous monitoring equipment is not provided. If significant activity levels are observed in the air ejector offgas, however, periodic sampling will be initiated to determine turbine building airborne activity. If sampling indicates the need for continuous sampling, it will be provided.

The turbine building ventilation system is a PG&E Design Class II system, and its complete failure has no safety implication. Because of the many fans in the system, failure of any fan or group of fans will not cause excessive temperatures in building compartments. Furthermore, if ventilation flow were not available, doors can be opened to provide natural flow. In the event that airborne activity exists in the building at a time when ventilation flow is not available, some increase in concentrations could occur. Because of the low level of this activity, however, it is not expected that concentrations above 10 CFR 20.1-20.601 (pre-1994) maximum permissible concentration levels could exist with the building doors open. In any event, sampling or monitoring will be in effect under these conditions.

Regarding the requirements for the treatment of exhaust air from the turbine building, a detailed analysis of potential doses to the public from various sources of gaseous release is provided in Chapter 11. As a result of this analysis, it can be concluded that, at the DCPP site, treatment of turbine building exhaust air is not required to meet the exposure limits listed in Appendix I to 10 CFR Part 50.

Air is drawn into the system, through roughing filters, by the system's fifteen (15) supply fans. The supply fans direct air into the turbine building through the east wall both above and below the elevation 140 foot operating deck. An exhaust fan for the lube oil reservoir room draws air through the room from the 104 foot elevation of the turbine building, which is supplied by the turbine building ventilation system. The turbine building ventilation system provides supply air as well as a discharge path to atmosphere for the non-Class 1E battery rooms.

The four exhaust fans and their related duct systems, located on the west side of the building, ensure air flow across the building below the operating deck. The exhaust fans discharge the air vertically up above the operating deck inside the turbine building. The turbine building ventilation system provides a discharge path to atmosphere for the exhaust air from the Class 1E 4.16-kV switchgear rooms and associated CSRs (refer to Section 9.4.8), the non-Class 1E 480-V switchgear room, and the non-Class 1E battery rooms. A vent located on the top of the turbine building provides the exhaust path to atmosphere for turbine building supply air that is not discharged from the building via other paths.

Each fan can be operated independently through a manual motor starter with power from a non-Class 1E bus. Should there be single or double fan failure, the net effect on the turbine building would be negligible. Should all non-Class 1E power to fans be de-energized, all doors on the east and west walls (elevation 85 foot) could be opened, allowing the building to function on gravity ventilation.

The locations of all fans, vents, and compartments are shown in the following: Figures 1.2-13 through 1.2-20, 1.2-24 through 1.2-27, and 1.2-30 through 1.2-32. The major equipment for the turbine building (general area) ventilation system is listed in Table 9.4-7.

The requirements for system temperature design are described in Section 9.4.

9.4.3.3 Safety Evaluation

9.4.3.3.1 General Design Criterion 3, 1971 – Fire Protection

The turbine building ventilation system is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1). Specifically, fire dampers are provided for the lube oil reservoir room to maintain CO_2 gas concentration upon actuation of the CO_2 flooding gas extinguishing system provided for that room.

9.4.3.4 Tests and Inspections

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.

Initial checks of the fan housings, bearings, motors, bolts, controls, etc., were made at the time of installation. A system air balance test and adjustment to design conditions was conducted.

9.4.3.5 Instrumentation Applications

Each of the 15 supply fans and 4 exhaust fans can be operated independently through manual motor starters.

9.4.4 FUEL HANDLING BUILDING VENTILATION SYSTEM

The non-ESF portion of the FHBVS serves the FHB and rooms containing the motor-driven and turbine-driven AFW pumps, SFP pumps, SFP heat exchanger, makeup water transfer pumps, and fire pumps.

The ESF function of the FHBVS is to sweep radiolytic gases from the surface of the SFP and to treat the exhaust air in order to remove radioactive iodine (refer to Section 6.1). The purpose of the treatment of the exhaust air is to reduce the offsite dose in the event of a FHA. The sweeping effect of the ventilation air over the surface of the pool will also reduce personnel exposures in the event of a FHA.

9.4.4.1 Design Bases

9.4.4.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the FHBVS is designed to withstand the effects of, or is protected against, natural phenomena such as earthquakes, tornadoes, flooding, winds, tsunamis, and other local site effects.

9.4.4.1.2 General Design Criterion 3, 1971 – Fire Protection

The PG&E Design Class I portion of the FHBVS is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.4.4.1.3 General Design Criterion 11, 1967 – Control Room

The PG&E Design Class I portion of the FHBVS is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.4.4.1.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the PG&E Design Class I portion of the FHBVS variables within prescribed operating ranges.

9.4.4.1.5 General Design Criterion 17, 1967 – Monitoring Radioactive Releases

The FHBVS is designed to provide means for monitoring the FHB atmosphere and the FHB effluent discharge path for radioactivity that could be released from normal operations from anticipated transients or from a FHA.

9.4.4.1.6 General Design Criterion 21, 1967 – Single Failure Definition

The PG&E Design Class I portion of the FHBVS is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event are treated as a single failure.

9.4.4.1.7 General Design Criterion 37, 1967 – Engineered Safety Features Basis for Design

The ESF portion of the FHBVS provides a significant reduction in the amounts of volatile radioactive materials that could be released to the atmosphere in the event of a FHA.

9.4.4.1.8 General Design Criterion 38, 1967 – Reliability and Testability of Engineered Safety Features

The ESF portion of the FHBVS is designed to provide high functional reliability and ready testability.

9.4.4.1.9 General Design Criterion 40, 1967 – Missile Protection

The PG&E Design Class I, ESF portion of the FHBVS is designed to be protected against dynamic effects and missiles that might result from plant equipment failures.

9.4.4.1.10 General Design Criterion 41, 1967 – Engineered Safety Features Performance Capability

The ESF portion of the FHBVS is designed to provide sufficient performance capability to accommodate a partial loss of installed capacity, such as a single failure of an active component, and still perform its required safety function.

9.4.4.1.11 General Design Criterion 69, 1967 – Protection Against Radioactivity Release from Spent Fuel and Waste Storage

The ESF portion of the FHBVS provides a significant reduction in the amounts of volatile radioactive materials that could be released to the atmosphere in the event of a FHA.

9.4.4.1.12 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The ESF portion of the FHBVS is designed with provisions for maintaining control over the plant's radioactive gaseous effluents.

9.4.4.1.13 Fuel Handling Building Ventilation System Safety Function Requirements

(1) <u>Protection from Missiles</u>

The PG&E Design Class I, non-ESF portion of the FHBVS is designed to be protected against internal missiles generated outside containment that might result from plant equipment failure and from events and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(2) <u>Protection Against High Energy Pipe Rupture Effects</u>

The PG&E Design Class I non-ESF FHBVS is designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(3) Protection from Moderate Energy Pipe Rupture Effects

The PG&E Design Class I FHBVS is designed to be protected against the effects of moderate energy pipe failure to the extent necessary in order to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(4) Protection from Flooding Effects

The PG&E Design Class I FHBVS is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(5) Cooling of PG&E Design Class I Equipment

The PG&E Design Class I, non-ESF portion of the FHBVS is designed to provide cooling to PG&E Design Class I pumps.

9.4.4.1.14 Safety Guide 13, March 1971 – Fuel Storage Facility Design Basis

Regulatory Position 4:

The PG&E Design Class I, ESF portion of the FHBVS is designed to limit the potential release of radioactive iodine and other radioactive materials in the event of a significant release of radioactivity from the fuel following a FHA.

9.4.4.1.15 Regulatory Guide 1.52, June 1973 – Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The ESF portion of the FHBVS design and maintenance conforms with the guidance provided in Regulatory Guide 1.52, June 1973 with exceptions noted in Table 9.4-2.

9.4.4.1.16 Regulatory Guide 1.52, Revision 2, March 1978 – Design, Testing, and Maintenance Criteria for Post-Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The VFTP conforms with the guidance provided in Regulatory Guide 1.52, Revision 2, Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.a as described in Table 9.4-2. The VFTP also conforms with the guidance provided in Generic Letter 83-13, March 1983.

9.4.4.1.17 Regulatory Guide 1.97, Revision 3, May 1983 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

Instrumentation is provided in the control room to monitor the FHBVS ventilation damper position status for post-accident indication.

9.4.4.1.18 NUREG-0737 (Item II.F.1), November 1980 – Clarification of TMI Action Plan Requirements

Item II.F.1 – Additional Accident Monitoring Instrumentation:

Position (1) – The FHBVS is designed with noble gas effluent monitors that are installed with an extended range designed to function during accident conditions.

Position (2) – The FHBVS is designed with provisions for sampling of plant effluents for post-accident releases of radioactive iodines and particulates and onsite laboratory capabilities.

9.4.4.2 System Description

Design values for the FHBVS are listed in Table 9.4-10. The requirements for the system design are as follows:

(1) Provide an air flow pattern sweeping the SFP surface

(2) Meet the single failure criteria (refer to Section 9.4.4.3.6)

(3) Remove more air than is supplied so that all potential air leakages will be inward

(4) Automatically function in the event of a FHA involving recently irradiated fuel

(5) Provide pretreatment of supply air

- (a) Roughing filters
- (b) Heating provisions

(6) Provide post-treatment of exhaust air

- (a) Particulate
- (b) Gaseous

(7) Provide ventilation to the FHB, including the equipment rooms containing the AFW pumps, SFP pumps, SFP heat exchanger, makeup water transfer pumps, and fire pumps.

(8) Design, build, and install equipment according to design classifications given in Table 3.2-3.

The evaluation of the fission product removal performance of the system is contained in Section 15.5, in connection with the description of the FHAs.

The FHBVS has the capability to provide ventilation air for the FHB separately from the auxiliary building. The FHB for each unit is physically isolated from the auxiliary building. The system as shown in Figure 9.4-3 consists of redundant supply and exhaust fans, and redundant HEPA and charcoal filter banks. A third full capacity exhaust fan and HEPA filter bank train is provided for normal operation. Each HEPA filter bank is preceded by a roughing filter bank. The major equipment for the system is listed in Table 9.4-6. The supply airflow, was selected on the basis of the heat dissipated by the equipment. The exhaust air flow (35,750 cfm) consists of approximately 81 percent exhausted from the SFP area by drawing the air flow over the pool. The balance of the flow is ducted and exhausted by the exhaust fan from other areas in the FHB.

The heating coil is used to temper the air that is supplied to the area. The supply air is ducted to corridors and equipment compartments on the floor levels below the operating level. The air flow pattern is from these compartments through the spent fuel shipping cask decontamination area up to the SFP. Exhaust grilles over and along one side of the SFP draw the air in a sweep across the pool surface. The exhaust air is then filtered and discharged to the plant vent.

The FHB is separated from the rest of the auxiliary building by partitions and doors on all floors. The separating doors, partitions, and outside walls are of standard construction with no particular leaktight consideration. The exhaust fans remove more air than is supplied. This extra air is made up by infiltration from the outside and adjoining areas of the auxiliary building.

The system is provided with three modes of operation that affect the filtering of the exhaust air.

Normal Mode: The first mode is for normal use. The mode may be selected manually from pool side or from the control room. Under this mode all of the exhaust air passes through roughing and HEPA filter banks only. The exhaust fan for this mode is powered from a non-Class 1E 480-V bus.

Iodine Removal Mode: The second mode of operation is for the removal of potential radioactive particulates and/or radioactive gases in the exhaust air. This mode of operation is automatically initiated by an exhaust fan failure while in

Normal Mode, or it may be selected manually from the pool side or from the control room. This mode routes all the exhaust air through roughing, HEPA, and charcoal filters. The fans and filter banks for this mode of operation are redundant. The fans are powered from separate Class 1E 480-V buses.

Automatic Iodine Removal Mode: The third mode of operation (emergency mode) is also for the removal of radioactive particulate and/or radioactive gases in the exhaust air. This mode is physically the same as Iodine Removal Mode except for automatic initiation by a radiation detector. The radiation detectors located near the fuel storage areas will automatically initiate this mode when radiation levels exceed the setpoint level in Table 11.4-1.

9.4.4.3 Safety Evaluation

9.4.4.3.1 General Design Criterion 2, 1967 – Performance Standards

The FHBVS is designed to perform its safety functions under the effects of earthquakes (refer to Section 3.10.3.33), winds and tornadoes (refer to Section 3.3.2.5.2.3), and external missiles (refer to Section 3.5.1.2). The metal siding and roofing of the FHB do not provide significant missile protection for the FHBVS due to a tornado event. However, the capability of the FHBVS to maintain a negative pressure in the FHB is not required after a tornado. The FHBVS exhaust ducts and FHB radiation monitors have limited tornado resisting capabilities. Failures of these components do not affect the safe operational status of the plant and do not result in significant radiation releases since damage to the spent fuel does not occur as a result of a tornado.

The FHBVS is located within the FHB; a PG&E Design Class I structure (refer to Section 3.8.2.3). The FHB is designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), and earthquakes (refer to Section 3.7). This design protects the FHBVS, ensuring its design functions can be performed during these events.

9.4.4.3.2 General Design Criterion 3, 1971 – Fire Protection

The FHBVS is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.4.4.3.3 General Design Criterion 11, 1967 – Control Room

The FHBVS is normally controlled from the control room. Controls are provided on the main control board which allow for system startup and shutdown, fan selection, and for the selection of system operating configuration and modes (refer to Section 9.4.4.2).

Indication is provided on the main control board to indicate fan failure or damper out of position. Annunciators in the control room alert operators to abnormal conditions and system or equipment failures (refer to Section 9.4.4.3.4).

9.4.4.3.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

The control logic devices for both DCPP Unit 1 and Unit 2 FHB are solid-state units providing a full selection of logic functions, designed for binary system operation. The control system is based on a PLC. The basis for qualification of the Triconex PLC and associated software follows the guidance of Branch Technical Position 7-14 and 7-18 (References 21 and 22). The logic control system has output to solenoids and dry contact relays for the control functions. Both the logic controls and relays are redundant and have power sources from Class 1E buses.

The FHBVS has manual selection for Normal Mode and Iodine Removal Mode operations. It also has an automatic initiation of the Iodine Removal Mode. The automatic initiation is from radiation monitors mounted near the fuel storage areas. Radiation levels greater than the setpoint level will change the operation of the system from Normal Mode to Automatic Iodine Removal Mode operation. For setpoint radiation level, refer to Table 11.4-1.

High temperature alarms are provided to the main annunciator for the FHB supply fan room.

9.4.4.3.5 General Design Criterion 17, 1967 – Monitoring Radioactive Releases

Radiation detection instruments are located in the SFP area and the new fuel storage area. These instruments continuously detect operating radiation levels. If the radiation levels should rise above the setpoints listed for each channel (refer to Table 11.4-1), an alarm is initiated in the control room. Local annunciation is also provided at the detectors to indicate high radiation levels to personnel in the area.

The air flow from the FHB is monitored for abnormal radiation levels of both particulate and gaseous nature at the plant vent monitor.

9.4.4.3.6 General Design Criterion 21, 1967 – Single Failure Definition

The FHBVS has redundancy for all non-static (active) components. The fans are each full capacity. The exhaust fan (E-4) and corresponding filter bank used for Normal Mode are not necessary for emergency operation and are not redundant. If the exhaust fan or its associated mode damper used in the Normal Mode were to fail, Iodine Removal Mode (fans E-5 and E-6) would be automatically initiated.

If a supply fan or its associated mode damper should fail, or if the exhaust fan or exhaust mode damper used in Iodine Removal Mode should fail, the redundant fan and damper system for that mode would be automatically started. Off-delay timers, set for a nominal time delay of 2 seconds, keep supply fans running during a change of ventilation mode to prevent the loss of supply fan air flow and the subsequent shutting down of the supply fan.

The power sources for the redundant fans are taken from Class 1E 480-V buses with separation of redundant components.

The initiating (logic) circuitry, including the relays and solenoids required to actuate the system, is redundant. Each control train serves similar control functions and switches circuits in the event of a failure to the other train. Separation has been maintained for the electrical circuits from each Class 1E bus.

All ductwork has been designed to withstand any internal pressure that may be generated by any fan in the system, and braced according to PG&E Design Class I criteria to prevent earthquake damage to the ducts (refer to Section 9.4.4.3.1). All static (passive) ventilation components (i.e., volume dampers, air diffusers) have locking devices to prevent accidental closing. No duct liner or other insulating substance that might sag or fall down, thereby blocking the duct, has been installed in any ductwork.

9.4.4.3.7 General Design Criterion 37, 1967 – Engineered Safety Features Basis for Design

9.4.4.3.7.1 Air Flow Pattern

The ventilating air is discharged from duct work into the corridors and equipment compartments below the SFP floor. The air exhausts from the FHB after passing over the pool surface. Two exhaust air headers remove the air from above and from one side of the pool to achieve a sweeping movement of gases above the pool surface.

The FHB is separated from the rest of the auxiliary building by partitions and doors on all floors. The separating doors, partitions, and outside walls are of standard construction with no particular leaktight consideration. The exhaust fans remove more air than is supplied. This extra air is made up by infiltration from the outside and adjoining areas of the auxiliary building.

The dampers are positioned with pneumatic actuators with air supplied from the plant compressed air system. The dampers are designed to assume the position required for emergency conditions on the failure of the control air supply.

9.4.4.3.7.2 Effects of Ventilation System Failure

The administrative controls specified in the Equipment Control Guidelines preclude the handling of recently irradiated fuel in the event of ventilation system inoperability or failure. Although it is conceivable that a FHA might occur coincident with a failure in the FHBVS, this combination of events is not regarded to be of sufficient likelihood to warrant system design changes. In any event, personnel will be leaving the FHB immediately following any indication of a FHA. Personnel exposure is not expected to be significant in any of these events because of the presence of the area monitor and continuous air monitor functioning during any operations involving irradiated fuel. These provisions are further described in Section 12.1.4 and the Technical Specifications.

9.4.4.3.7.3 Treatment of Air

The supply air for the FHB passes through roughing filters to remove dust and lint that may be in the atmosphere. The supply filters have a minimum dust spot efficiency of 30 percent for atmospheric dust (National Bureau of Standards or Air Filter Institute method).

The exhaust air from the FHB is filtered through roughing filters and HEPA filters during normal operation. Iodine Removal Mode operation has, in addition, charcoal filters (refer to Section 9.4.4.2). The charcoal filter banks have been sized to take the full air flow of the ventilation system without exceeding the manufacturer's recommendations for flow through each individual module of the bank. Thirty-six modules with three filter trays per module are provided for each full capacity filter bank. The total amount of activated impregnated charcoal in each filter bank is a function of the charcoal-containing capacity and the density of the charcoal. The amount of charcoal contained in each filter bank is adequate to adsorb radioactive gases from any projected design FHA without overloading with iodine or overheating from decay heat. The exhaust air for the system is routed to the plant vent. All air flow through the plant vent is monitored for radioactivity.

9.4.4.3.8 General Design Criterion 38, 1967 – Reliability and Testability of Engineered Safety Features

The ESF portion of the FHBVS is designed with provisions for periodic tests, inspections, and surveillance to ensure that the systems will be dependable and effective when called upon to function.

Testing of the FHBVS is in accordance with the requirements of the Technical Specifications, including the VFTP as described in Sections 9.4.4.3.15 and 9.4.4.3.16.

9.4.4.3.9 General Design Criterion 40, 1967 – Missile Protection

There are no credible missiles outside of containment resulting from plant equipment failure that would prevent the PG&E Design Class I, ESF portion of the FHBVS from performing its design functions (refer to Section 3.5.1.2).

Dynamic effects as a result of plant equipment failure will not prevent the PG&E Design Class I, ESF portion of the FHBVS from performing its design function of supporting ESFs (refer to Section 3.6).

9.4.4.3.10 General Design Criterion 41, 1967 – Engineered Safety Features Performance Capability

The ESF portion of the FHBVS has redundancy for all non-static (active) components. The power sources for the redundant fans are taken from Class 1E 480-V buses with separation of redundant components (refer to Section 9.4.4.3.6). Therefore, the FHBVS provides sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required functions of sweeping radiolytic gases from the surface of the SFP and treating the exhaust air in order to remove radioactive iodine.

9.4.4.3.11 General Design Criterion 69, 1967 – Protection Against Radioactivity Release from Spent Fuel and Waste Storage

The FHBVS consists of redundant supply and exhaust fans, redundant HEPA and charcoal filter banks, and redundant controls and Class 1E 480-V power (refer to Section 9.4.4.3.6).

In the event of a FHA, radiation detectors located near the fuel storage areas will automatically initiate the iodine removal mode when radiation levels exceed the setpoint level in Table 11.4-1. This mode routes all the exhaust air through roughing, HEPA, and charcoal filters (refer to Section 9.4.4.2).

The charcoal filters are designed to adsorb radioactive gases from any projected design FHA without overloading with iodine or overheating from decay heat. The exhaust air for the system is routed to the plant vent. All air flow through the plant vent is monitored for radioactivity (refer to Section 9.4.2.3.6).

The assumptions used in the FHA analyses are presented in Section 15.5.22.

9.4.4.3.12 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The ESF portion of the FHBVS has been designed to pass exhaust air from the FHB through HEPA filters and charcoal filter banks before reaching the plant vent

9.4.4.3.13 Fuel Handling Building Ventilation System Safety Function Requirements

(1) Protection from Missiles

There are no credible missiles outside of containment resulting from plant equipment failure that would prevent the non-ESF portion of the PG&E Design Class I FHBVS from performing its design functions (refer to Section 3.5.1.2).

(2) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect the non-ESF portions of the PG&E Design Class I FHBVS from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained as discussed in Section 3.6.

(3) Protection from Moderate Energy Pipe Rupture Effects

The provisions taken to provide protection to the PG&E Design Class I essential systems located outside containment from moderate energy pipe rupture effects (water spray) are discussed in Section 3.6.

(4) Protection from Flooding Effects

The provisions taken to provide protection to the PG&E Design Class I essential systems from flooding that might result from the effects associated with a postulated rupture of high and moderate energy piping are discussed in Section 3.6.

(5) Cooling of PG&E Design Class I Equipment

Although not an ESF function, the FHBVS supplies cooling air to the ESF AFW pump rooms.

The FHBVS provides ventilation at ambient design air temperature, as described in the introduction of Section 9.4, to the FHB operating floor and the equipment described in Section 9.4.4.2(7). The ambient indoor air temperature in the FHB will be maintained below 104°F except the Unit 1 SFP pump room has a design room temperature of 109°F and the Unit 2 SFP pump room has a design room temperature of 112°F.

9.4.4.3.14 Safety Guide 13, March 1971 – Fuel Storage Facility Design Basis

Regulatory Position 4:

The PG&E Design Class I, ESF portion of the FHBVS is designed with appropriate containment, confinement, and filtering systems to limit the potential release of radioactive iodine and other radioactive materials in the event of a FHA in which the cladding of all of the fuel rods in one fuel bundle are breached. Refer to Section 15.5.22 for the radiological consequences of a FHA.

The FHB is separated from the rest of the auxiliary building by partitions and doors on all floors. The separating doors, partitions, and outside walls are of standard construction with no particular leaktight consideration. The exhaust fans remove more air than is supplied. This extra air is made up by infiltration from the outside and adjoining areas of the auxiliary building.

The exhaust air for the FHB is filtered through roughing filters and HEPA filters during normal operation. Iodine Removal Mode operation has, in addition, charcoal filters. The charcoal filter banks have been sized to take the full air flow of the FHBVS without exceeding the manufacturer's recommendations for flow through each individual module of the bank.

The radiation detectors located near the fuel storage areas will automatically initiate this mode when radiation levels exceed the setpoint level in Table 11.4-1 (refer to Sections 9.1.2.3.5 and 11.4.2.1.4.1).

In the event of SFP cooling system failure, exhaust filters of the FHBVS are tested after the period of emergency conditions occurs. The filter efficiency tests are conducted in accordance with the FHBVS testing requirements discussed in Section 9.4.4.4.2. These tests are performed in addition to the periodic testing described in the Plant Surveillance Test Procedure.

9.4.4.3.15 Regulatory Guide 1.52, June 1973 – Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The ESF portion of the FHBVS provides the capability to control the volatile radioactive material that could exist to the atmosphere. The system is analyzed to the positions in Regulatory Guide 1.52, June 1973 (Reference 11) as shown in Table 9.4-2 with exceptions noted for specific items. Requirements for the performance of ventilation filter testing are stated in the DCPP VFTP.

9.4.4.3.16 Regulatory Guide 1.52, Revision 2, March 1978 – Design, Testing, and Maintenance Criteria for Post-Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

The VFTP provides the requirements for the testing of charcoal samples from the FHBVS charcoal adsorbers in accordance with ASTM D 3803-1989, which complies with Regulatory Guide 1.52, Revision 2, and Generic Letter 83-13, March 1983, for laboratory testing of charcoal samples, and in-place penetration and bypass leakage testing in accordance with ANSI N510-1980. The VFTP complies with the intent of Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.a requirements as described in Table 9.4-2, with exceptions noted.

9.4.4.3.17 Regulatory Guide 1.97, Revision 3, May 1983 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

Damper position indication in the control room provides status of the FHBVS supply and exhaust fan damper positions (refer to Table 7.5-6 for a summary of compliance to Regulatory Guide 1.97, Revision 3).

9.4.4.3.18 NUREG-0737 (Item II.F.1), November 1980 – Clarification of TMI Action Plan Requirements

Item II.F.1 – Additional Accident Monitoring Instrumentation:

Position (1) – Extended range noble gas effluent monitoring is installed in the plant vent and is designed to function during accident conditions (refer to Sections 7.5.2.3 and 11.4.2.1.2.1).

Position (2) – Installed capability is provided in the plant to obtain samples of the particulate and iodine radioactivity concentrations which may be present in the gaseous effluent being discharged to the environment from the plant under accident and post-accident conditions. The TSC laboratory is available for onsite testing of the containment air samples.

9.4.4.4 Tests and Inspections

9.4.4.4.1 Initial System Tests and Inspections

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

The system was installed under field inspection by PG&E General Construction and Quality Assurance personnel. An initial checkout of the motors, dampers, controls, etc., was made at that time. A system air balance test and adjustment to design conditions were conducted. The final tests performed prior to actual operation of the FHBVS were the functional tests of the HEPA and charcoal filter banks. These tests determined the overall efficiency of the filter banks.

9.4.4.4.2 Routine System Tests

Refer to Section 9.4.4.3.8 for information on routine system testing requirements.

9.4.4.5 Instrumentation Applications

Refer to Sections 9.4.4.3.3 and 9.4.4.3.4 for discussions of the instrumentation requirements and controls for the FHBVS.

9.4.5 CONTAINMENT

The containment HVAC system is designed to maintain temperature and pressure within the containment at acceptable levels for equipment operation and personnel access at power for inspection, maintenance, and testing. The ESF function of the containment fan coolers is to reduce the containment atmosphere temperature and pressure following a LOCA or MSLB and is addressed in Section 6.2.2.

The containment HVAC system includes the following:

(1) Containment fan cooler system (CFCS)

(2) Control rod drive mechanism (CRDM) exhaust system

(3) Iodine removal system

(4) Incore instrument room cooling system (abandoned in place)

(5) Containment purge system

(6) Pressure relief line

(7) Vacuum relief line

The CFCS is also designed to operate during accident conditions as a part of the containment heat removal system and is described in Section 6.2.2. Codes and standards applicable to the containment HVAC system are listed in Table 9.4-8.

9.4.5.1 Design Bases

9.4.5.1.1 General Design Criterion 2, 1967 - Performance Standards

The PG&E Design Class I portions of the containment HVAC system are designed to withstand the effects of, or are protected against natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis, and other local site effects.

9.4.5.1.2 General Design Criterion 3, 1971 - Fire Protection

The containment HVAC system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.4.5.1.3 General Design Criterion 11, 1967 - Control Room

The PG&E Design Class I portion of the containment HVAC system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.4.5.1.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the containment HVAC variables within prescribed operating ranges.

9.4.5.1.5 General Design Criterion 17, 1967 - Monitoring Radioactive Releases

The containment HVAC system is designed to provide means for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

9.4.5.1.6 General Design Criterion 40, 1967 - Missile Protection

The ESF portions of the CFCS, containment purge system, and containment pressure/vacuum relief system are designed to be protected against dynamic effects and missiles that might result from plant equipment failures.

9.4.5.1.7 General Design Criterion 54, 1971 - Piping Systems Penetrating Containment

The PG&E Design Class I containment purge system isolation valves and the containment pressure/vacuum relief systems penetrating primary reactor containment are provided with leak detection, isolation and containment capabilities having redundancy, reliability and performance capabilities which reflect the importance to safety of isolating these piping systems. These systems are designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if the valve leakage is within acceptable limits.

9.4.5.1.8 General Design Criterion 56, 1971 - Primary Containment Isolation

The supply and exhaust lines connect directly to the containment atmosphere and penetrate primary containment. The supply and exhaust lines are provided with one automatic isolation valve inside and one automatic isolation valve outside containment. The purge system isolation valves outside containment are located as close to the containment as practical and upon loss of actuating power, the automatic isolation valves are designed to take the closed position.

9.4.5.1.9 Containment HVAC System Safety Function Requirements

(1) Cooling of PG&E Design Class I Equipment

The containment HVAC system is designed to provide cooling to the containment atmosphere to maintain the temperature environment within design conditions during normal operation.

(2) Protection from Jet Impingement – Inside Containment

The CFCS and CIS portion of the containment purge system and containment pressure/vacuum relief system located inside containment are designed to be protected against the effects of jet impingement which may result from high energy pipe rupture.

9.4.5.1.10 10 CFR 50.49 - Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants

The containment HVAC components that require EQ are qualified to the requirements of 10 CFR 50.49.

9.4.5.1.11 10 CFR 50.55a(f) - Inservice Testing Requirements

ASME code components of the containment purge system and the containment pressure/vacuum relief system CIVs are tested to the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5) to the extent practical.

9.4.5.1.12 10 CFR 50.55a(g) - Inservice Inspection Requirements

ASME code components of the containment purge system and the containment pressure/vacuum relief system piping and valves, and their supports are inspected to the requirements of 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(5) to the extent practical.

9.4.5.1.13 Regulatory Guide 1.97, Revision 3, May 1983 - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

The containment HVAC system provides instrumentation to monitor system variables during and following an accident.

9.4.5.1.14 NUREG-0737 (Item II.F.1), November 1980 - Clarification of TMI Action Plan Requirements

Item II.F.1 - Additional Accident Monitoring Instrumentation:

Position (1) – The containment HVAC system is designed with noble gas effluent monitors with an extended range designed to function during accident conditions.

Position (2) – The containment HVAC system is designed with provisions for sampling of plant effluents for post-accident releases of radioactive iodines and particulates and onsite laboratory capabilities.

9.4.5.2 System Description

The containment HVAC system is designed to perform as follows:

(1) Maintain the containment ambient temperature between 50 and 120°F during normal plant operation

(2) Maintain temperatures of 150°F or below in the CRDM shroud area and 135°F or below inside the primary concrete shield during normal plant operation

(3) Maintain a pressure between –1.0 psig and +1.2 psig in the containment during normal operation

(4) Provide the proper atmosphere and adequate ventilation for personnel before and during periods of personnel access for refueling operations and maintenance when the plant is shut down

(5) CFCS works in conjunction with the containment spray system to reduce the containment ambient temperature and pressure during accident conditions (refer to Section 6.2.2)

(6) Accept a single active failure and still provide adequate cooling to the components inside the containment (refer to Section 6.2.2)

The containment HVAC system shares some of these design bases with the CCS, which is described in detail in Section 6.2.3.

The containment HVAC system is designed, built, and installed according to the design classifications given in Table 3.2-3 and the DCPP Q-List (Reference 8 of Section 3.2).

The containment HVAC system is shown schematically in Figures 9.4-3 and 9.4-3A.

The fan coolers and their fan/motor couplings that limit reverse rotation (anti-reverse rotation device) are PG&E Design Class I. The connected system ductwork is PG&E Design Class II and the supports are seismically qualified.

9.4.5.2.1 Containment Fan Cooler System

The design bases for the sizing of the HVAC equipment are the normal operational heat sources in the containment as given in Table 9.4-11.

The CFCS is located inside the containment but outside the missile shield and consists of five coolers, ductwork and supports as shown in Figure 9.4-4. The fan coolers are designed to cool the containment air. The air flow for normal operation is 110,000 cfm per unit for the Westinghouse cooling coils and 98,000 cfm per unit for the Super Radiator Coil (SRC) cooling coils with a total system static pressure of 8 inches w.g. at 0.075-lb/ft^3 air density. This system is a total recirculation system. The cooling coils are each sized to remove 3.14 x 10^6 Btu/hr for the Westinghouse cooling coils and 2.501 x 10^6 Btu/hr for the SRC cooling coils from 120°F entering air when supplied with 90°F cooling water. The normal operational design requires that up to four out of the five fan coolers remain in operation. Normally up to four out of five fan coolers operate to recirculate and cool the air within the containment during normal plant operation by drawing the air through their inlet dampers, cooling coils, fans, and ductwork, which allows for the estimated heat removal capacity given in Table 9.4-12. The total heat load for the NSSS is given in Table 9.4-11.

The CFCS is designed to operate during accident conditions as a part of the containment heat removal system and is described in Section 6.2.2. The fan coolers will reduce the motor speed from nominal 1200 to 600 rpm, draw the high density steam through cooling coils, and provide 81 x 10^6 Btu/hr cooling capacity per cooler for the Westinghouse cooling coils and 75.489 x 10^6 Btu/hr for the SRC cooling coils during accident conditions. The safety evaluation is described in Section 6.2.2.

The fan cooler units are powered from the Class 1E 480-V system.

9.4.5.2.1.1 Evaluation of Cooling Water Supply

The normal cooling water requirements for all five fan coolers can be supplied by any two of the three CCW pumps. In addition, one of the two ASW pumps is required to provide ASW cooling water to the CCW heat exchanger.

Water flow through each fan cooler is balanced to the design flow by a manual valve on the discharge header from the cooling units.

Fouling of the waterside of the heat transfer area is minimized with the use of buffered condensate in the component cooling system. If a complete severance of a fan cooler water tube is postulated, double-ended flow must be assumed. This flow can be accommodated by the trough under the fan coolers and is piped to the containment sump.

The fan coolers are supplied by individual lines from the CCW headers. Each unit inlet and discharge line is provided with a manual shutoff valve and drain valve. This permits isolation of each cooler for testing purposes.

The evaluation of cooling water supply following accident conditions is discussed in Section 6.2.2.

9.4.5.2.2 Control Rod Drive Mechanism Exhaust System

The CRDM ventilation system consists of three exhaust fans mounted on the air plenum of the IHA. Two out of three fans operate at 73,500 cfm total, exhaust air from the area surrounding the CRDMs, and discharge to the containment atmosphere to remove heat from the CRDM area during normal plant operation. This system is not designed to operate during accident conditions.

Evaluation of the ventilation provisions for the primary shield, neutron detectors and cables, and CRDMs indicates that the present designs are adequate to ensure plant safety during normal plant operating conditions. Loss of air cooling during normal plant operation would be indicated by temperature instrumentation provided for this purpose. In general, the effects of elevated temperature on the above equipment take place gradually over a period of hours, so that sufficient time would be available to take appropriate corrective action, including an orderly plant shutdown, to avert any possible safety problem. With respect to accident conditions, none of this equipment is required to function during the post-accident recovery period.

9.4.5.2.3 Iodine Removal System

The iodine removal system consists of two one-half capacity iodine removal units that have roughing filters, HEPA filters, and charcoal filter banks. These units are provided for pre-entry cleanup of the containment atmosphere. The iodine removal system is not designed to be operated during accident conditions.

The iodine removal system is used to reduce the concentration of fission product particulate activities in the containment atmosphere prior to routine personnel access at power or in advance of a scheduled reactor shutdown. With sufficient reduction of these activities, particularly iodine and cesium, the personnel dose is due mainly to whole body and inhalation exposures from the unfilterable noble gases. Total capacity is based on the I-131 activity required to limit the occupational airborne concentration of this isotope to approximately seven times 10 CFR 20.1-20.601 maximum permissible concentration. The original design of the iodine removal system was to the pre-1994 regulation maximum permissible concentration value of 9×10^{-9} μCi/cc. The result is consistent with the total exposure limitation of 100 mR received during a 2-hour access period. On this basis, two 12,000 cfm capacity units are provided. This capacity is based on assumptions of 1 percent defective fuel cladding and a 50 pound/day leakage from the RCS. The I-131 activity is reduced to the design equilibrium value by operating the two units for 15 hours.

9.4.5.2.4 Containment Purge System

Prior to entry of personnel into the containment shortly before or after shutdown from normal power operation, the airborne radioactive concentration in the containment atmosphere can be reduced as necessary by employing the iodine cleanup and containment purge systems. After the cleanup process, the containment purge system provides the supply air to and exhaust air from the containment for purge and ventilation.

The section of duct between the containment purge exhaust isolation valve and its debris screen, including the flexible connection, is classified as PG&E Design Class I. The remaining ductwork is PG&E Design Class II and the supports are seismically qualified.

The isolation valves are air-operated valves and are designed to require air pressure to stay open, and fail closed on a loss of air or loss of power to the solenoid valves on their air supply valves. Refer to Section 9.4.5.3.8 for a discussion of GDC 56, 1971, Primary Containment Isolation.

9.4.5.2.5 Pressure Relief Line

The containment pressure relief line connects to the suction side of the containment purge fan, E-3, which then discharges into the plant vent. The pressure relief flow is driven by the pressure differential between the containment and the outside atmosphere, and does not require operation of fan E-3.

9.4.5.2.6 Vacuum Relief Line

The vacuum relief line uses suction air from the containment purge air supply fan plenum to release the vacuum inside the containment. The vacuum relief line takes air from the containment purge system supply fan plenum so that the suction air can be filtered and heated before it goes into the containment. The function is independent of the purge system operation and manually operated.

9.4.5.2.7 Incore Instrument Room Cooling System

The incore instrument room cooling system is abandoned in place and is no longer in use.

9.4.5.2.8 Performance Objectives

The performance objectives of the containment HVAC system design are as follows:

(1) Normal Operation

(a)	Flowrates (4 of 5 fan coolers operating)	440,000 cfm
(b)	Heat transfer (4 of 5 fan coolers operating)	12.56×10^6 Btu/hr
(c)	Temperature range	50 to 120°F max
(d)	Humidity	no requirement
(e)	Flowrate iodine removal units (2 units operating)	24,000 cfm
(f)	Time to achieve equilibrium value I-131 - 1 percent defective cladding - 50 lb/day RCS leakage	15 hours
(g)	Flowrate CRDM ventilation (2 out of 3 fans operating)	73,500 cfm
(h)	Temperature in the cavity area above the CRDM shroud	127°F max
(i)	Heat removal for CRDM	2.26×10^6 Btu/hr
(j)	Containment purge flow inlet outlet	 50,000 cfm 55,000 cfm

(k) Pressure relief flow instantaneous, maximum — 4655 cfm

(l) Incore instrument room fan coil unit (Abandoned in place)

(2) Emergency Operation

(a) Flowrates, fan coolers (2 out of 5 units operating) — 94,000 cfm

(b) Pressure in containment — 47 psig at saturated steam-air mixture

(c) Heat transfer - fan cooler units (2 out of 5 operating) — 81×10^6 Btu/hr (each cooler)

(d) Iodine removal units — shut down

(e) CRDM ventilation — shut down

(f) Containment purge system — shut down

(g) Incore instrument room fan coil unit(Abandoned in place)

9.4.5.2.9 Single Failure Criteria

The containment HVAC system is designed to meet single failure criteria. The fan cooler units are powered from Class 1E buses and have a standby unit. The iodine removal units are not necessary for cleanup during accident conditions, so they are neither redundant nor powered from Class 1E buses. The CRDM exhaust fans are powered from a non-Class 1E bus. The iodine cleanup system is not considered an ESF and is not designed to the requirements of Regulatory Guide 1.52. Because of its very sporadic use, it is unlikely to be operating at the time of a LOCA. If operating, the units will be manually shut off (refer to Section 6.2.2).

9.4.5.3 Safety Evaluation

9.4.5.3.1 General Design Criterion 2, 1967 - Performance Standards

The containment building and the auxiliary building are PG&E Design Class I (refer to Section 3.8). These buildings or portions thereof are designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), and earthquakes (refer to Section 3.7), and to protect SSCs from damage that may result from these events.

The PG&E Design Class II CFCS ductwork and duct supports are seismically qualified to protect other equipment important to safety during a seismic event.

The containment purge and vacuum/pressure relief line containment penetration piping and isolation valves are PG&E Design Class I. The remainder of the purge system is PG&E Design Class II. The PG&E Design Class II duct and duct supports of the purge system are seismically qualified where required for systems interaction concerns or to assure that the containment effluent flowpath is monitored by radiation monitors.

The CRDM exhaust system is PG&E Design Class II. Equipment supports are seismically qualified.

9.4.5.3.2 General Design Criterion 3, 1971 - Fire Protection

The containment HVAC system is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.4.5.3.3 General Design Criterion 11, 1967 - Control Room

The CFCS is initiated for normal operation by manual switches in the control room. The HSP also provides for remote control of the CFCS. The design details and logic of the instrumentation are discussed in Chapter 7.

Instrumentation is provided for indication and monitoring of containment air temperature in the control room.

9.4.5.3.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

The CFCS is initiated for normal operation by manual switches in the control room. The design details and logic of the instrumentation are discussed in Chapter 7.

The iodine removal units and the CRDM fans are operated by manual switches in the control room and the 480-V switchgear room area, respectively. Indication is provided to the operator as to the operation of these fans.

Evaluation of the ventilation provisions for the primary shield, neutron detectors and cables, and CRDMs indicates that the present designs are adequate to ensure plant safety during normal plant operating conditions. Loss of air cooling during normal plant operation would be indicated by temperature instrumentation provided for this purpose. In general, the effects of elevated temperature on the above equipment take place gradually over a period of hours, so that sufficient time would be available to take appropriate corrective action, including an orderly plant shutdown, to avert any possible safety problem.

9.4.5.3.5 General Design Criterion 17, 1967 - Monitoring Radioactive Releases

The purge exhaust fan takes suction from the containment ventilation distribution duct system via a branch duct off the annular ring connecting to the containment purge exhaust isolation valve. The containment purge flow, also used for ventilation during extended outages, is routed to the plant vent for monitored exhaust. The containment atmosphere is monitored for radioactivity by the containment and plant vent air particulate and/or gas effluent monitors.

The radiological monitoring systems are further described in Section 11.4. The offsite radiological monitoring program is described in Section 11.6.

9.4.5.3.6 General Design Criterion 40, 1967 - Missile Protection

The provisions taken to protect the ESF portion of the CFCS, containment purge system, and containment pressure/vacuum relief system from damage that might result from missiles and dynamic effects associated with equipment and high-energy pipe failures are discussed in Sections 3.5, 3.6, and 6.2.4.

9.4.5.3.7 General Design Criterion 54, 1971 - Piping Systems Penetrating Containment

The containment HVAC system isolation valves required for containment closure are periodically tested for operability. Testing of the components required for the CIS is discussed in Section 6.2.4.

9.4.5.3.8 General Design Criterion 56, 1971 - Primary Containment Isolation

The containment HVAC system containment penetrations comply with the requirements of GDC 56, 1971, as described in Section 6.2.4 and Table 6.2-39.

9.4.5.3.9 Containment HVAC System Safety Function Requirements

(1) <u>Cooling of PG&E Design Class I Equipment</u>

The CFCS is designed to cool the containment air during normal operation (refer to Section 9.4.5.2.1).

(2) <u>Protection from Jet Impingement – Inside Containment</u>

The provisions taken to provide protection of the CFCS and CIS portion of the containment purge system and containment pressure/vacuum relief system located inside containment from the effects of jet impingement which may result from high energy pipe rupture are discussed in Section 3.6.

9.4.5.3.10 10 CFR 50.49 - Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants

Containment HVAC system SSCs required to function in harsh environments under accident conditions are qualified to the applicable environmental conditions to ensure that they will continue to perform their safety functions. Section 3.11 describes the DCPP EQ Program and the requirements for the environmental design of electrical and related mechanical equipment. The affected equipment are listed on the EQ Master List.

9.4.5.3.11 10 CFR 50.55a(f) - Inservice Testing Requirements

The DCPP IST Program applies to the PG&E Design Class I CIVs. The IST requirements for these components are contained in the IST Program Plan.

9.4.5.3.12 10 CFR 50.55a(g) - Inservice Inspection Requirements

The ISI boundary for the containment HVAC system is defined on the DCPP ISI/IST drawings. The ISI Program Plan identifies applicable inspections for the containment HVAC system.

9.4.5.3.13 Regulatory Guide 1.97, Revision 3, May 1983 - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

CIV position, containment temperature, and containment cooler operation indication is provided in the control room for Regulatory Guide 1.97, Revision 3, monitoring (refer to Table 7.5-6).

9.4.5.3.14 NUREG-0737 (Item II.F.1), November 1980 - Clarification of TMI Action Plan Requirements

Item II.F.1 - Additional Accident Monitoring Instrumentation:

Position (1) – The containment purge vents to the plant vent. Instrumentation to monitor noble gas effluents is provided in the plant vent. Refer to Section 9.4.2 for discussion of the plant vent, and Section 11.4 for discussion of the plant vent radiation monitors.

Position (2) – Installed capability is provided to obtain samples of the particulate and iodine radioactivity concentrations which may be present in the gaseous effluent discharged from the containment purge system via the plant vent. The TSC laboratory is available for onsite testing of the containment air samples.

9.4.5.4 Tests and Inspections

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

9.4.5.4.1 Initial System Tests and Inspections

An initial checkout of the motors, dampers, and controls was made at the time of installation. A system air balance test and adjustment to design conditions were conducted.

9.4.5.4.2 Routine System Tests

The containment HVAC system is tested in accordance with the requirements of the Technical Specifications.

9.4.5.5 Instrumentation Applications

Instrumentation requirements are described in Chapter 7 and Sections 9.4.5.3.3 and 9.4.5.3.4.

Instrumentation is also provided in the drain line from each fan cooler unit to indicate abnormally large flow.

9.4.6 AUXILIARY SALTWATER VENTILATION SYSTEM

The ASW ventilation system has the function of maintaining the temperature of the ASW pump motors within acceptable limits during their operation. The ASW ventilation system is the only PG&E Design Class I ventilation system located within the Intake Structure and must be in operation while the ASW pumps are operating. The fans and ductwork of the ASW ventilation system are PG&E Design Class I, while the Intake Structure itself is PG&E Design Class II. The ventilation shaft (or “snorkel”) is discussed in the ASW section (refer to Section 9.2.7.3).

9.4.6.1 Design Bases

9.4.6.1.1 General Design Criterion 2, 1967 – Performance Standards

The ASW ventilation system is designed to withstand the effects of, or is protected against, natural phenomena such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.4.6.1.2 General Design Criterion 3, 1971 – Fire Protection

The ASW ventilation system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.4.6.1.3 General Design Criterion 11, 1967 – Control Room

The ASW ventilation system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.4.6.1.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the ASW ventilation system variables within prescribed operating ranges.

9.4.6.1.5 General Design Criterion 21, 1967 – Single Failure Definition

The ASW ventilation system is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event shall be treated as a single failure.

9.4.6.1.6 Auxiliary Saltwater Ventilation System Safety Function Requirements

(1) Cooling of PG&E Design Class I Equipment

The ASW ventilation system is designed to provide heat removal to ASW pumps, ASW MOVs, and ASW piping to maintain the environment within design conditions.

(2) Protection from Missiles

The PG&E Design Class I portion of the ASW ventilation system is designed to be protected against missiles that might result from plant equipment failure and from events and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(3) Protection from Moderate Energy Pipe Rupture Effects

The PG&E Design Class I portion of the ASW ventilation system is designed to be protected against the effects of moderate energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(4) Protection from Flooding Effects

The PG&E Design Class I portion of the ASW ventilation system is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

9.4.6.1.7 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The ASW ventilation system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.4.6.1.8 10 CFR 50.63 – Loss of All Alternating Current Power

The ASW ventilation system is required to provide ventilation cooling for the ASW equipment in the ASW pump room needed to bring the unit to Mode 3 following an SBO and maintain ventilation for the 4 hour SBO coping period.

9.4.6.2 System Description

The ASW ventilation system must be in operation when an ASW pump is operating. The requirements for the system temperature design are discussed in Sections 9.4 and 9.4.6.3.6.

Outside air is drawn into an ASW pump room through ducting, passes through the motor area, and is exhausted to the atmosphere through the exhaust fan and related duct system.

The system is designed, built, and installed according to the design classifications given in Table 3.2-3.

As described in Section 9.2.7, each unit is provided with two 100 percent redundant ASW pumps, each of which is installed in a separate watertight compartment to ensure continued operation during combined tsunami and storm wave runup conditions. Proper ventilation of these compartments is ensured by providing each compartment with a separate ventilation system. Each system consists of a PG&E Design Class I coaxial supply and exhaust duct and an exhaust fan. The outside air is drawn into the compartment through the outer space of the coaxial ducts. The air passes through the ASW pump motor area and is exhausted to the atmosphere by the in-line exhaust fan through the inner space of the coaxial exhaust duct as shown in Figure 9.4-5. The intake and exhaust duct discharge points are located above the highest water level resulting from the combined effects of tsunami and storm wave runup (refer to Section 2.4.6.6).

Each exhaust fan starts automatically whenever its associated ASW pump is started. One pump and its associated ventilation system normally operate, with the second set providing system redundancy.

9.4.6.3 Safety Evaluation

9.4.6.3.1 General Design Criterion 2, 1967 – Performance Standards

The ASW ventilation system is located in the Intake Structure (refer to Figure 9.4-5), which is a PG&E Design Class II structure. The Intake Structure is designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Sections 2.4.6.7, 3.4 and 9.2.7.3.1), external missiles (refer to Section 3.3.2.5.2.10), and earthquakes (refer to Sections 3.7 and 3.8.3.2) to protect the PG&E Design Class I ASW ventilation system allowing it to perform its design function.

The ASW ventilation system is seismically designed to perform its safety functions under the effects of earthquakes (refer to Section 3.10.3.33).

9.4.6.3.2 General Design Criterion 3, 1971 – Fire Protection

The ASW ventilation system is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.4.6.3.3 General Design Criterion 11, 1967 – Control Room

The ASW ventilation system is activated whenever the pumps are activated. A start and stop switch is located in the control room for the ASW pumps, which also activates the ASW ventilation. The HSP also provides for remote control of the ASW pumps, and acts as a start and stop switch for the ASW ventilation system.

Indication is provided on the main control board to indicate fan failure while the pump is operating. Annunciators in the control room alert operators to high temperature conditions for the ASW ventilation.

9.4.6.3.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

In each pump room there is a temperature switch provided to monitor temperature within the ASW pump room.

High temperatures exceeding system set points are monitored and alarmed in the control room.

9.4.6.3.5 General Design Criterion 21, 1967 – Single Failure Definition

Redundant components, along with separated Class 1E power sources, give the system the capability of meeting single failure criteria. The redundant ASW pumps are located in separate vaults, are provided with a completely independent ventilation system, and therefore share no common components between trains. Each fan in the system receives power from the same Class 1E bus as its respective ASW pump. Separation has been maintained for the electrical circuits from each Class 1E bus.

9.4.6.3.6 Auxiliary Saltwater Ventilation System Safety Function Requirements

(1) Cooling of PG&E Design Class I Equipment

The system has been designed to maintain the inside air temperature below 104°F with an outdoor ambient design temperature described in Section 9.4. The 1 hp vane axial fans have been sized to provide a minimum air flow of 4000 cfm. The air flow requirements have been based on the ASW pump motor cooling load and the heat transfer through the coaxial duct. No credit is taken for wind cooling the intake structure. The maximum solar load on the ventilating system is negated by the heat losses through the slab floor.

(2) Protection from Missiles

The PG&E Design Class I equipment was reviewed to determine those that could possibly be affected by potential missiles. For postulation missile sources, it was determined that release of a missile would not endanger the ASW ventilation system (refer to Section 3.5.1.2).

(3) Protection from Moderate Energy Pipe Rupture Effects

The PG&E Design Class I ASW ventilation system is designed to be protected against moderate energy pipe rupture effects which may result from equipment failures (refer to Section 3.6.6..3(2)).

Provisions have been taken to protect the ASW ventilation system from the effects of moderate energy pipe failure to the extent necessary to ensure that a safe shutdown condition of the reactor can be accomplished and maintained (refer to Section 3.6). Each ASW pump and its associated ASW ventilation system is housed in a watertight compartment. The ASW ventilation system is a moderate energy system that is physically separate from high energy fluid systems.

(4) Protection from Flooding Effects

The provisions taken to protect ASW ventilation system from the effects of internal flooding to the extent necessary are discussed in Section 3.6.

9.4.6.3.7 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805 The ASW ventilation system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition (refer to Section 9.5.1).

9.4.6.3.8 10 CFR 50.63 – Loss of All Alternating Current Power

Following an SBO, 100 percent cooling of the ASW pump room containing the operating ASW pump is provided by the ASW ventilation system exhaust fan powered from the operating EDG (AAC source) in the blacked-out unit for the 4 hour SBO coping period.

Refer to Section 8.3.1.6 for further discussion of station blackout.

9.4.6.4 Inspection and Testing Requirements

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.

Initial checks of the motors, controls, etc., were made at the time of installation. A verification of air flow and necessary adjustments to design conditions was made.

9.4.6.5 Instrumentation Requirements

Refer to Sections 9.4.6.3.3 and 9.4.6.3.4.

9.4.7 DIESEL GENERATOR COMPARTMENTS VENTILATION SYSTEM

Ventilation of diesel generator compartments is accomplished through the use of the same engine-driven fans that provide cooling air to the diesel generator radiators. The diesel generator cooling air system is described in Section 9.5.5.

The following sections provide information on (a) design bases, (b) system description, (c) safety evaluation, and (d) inspection and testing requirements for diesel generator compartment ventilation systems.

9.4.7.1 Design Bases

9.4.7.1.1 General Design Criterion 2, 1967 - Performance Standards

The EDG compartment ventilation system is designed to withstand the effects of, or is protected against natural phenomena such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.4.7.1.2 General Design Criterion 3, 1971 - Fire Protection

The EDG compartment ventilation system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.4.7.1.3 General Design Criterion 11, 1967 - Control Room

The EDG compartment ventilation system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.4.7.1.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain EDG compartment ventilation system variables within prescribed operating ranges.

9.4.7.1.5 General Design Criterion 21, 1967 - Single Failure Definition

The EDG compartment ventilation system is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event are treated as a single failure.

9.4.7.1.6 Emergency Diesel Generator Compartment Ventilation System Safety Function Requirements

(1) Protection from Missiles

The EDG compartment ventilation system is designed and located to be protected against the effects of missiles which may result from plant equipment failure and from events and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(2) Protection Against High Energy Pipe Rupture Effects

The EDG compartment ventilation system is designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(3) Protection from Moderate Energy Pipe Rupture Effects – Outside Containment

The EDG compartment ventilation system is designed to be protected against the effects of moderate energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(4) Protection from Flooding Effects

The EDG compartment ventilation system is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

9.4.7.1.7 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The EDG compartment ventilation system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.4.7.2 System Description

The ventilation system for each diesel generator compartment has the function of maintaining compartment air temperature within acceptable limits during operation of the diesel generator. The system satisfies the requirements for system temperature design as described in Section 9.4 and satisfies the following design bases:

(1)	Design compartment temperature at diesel engine	120°F
(2)	Heat removal from generator	6,830 Btu/min
(3)	Heat loss from engine surfaces	12,000 Btu/min

The above design bases are for continuous operation at rated diesel generator load. No special provision for heating diesel generator compartments is required since diesel engine generator jacket water and lubricating oil are kept warm by thermostatically controlled heaters during periods when diesel generators are not operating.

Because no significant potential for airborne radioactivity exists in the vicinity of the diesel generator compartments, no filtration or treatment of ventilating air is required.

The ventilation system for each diesel generator compartment is designed, built, and installed according to the design classifications given in Table 3.2-3.

As described in Section 8.3.1.1.6.3.1, each diesel generator is located in a separate compartment in the turbine building. Diesel engine cooling is provided by a closed-loop jacket water system with a radiator and a direct engine-driven fan. Approximately 70 percent of the required radiator cooling air is outside ambient air drawn by the fan from outside the compartment. The remaining 30 percent (approximately 36,000 cfm) of outside ambient air is drawn through duct work, providing ventilation for the diesel generator compartment. The ventilation air flow passes through the compartment, cooling the generator and absorbing surface heat losses from the diesel engine. Other heat loads in the compartment are negligible. In passing through the compartment, the design temperature of the ventilation air is raised by approximately 30°F when the diesel generators are operating continuously at rated load. The ventilation air then passes through the radiator and is exhausted outside the compartment by the direct engine-driven fan. The ventilation system for the diesel generator compartments is shown in Figure 9.4-6.

No credit is taken for wind cooling of the turbine building containing the diesel generator compartments. The maximum solar load was determined by the method in Reference 8 for the exposed outside west wall of the radiator compartment. It was found to add less than 0.1°F to the temperature of the air drawn through the radiator compartment with the ventilation system in operation. Although the air flow through the diesel generator compartment is less, the effect of the maximum solar load on that compartment will also be negligible since it has no outside walls.

The design value for outdoor ambient air temperature for the HVAC systems is 76°F as described in Section 9.4.

The design condition for the diesel generator compartment ventilation is for the long-term recirculation period when the diesel generators are required for continuous emergency power following a LOCA. It is considered highly unlikely that a LOCA and a loss of offsite power would occur simultaneously with an ambient outside air temperature in excess of the design value. It is considered incredible that the two events would occur simultaneously with the maximum recorded outside air temperature at DCPP (refer to Section 2.3.3.2.2).

However, if the outside air temperature is postulated at a peak above the maximum recorded outside air temperature when the diesel generator units are required for emergency power immediately following a LOCA, the thermal capacity and overload capability of the diesel generator units ensure satisfactory performance during the short period until the outside air temperature drops. The temperatures at only a few locations in the diesel generator compartments would rise above 120°F under these conditions, but the units are capable of operating satisfactorily for at least 4 hours, which is longer than any peak is expected to last. The diesel engine itself is not a concern since engines of this type are operated continuously at similar temperatures in hot weather environments. For example, this particular type of engine is employed in locomotive and shipboard duty where continuous operation at temperatures over 120°F is common; e.g., the Great Pacific Southwest Desert and the eastern Atlantic Ocean off the coast of Africa.

9.4.7.3 Safety Evaluation

9.4.7.3.1 General Design Criterion 2, 1967 - Performance Standards

The EDGs are located in the turbine building, which is a PG&E Design Class II structure (refer to Figures 1.2-16 and 1.2-20). This building or applicable portions have been designed not to impact PG&E Design Class I components and associated safety functions (refer to Section 3.7.2.2.2). The turbine building is designed to withstand the effects of winds and tornadoes (refer to Sections 3.3.1.2 and 3.3.2.5.2.8), floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), and earthquakes (refer to Section 3.7.2.2.2) to protect the EDGs, ensuring their design function will be performed.

The EDG units and their associated auxiliary systems, as shown for Unit 1 in Figures 9.5-8 through 9.5-10, and similarly for Unit 2, are installed in separate compartments that are protected from fires, flooding, and external missiles.

An engineering review of the turbine building has shown that during a seismic event, the building may deform, but will not collapse. This analysis is discussed in Section 3.7.2. The design classifications for the walls around each engine generator compartment and for the walls isolating the engine generator compartments from other parts of the turbine building are given in Table 3.2-3.

The EDG compartments are isolated from the turbine building with normally closed fire doors. In the event of a high-energy line break in the turbine building, it is not possible that steam could flow from the turbine building into the engine generator compartments.

9.4.7.3.2 General Design Criterion 3, 1971 - Fire Protection

The EDG areas are designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.4.7.3.3 General Design Criterion 11, 1967 - Control Room

The DCPP Equipment Control Guidelines identify rooms/areas monitored by the area temperature monitoring system, along with their corresponding temperature limits. The ambient air temperatures in these areas are monitored continuously. Air temperatures exceeding the established setpoints are recorded along with the times. An alarm for high temperature is also transmitted to the control room. The cause and effects of high temperature are investigated and corrected in accordance with the DCPP Equipment Control Guidelines.

Refer to Section 8.3.1.1.6.3.4 for additional discussion related to compliance to GDC 11, 1967.

9.4.7.3.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

The EDG compartment ventilation system instrumentation and controls are discussed in Sections 8.3.1.1.6.3.5 and 8.3.1.1.6.5.

9.4.7.3.5 General Design Criterion 21, 1967 - Single Failure Definition

Ventilation of a diesel generator compartment is required only when the diesel generator is operating. This is assumed because ventilation for each compartment is provided by the same direct engine-driven fan that provides cooling air to the radiator. There are no active components in the ventilation system other than the diesel generator itself and the direct engine-driven fan. Refer to Section 8.3.1.1.6.3.8 for additional discussion regarding single failure.

9.4.7.3.6 Emergency Diesel Generator Compartment Ventilation System Safety Function Requirements

(1) <u>Protection from Missiles</u>

The EDG units and their associated auxiliary systems, as shown for Unit 1 in Figures 9.5-8 through 9.5-10, and similarly for Unit 2, are installed in separate compartments that are protected from internal missiles.

Because engine generator units are separated from each other by the concrete walls of the compartments, the units are protected from postulated internal missiles. Any missile created by an explosion within a compartment would remain in that compartment.

The provisions taken to protect the EDG compartment ventilation system from missiles resulting from plant equipment failures and from events and conditions outside the plant are discussed in Sections 3.5 and 8.3.1.4.10.2.

(2) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect the EDG compartment ventilation system from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Sections 3.6 and 8.3.1.4.10.3.

(3) Protection from Moderate Energy Pipe Rupture Effects – Outside Containment

The provisions taken to provide protection of the EDG compartment ventilation system portion located outside containment from the effects of moderate energy pipe failure are discussed in Section 3.6.

(4) Protection from Flooding Effects

The provisions taken to provide protection of the EDG compartment ventilation system from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

Refer to Section 8.3.1.1.6.3.9 for additional information on the possibility of flooding in the turbine building and diesel generator compartments.

9.4.7.3.7 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The EDG compartment ventilation system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition (refer to Section 9.5.1).

The approximate distance between air supply intakes and the nearest exhaust air outlets for the EDG area is 40 feet.

Refer to Section 8.3.1.1.6.3.12 for additional discussion on compliance to 10 CFR Part 50.48(a) and (c).

9.4.7.4 Tests and Inspections

No separate tests or inspections of the ventilation system are required because tests and inspections associated with the diesel generators will also serve the ventilation system. Initial testing of EDGs has verified the adequacy of the ventilating system.

9.4.7.5 Instrumentation Application

The criterion for monitoring air temperature in an EDG compartment is as follows:

Where only one Class 1E redundant train or division is in a given room or area and the ventilation is Class 1E, but is not redundant within the area, one temperature monitor will be used to monitor the area ambient temperature. The temperature monitor will meet Class 1E requirements for supply and separation or have two reliable and redundant non-Class 1E supplies.

Refer to Section 8.3.1.1.6.5 for additional discussion on EDG compartment ventilation system instrumentation.

9.4.8 4.16-kV SWITCHGEAR ROOM

The 4.16-kV switchgear room ventilation system provides cooling to the Class 1E 4.16-kV switchgear rooms and associated CSRs.

9.4.8.1 Design Bases

9.4.8.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I 4.16-kV switchgear room ventilation system is designed to withstand the effects of, or is protected against, natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects. The effects of a tornado on the 4.16-kV switchgear room ventilation system are addressed to ensure that plant safe shutdown can be achieved.

9.4.8.1.2 General Design Criterion 3, 1971 – Fire Protection

The PG&E Design Class I 4.16-kV switchgear room ventilation system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.4.8.1.3 General Design Criterion 11, 1967 – Control Room

The PG&E Design Class I 4.16-kV switchgear room ventilation system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.4.8.1.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain the PG&E Design Class I 4.16-kV switchgear room ventilation system variables within prescribed operating ranges.

9.4.8.1.5 General Design Criterion 21, 1967 – Single Failure Definition

The PG&E Design Class I 4.16-kV switchgear room ventilation system is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event shall be treated as a single failure.

9.4.8.1.6 4.16-kV Switchgear Room Ventilation System Safety Function Requirements

(1) Cooling of PG&E Design Class I Equipment

The PG&E Design Class I 4.16-kV switchgear room ventilation system is designed to provide cooling to the Class 1E 4.16-kV switchgear and associated CSR areas to maintain the temperature within acceptable limits.

(2) Protection from Missiles

The PG&E Design Class I 4.16-kV switchgear room ventilation system is designed to be protected against missiles that might result from plant equipment failure and conditions outside the plant.

(3) Protection Against High Energy Pipe Rupture Effects

The PG&E Design Class I, 4.16-kV switchgear room ventilation system is designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure.

(4) Protection from Moderate Energy Pipe Rupture Effects

The PG&E Design Class I, 4.16-kV switchgear room ventilation system is designed to be protected against the effects of moderate energy pipe failure.

(5) Protection from Flooding Effects

The PG&E Design Class I, 4.16-kV switchgear room ventilation system is designed to be protected from the effects of internal flooding.

9.4.8.2 System Description

The 4.16-kV switchgear room ventilation system is designed to be in operation at all times to provide adequate ventilation for the 4.16-kV switchgear and cabling. This will also provide for the safety and comfort of operating personnel during normal conditions. Other atmospheric conditions, including humidity, atmospheric chemicals, smoke, radiation, and other contaminants are not considered to have significant effect on the switchgear and are not controlled. However, these abnormal conditions may limit operator access to these areas.

The areas served are the switchgear and associated CSRs for the three trains of Class 1E 4.16-kV switchgear. Each ventilation train consists of a supply fan, supply duct, and a vent stack to the turbine building operating floor, as shown in Figure 9.4-7.

Outside air enters the ventilation equipment room through louvers on the north/south wall of the turbine building. Each fan draws air through a roughing filter integral with the fan and supplies the air through the supply duct to the associated 4.16-kV cable spreading and switchgear rooms. The rooms exhaust to the turbine building operating floor without the use of exhaust fans. The trains operate completely independently and are powered independently, receiving power from the same electrical train as the switchgear that they each serve.

The system is designed, built, and installed according to the design classifications given in Table 3.2-3. Applicable codes and standards are listed in Table 9.4-8.

9.4.8.3 Safety Evaluation

9.4.8.3.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I 4.16-kV switchgear room ventilation system equipment is located within the turbine building, as depicted in Figure 9.4-7. The turbine building is a PG&E Design Class II structure and is designed to withstand the effects of floods and tsunamis (refer to Section 3.4), external missiles (refer to Section 3.5), and earthquakes (refer to Section 3.7), ensuring the 4.16-kV switchgear room ventilation system design function will be performed. The area of the turbine building that contains the 4.16-kV switchgear room ventilation system fans and motors is not designed to protect the ventilation system from tornado missiles; however, loss of the 4.16-kV switchgear room ventilation system due to tornado missiles will not impair the ability of the plant to achieve safe shutdown (refer to Section 3.3.2.5.1(4)(j)).

The PG&E Design Class I 4.16-kV switchgear room ventilation system is seismically designed to perform safety functions under the effects of earthquakes (refer to Section 3.10.3.33).

9.4.8.3.2 General Design Criterion 3, 1971 – Fire Protection

The PG&E Design Class I 4.16-kV switchgear room ventilation system is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.4.8.3.3 General Design Criterion 11, 1967 – Control Room

An annunciator alarm is provided at the control room indicating when the temperature within one of the monitored switchgear rooms has exceeded its setpoint or a selected exhaust duct fire damper has closed.

9.4.8.3.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Each fan is turned on and off by a one-stage thermostat located in the associated switchgear room. A switch is provided on each local motor starter panel for manual starting and stopping of the fans and for selecting automatic operation of the fans.

9.4.8.3.5 General Design Criterion 21, 1967 – Single Failure Definition

Functional and electrical independence and physical separation and protection ensure that no single failure of any active component can affect more than one ventilation train.

Each Class 1E 4.16-kV switchgear room and associated CSR is provided with an independent ventilation train. There are no common components. Each ventilation train receives power from the same electrical train as the switchgear that it serves. This provides separate Class 1E electrical supplies to power redundant equipment.

9.4.8.3.6 4.16-kV Switchgear Room Ventilation System Safety Function Requirements

(1) <u>Cooling of PG&E Design Class I Equipment</u>

The 4.16-kV switchgear room ventilation system is designed to be in operation at all times to provide adequate ventilation for the 4.16-kV switchgear and cabling. Each ventilation train provides adequate air flow to maintain the temperature of its associated switchgear and 4.16-kV CSRs within acceptable limits. The requirements for the system temperature design are as described in Section 9.4.

The design basis of the system is to minimize temperature excursions above the temperature assumed in the EQ aging analysis (refer to Section 3.11). The ventilation flow consists of 100 percent outside air. The heat load of the switchgear is such that no tempering of the outside air supply is required. Wind cooling and solar effects are negligible.

(2) <u>Protection from Missiles</u>

The provisions taken to protect the PG&E Design Class I, 4.16-kV switchgear room ventilation system from missiles resulting from plant equipment failures and from events and conditions outside the plant are discussed in Sections 3.5.

Adequate physical separation between the trains has been provided. The system design is such that no unacceptable component failures can occur and adequate physical protection is provided against physical hazards in the areas through which the system is routed.

Main turbine failure is not considered a credible event and failure of the main feedwater pump turbines would not result in missiles that could prevent fulfilment of 4.16-kV switchgear room ventilation system functions.

Refer to Section 9.4.8.3.1 for a discussion on tornado missiles.

(3) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect essential systems and components from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Sections 3.6.

(4) Protection from Moderate Energy Pipe Rupture Effects

The provisions taken to protect essential systems and components from the effects of moderate energy pipe failure are discussed in Section 3.6.

(5) Protection from Flooding Effects

The provisions taken to protect essential systems and components from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

9.4.8.4 Tests and Inspections

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Initial checks of the motors, controls, system balance, etc. were made at the time of installation. This included a verification of the adequacy of the calculated flowrates.

The system will be periodically inspected to ensure that all equipment is functioning properly.

9.4.8.5 Instrumentation Applications

Refer to Sections 9.4.8.3.3 and 9.4.8.3.4 for a discussion on the application of instrumentation and controls used for the 4.16-kV switchgear room ventilation system.

9.4.9 125-VDC AND 480-V SWITCHGEAR AREA

The 125-Vdc and 480-V switchgear area ventilation system serves the following electrical areas:

(1) 125-Vdc switchgear, inverters and battery chargers

(2) 480-V switchgear

(3) HSP

(4) Process control system and plant protection system rack area in the CSR. To provide partial backup cooling whenever the PG&E Design Class II cable spreading room air conditioning (CSR/AC) system is not functional

9.4.9.1 Design Bases

9.4.9.1.1 General Design Criterion 2, 1967 - Performance Standards

The PG&E Design Class I portion of the 125-Vdc and 480-V switchgear area ventilation system is designed to withstand the effects of, or is protected against natural phenomena such as earthquakes, tornadoes, flooding, winds tsunamis and other local site effects.

9.4.9.1.2 General Design Criterion 3, 1971 - Fire Protection

The 125-Vdc and 480-V switchgear area ventilation system and CSR/AC systems are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.4.9.1.3 General Design Criterion 11, 1967 – Control Room

The PG&E Design Class I portion of the 125-Vdc and 480-V switchgear area ventilation system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.4.9.1.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation and controls related to the PG&E Design Class I portion of the 125-Vdc and 480-V switchgear area ventilation system are provided as required to monitor and maintain applicable variables within prescribed operating ranges.

9.4.9.1.5 General Design Criterion 21, 1967 - Single Failure Definition

The PG&E Design Class I portion of the 125-Vdc and 480-V switchgear area ventilation system is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event are treated as a single failure.

9.4.9.1.6 125-Vdc and 480-V Switchgear Area Ventilation System Safety Function Requirements

(1) Cooling of PG&E Design Class I Equipment

The PG&E Design Class I portion of the 125-Vdc and 480-V switchgear area ventilation system is designed to provide cooling to the 480-V switchgear, the 125-Vdc switchgear and inverters, the battery chargers and the HSP areas to maintain the environment within design conditions.

(2) Protection from Missiles

The PG&E Design Class I portion of the 125-Vdc and 480-V switchgear area ventilation system is designed to be protected against missiles that might result from plant equipment failure and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(3) Protection Against High Energy Pipe Rupture Effects

The PG&E Design Class I portion of the 125-Vdc and 480-V switchgear area ventilation system is designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(4) Protection from Moderate Energy Pipe Rupture Effects

The PG&E Design Class I portion of the 125-Vdc and 480-V switchgear area ventilation system is designed to be protected against the effects of moderate energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(5) Protection from Flooding Effects

The PG&E Design Class I portion of the 125-Vdc and 480-V switchgear area ventilation system is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

9.4.9.1.7 10 CFR 50.63 - Loss of All Alternating Current Power

The 125-Vdc and 480-V switchgear area ventilation system is required to provide ventilation cooling for the 480-V switchgear, 125-Vdc switchgear, inverters, and battery chargers to bring the unit to Mode 3 following an SBO, and maintain ventilation for the 4 hour SBO coping period.

9.4.9.1.8 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The Design Class I portion of the 125-Vdc and 480-V switchgear area ventilation system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.4.9.2 System Description

The 125-Vdc and 480-V switchgear area ventilation system provides adequate ventilation for the switchgear. This also provides for the safety and comfort of operating personnel during normal conditions. Other atmospheric conditions, including humidity, atmospheric chemicals, smoke, radiation, and other contaminants are not considered to have significant effects on the switchgear and are not controlled. However, these abnormal conditions may limit operator access to these areas.

The areas served by the PG&E Design Class I 125-Vdc and 480-V switchgear ventilation system are the compartments housing the three redundant trains of 480-V switchgear and dc switchgear, the four redundant trains of inverters, battery chargers, and HSP areas. The system consists of two sets of redundant 100 percent capacity supply and exhaust fans, a common supply and exhaust duct, dampers, air outlets and inlets, and fire dampers as shown in Figure 9.4-8.

The ventilation system is a once-through type with 100 percent outside air supply. In Unit 1 outside air is drawn by one of the two redundant supply fans, discharging through a common supply ductwork and roughing filter and introduced to each area by supply registers. The Unit 2 configuration is the same except that the filter is located at the inlet side of the fans.

Exhaust air from each area is drawn by one of the two redundant exhaust fans and discharged to the atmosphere. The 125-Vdc and 480-V switchgear area ventilation system supply and exhaust fans are powered from the 480-V Class 1E buses (refer to Figures 8.3-6 and 8.3-8).

The system intake louver is located in the intake plenum and the exhaust air is discharged away from the supply air intake.

The system also provides partial backup ventilation for the CSR, which is normally served by PG&E Design Class II ventilation air conditioning systems as shown in Figure 9.4-8. Normally, the CSR is cooled by a PG&E Design Class II air conditioning system (CSR/AC). The backup service in the CSR requires manual alignment of dampers in the interconnecting ductwork.

The PG&E Design Class II CSR/AC system consists of an ACU with chilled water coil that recirculates air from the CSR. One of two, redundant, 100 percent capacity, PG&E Design Class II air-cooled water chillers servicing Unit 1 and Unit 2 provides chilled water to the ACUs, one ACU per CSR, and to a condensing coil (one per unit) The CSR/AC system is interconnected with the 125-Vdc and 480-V switchgear area ventilation (PG&E Design Class I) system to provide partial back up cooling whenever the CSR/AC system is not functional.

The CSR temperature is controlled, when the CSR/AC system is in operation, by modulating the amount of chilled water flowing through the cooling coil in the ACU in response to a room thermostat. The CSR humidity is stabilized by use of a condensing cooling coil in the supply air duct bringing outside air into the room.

The chillers, circulating chilled water pumps, and ACUs are manually started through local control stations.

The system is designed, built, and installed according to the design classifications given in Table 3.2-3. Applicable codes and standards are listed in Table 9.4-8.

9.4.9.3 Safety Evaluation

9.4.9.3.1 General Design Criterion 2, 1967 - Performance Standards

The PG&E Design Class I 125-Vdc and 480-V switchgear area ventilation system equipment is located on or within the auxiliary building, a PG&E Design Class I structure (refer to Figure 1.2-6). Both redundant supply and exhaust fans are located on the roof of the auxiliary building. The supply duct is routed from the supply fans at the roof of the auxiliary building, penetrating the turbine building (a PG&E Design Class II structure) siding, then down along the wall outside the auxiliary building where it then enters the auxiliary building in the dc switchgear area. The exhaust duct is routed alongside the supply duct up to their respective penetrations in the wall of the auxiliary building and the turbine building siding.

The auxiliary building is designed to withstand the effects of winds and tornadoes (refer to Section 3.3), floods and tsunamis (refer to Section 3.4), and earthquakes (refer to Sections 3.7 and 3.8.2.3) to protect the 125-Vdc and 480-V switchgear area ventilation system, ensuring its design function will be performed.

The turbine building, which contains portions of the125-Vdc and 480-V switchgear area ventilation system duct work and support is designed to withstand the effects of floods and tsunamis (refer to Section 3.4), the effects of winds (refer to Section 3.3), and the effects of earthquakes (refer to Sections 3.7 and 3.8.2.3) to protect the 125-Vdc and 480-V switchgear area ventilation system, ensuring its design function will be performed.

Loss of the125-Vdc and 480-V switchgear area ventilation system following a tornado does not compromise the capability to safely shutdown the plant (refer to Section 3.3.2.5.1(4)(h)).

The 125-Vdc and 480-V switchgear area ventilation system is seismically designed to perform its safety functions under the effects of earthquakes (refer to Section 3.10.3.33).

The interconnection between the PG&E Design Class II CSR/AC system and the 125-Vdc and 480-V switchgear area ventilation system (PG&E Design Class I) is made through normally closed manually operated dampers which are part of the PG&E Design Class I system. To avoid transfer of seismic loads between the systems, the intertie is made via a flexible connection.

9.4.9.3.2 General Design Criterion 3, 1971 - Fire Protection

The 125-Vdc and 480-V switchgear area ventilation system and CSR/AC systems are designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1). Ventilation of the dc battery rooms also prevents the accumulation of hydrogen (refer to Section 8.3.2.3.2).

9.4.9.3.3 General Design Criterion 11, 1967 – Control Room

Annunciation is provided in the control room to alarm temperatures in both the 125-Vdc and 480-V switchgear areas and the CSR area.

In the event the control room is lost due to fire or other causes, the switchgear area ventilation system provides ventilation for the HSP.

9.4.9.3.4 General Design Criterion 12, 1967 – Instrumentation and Control Systems

Instrumentation is provided to monitor the 125-Vdc and 480-V switchgear ventilation system components.

A switch in a local control station is used to select either automatic operation or to start each fan manually. Automatic operation is for "operational convenience" and is not a design function credited in any safety analysis.

When the switch is in the automatic mode, the exhaust fan starts after its corresponding supply fan is started. The redundant set of supply and exhaust fans automatically starts if the normally operating set fails.

9.4.9.3.5 General Design Criterion 21, 1967 - Single Failure Definition

The redundant trains of supply and exhaust fans of the 125-Vdc and 480-V switchgear ventilation system are physically separated and powered from separate Class 1E electrical power supplies to ensure that any single failure of an active component will not prevent the ventilation system from supplying the required air flow.

The system design is such that no unacceptable passive component failures can occur and such that physical protection is adequately provided against physical hazards in the areas through which the system is routed.

9.4.9.3.6 125-Vdc and 480-V Switchgear Area Ventilation System Safety Function Requirements

(1) Cooling of PG&E Design Class I Equipment

The requirements for the system temperature design are described in Section 9.4. In addition, the system design indoor temperature is 104°F for the 125-Vdc switchgear and battery chargers, the 480-V switchgear, and the HSP. The ventilation system with only one set of supply and exhaust fans operating was designed to provide adequate air flow in all the areas served by the system to maintain ambient temperature below 104°F with outside ambient temperature as described in Section 9.4, except for the CSR which is also served by a PG&E Design Class II system. The system is designed to provide a relatively constant temperature and humidity within the room to enhance the service life of electronic devices installed in the room.

Indoor design temperature for the process control system and plant protection system rack area in the CSR is 72 ±5°F during normal operation of PG&E Design Class II air conditioning system and 108°F when the backup PG&E Design Class I 480-V switchgear area ventilation system serves that area.

The heat load of all electrical equipment inside the areas served by the system requires no tempering of the air supply. Wind cooling and solar effects are negligible.

(2) Protection from Missiles

The provisions taken to protect the PG&E Design Class I, 125-Vdc and 480-V switchgear area ventilation system from missiles resulting from plant equipment failures and from events and conditions outside the plant are discussed in Sections 3.5.

The system design is such that physical protection is adequately provided against physical hazards in the areas through which the system is routed.

The PG&E Design Class I equipment was reviewed to determine those that could possibly be affected by potential missiles. For postulated missile sources, it was determined that release of a missile would not endanger the 125-Vdc and 480-V switchgear ventilation system (refer to Section 3.5.1.2).

(3) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect the PG&E Design Class I, 125-Vdc and 480-V switchgear area ventilation system from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Sections 3.6.

Protection against the effects resulting from the rupture of HELB piping is provided for the 125-Vdc and 480-V switchgear ventilation system and such that physical protection is adequately provided against physical hazards in the areas through which the system is routed.

(4) Protection from Moderate Energy Pipe Rupture Effects

The provisions taken to provide protection of the PG&E Design Class I, 125-Vdc and 480-V switchgear room ventilation system from the effects of moderate energy pipe failure are discussed in Section 3.6.

(5) Protection from Flooding Effects

The provisions taken to provide protection of the PG&E Design Class I, 125-Vdc and 480-V switchgear room ventilation system from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

Sufficient design provisions are provided (e.g., door curbs, hatch curbs) to limit any flooding to the Elevation 115 foot area due to a chilled water line break of the CSR ACU. Cable penetrations are sealed to prevent carryover of water to other areas. The chilled water system does not have automatic makeup. Therefore, the total water volume of the chilled water system, less than 390 gallons, would cause the water to rise to a level of no more than 0.8 inches within the curbed area. The minimum height of the curbs is 2 inches.

9.4.9.3.7 10 CFR 50.63 - Loss of All Alternating Current Power

Following an SBO, 100 percent cooling for the 480-V switchgear, 125-Vdc switchgear, inverters, and battery chargers is provided by the 125-Vdc and 480-V switchgear area ventilation system powered from the operating EDG (AAC source) in the blacked-out unit for the 4 hour SBO coping period.

Refer to Section 8.3.1.6 for further discussion of station blackout.

9.4.9.3.8 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The 125-Vdc and 480-V switchgear area ventilation system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition (refer to Section 9.5.1).

9.4.9.4 Test and Inspections

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.

Initial checks of the motors, controls, system balance, etc., were made at the time of installation. This included a verification of the adequacy of the calculated flow rates. The system is periodically inspected to ensure that all normally operating equipment is functioning properly. Redundant components are periodically tested to ensure system availability.

9.4.9.5 Instrumentation Applications

Instrumentation used to control and operate the ventilation system is discussed in Sections 9.4.9.3.3 and 9.4.9.3.4.

9.4.10 POST-ACCIDENT SAMPLE ROOM

The post-accident sample room complex is located in the auxiliary building; however, it is provided with its own HVAC system. The system may also be referred to as the post-LOCA sample room HVAC system in other plant documentation.

9.4.10.1 Design Bases

9.4.10.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class II post-accident sample room HVAC system is designed to withstand the effects of earthquakes.

9.4.10.1.2 General Design Criterion 3, 1971 – Fire Protection

The PG&E Design Class II post-accident sample room HVAC system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.4.10.1.3 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The PG&E Design Class II post-accident sample room HVAC system has been designed based on 10 CFR Part 20 requirements for maintaining control over the plant’s radioactive gaseous effluents. Appropriate holdup capacity is provided for retention of gaseous effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment.

9.4.10.1.4 NUREG-0737 (Items II.B.3.9.b and II.B.3.11.b), November 1980 – Clarification of TMI Action Plan Requirements

Item II.B.3.9.b – Post-Accident Sampling Capability: The PG&E Design Class II post-accident sample room HVAC system is designed to restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This is supported through the use of the system design, which will control the presence of airborne radioactivity.

Item II.B.3.11.b – Post-Accident Sampling Capability: The exhaust air from the panels and hoods is filtered with charcoal adsorbers and HEPA filters.

9.4.10.2 System Description

The post-accident sample room HVAC system is an independent system as shown in Figure 9.4-10.

The post-accident sample room HVAC system is designed to provide ventilation, heating, and cooling to the sample room for plant personnel comfort during normal plant operation.

During normal plant operation, a ventilation fan will deliver 300 cfm of outside air to the sample room complex.

Following an accident, the system provides protection for plant personnel from radiological contaminants.

The post-accident sample room HVAC system is a PG&E Design Class II system. The design classification of the post-accident sample room HVAC system is given in Table 3.2-3.

Redundant supply and exhaust fans and filters, the seismic design of the system, and the shielding provided for in the sample room complex will provide the required personnel protection and equipment ventilation following an accident.

The air conditioning portion of the HVAC system will maintain the sample room at a temperature ranging between 65 and 90°F during normal plant operation.

The fans are manually initiated from the PASS ventilation control panel.

9.4.10.3 Safety Evaluation

9.4.10.3.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class II post-accident sample room HVAC system equipment is located within the auxiliary building, a PG&E Design Class I structure (refer to Figure 1.2-6 and Section 3.7.2.2.1).

The design class requirements for the system, except the air conditioner, are supplemented with more stringent seismic criteria than those required for a PG&E Design Class II system; the system has been evaluated for the HE. The calculated stresses are within the allowables for PG&E Design Class I systems.

The air conditioner is so supported that it will remain in place after a HE.

9.4.10.3.2 General Design Criterion 3, 1971 – Fire Protection

The PG&E Design Class II post-accident sample room HVAC system is designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.4.10.3.3 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The PG&E Design Class II post-accident sample room HVAC system has been designed with appropriate holdup capacity for the retention of radioactive gaseous effluents. Exhaust air is charcoal- and HEPA-filtered before being discharged to the atmosphere.

9.4.10.3.4 NUREG-0737 (Items II.B.3.9.b and II.B.3.11.b), November 1980 – Clarification of TMI Action Plan Requirements

Item II.B.3.9.b – Post-Accident Sampling Capability: Following an accident, one of two 100 percent capacity redundant pressurization fans will deliver 1000 cfm of charcoal-filtered outside air to the complex and maintain it at a positive pressure with respect to surrounding plant areas.

The sample panel and the sample station hood located in the complex are maintained at a negative pressure when an exhaust fan is in operation. One of the two 100 percent capacity, redundant, manually initiated, ventilation exhaust fans will discharge 700 cfm exhaust air to atmosphere.

The post-accident sample room HVAC system supports DCPP sampling contingency plan requirements (Reference 23), and continues to meet the above original design basis requirements.

Item II.B.3.11.b – Post-Accident Sampling Capability: The exhaust air from the panels and hoods is manifolded together, charcoal-filtered, HEPA-filtered, and discharged by the exhaust fan to the atmosphere.

The post-accident sample room HVAC system supports DCPP sampling contingency plan requirements (Reference 23), and continues to meet the above original design basis requirements.

9.4.10.4 Tests and Inspections

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

The initial checks of the motors, controls, system balance, etc. were made at the time of installation. The verification of the calculated flowrates was also accomplished at this time.

The system is periodically inspected and tested to ensure that all equipment is functioning properly.

9.4.10.5 Instrumentation Applications

Instrumentation provides local indication of the system's operating parameters; i.e., filter differential pressure (indicating usage and cleanliness), and subsystem temperatures and pressures. An interlock is also provided to prevent climate control unit heater operation under a low supply air flow condition.

In addition, an area radiation monitor with local annunciation is installed in the post-accident sample room to warn personnel of high or increasing radiation. High radiation will alarm at the main annunciator (refer to Section 11.4).

9.4.11 TECHNICAL SUPPORT CENTER

The basic function of the TSC HVAC system is to provide protection for personnel working in the TSC from radiological contaminants and to provide HVAC for working areas and equipment.

9.4.11.1 Design Bases

9.4.11.1.1 General Design Criterion 4, 1967 - Sharing of Systems

The TSC HVAC system or components are not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.4.11.1.2 10 CFR 50.47 - Emergency Plans

The TSC HVAC system is adequate to support the use of the TSC for emergency response.

9.4.11.1.3 NUREG-0737 (Items II.B.2 and III.A.1.2), November 1980 - Clarification of TMI Action Plan Requirements

II.B.2 – Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-Accident Operations: Adequate access to the TSC is provided by design changes, increased permanent or temporary shielding, or post-accident procedural controls.

III.A.1.2 – Upgrade Emergency Support Facilities: NUREG-0737, Supplement 1, January 1983 provides the requirements for III.A.1.2 as follows:

> Section 8.2.1.e - The TSC is environmentally controlled to provide room air temperature, humidity, and cleanliness appropriate for personnel and equipment.
>
> Section 8.2.1.f - The TSC is provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem TEDE for the duration of the accident.

9.4.11.2 System Description

The TSC is provided with its own PG&E Design Class II HVAC system that is schematically shown in Figure 9.4-10. The entire system is manually initiated and is designed to maintain the occupied areas of the TSC at a temperature below 85ºF. During normal operation, all the makeup air and recirculated air passes through a roughing filter, a HEPA filter, and the ACU. Makeup air in the normal operation mode is supplied via the outside air intake by a single makeup air fan. The TSC HVAC system has the capability to manually isolate the area from outside air and to recirculate air via the air conditioning system.

In the radiological accident mode of operation, the TSC HVAC system makeup air is supplied by the CRPS in order to maintain the TSC area at a minimum of +1/8 inch water gauge pressure. Penetrations into the TSC are equipped with penetration seals and floor drain traps with make-up water supplies to prevent exfiltration of the positive pressure from within the TSC through these paths. The pressurization air, and a portion of the recirculated air, is passed through a filter bank containing a HEPA and charcoal filters for cleanup purposes and supplied to the general area rooms along with the majority of the recirculated air. Exhaust air leaves the TSC by exfiltration.

The TSC HVAC system is fed from a non-Class 1E power source, although it has the capability to be supplied power from a Class 1E bus. The system is not seismically qualified but the ducting, duct supports, and equipment supports are designed and analyzed to seismic requirements. The ducting and components associated with the TSC pressurization air supply are PG&E Design Class I up to and including the manual damper upstream of the redundant duct heaters associated with the TSC filter bank (Reference 8 of Section 3.2).

The duct heaters maintain the relative humidity of the pressurization air below 70 percent.

9.4.11.3 Safety Evaluation

9.4.11.3.1 General Design Criterion 4, 1967 - Sharing of Systems

The TSC HVAC system is common to Unit 1 and Unit 2 and therefore requires sharing of SSCs between units. The TSC HVAC system serves no safety functions. The TSC HVAC system is designed to provide adequate HVAC for working areas and equipment within the TSC. In addition, the CRPS is shared between the control room and the TSC. Sharing of the CRPS by the control room and the TSC is addressed in Section 9.4.1.3.3.

9.4.11.3.2 10 CFR 50.47 - Emergency Plans

The TSC HVAC system meets applicable requirements and is maintained in support of emergency response (refer to Sections 9.4.11.1.3 and 9.4.11.3.3).

9.4.11.3.3 NUREG-0737 (Items II.B.2 and III.A.1.2), November 1980- Clarification of TMI Action Plan Requirements

Item II.B.2 - Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-Accident Operations: The acceptance criteria for the TSC dose is based on 10 CFR 50.67, in which dose to an operator in the TSC shall not exceed 5 rem TEDE for the duration of the accident. The adequacy of TSC shielding is evaluated for post-LOCA conditions in Sections 12.1.2.7 and 15.5.

Item III.A.1.2 - Upgrade Emergency Support Facilities: NUREG-0737, Supplement 1, January 1983 provides the requirements for III.A.1.2 as follows: Section 8.2.1.e - During normal operation, all the TSC HVAC makeup air and recirculated air passes through the climate control units. The climate control units are designed to maintain the occupied areas of the TSC below 85°F, and provide for humidity and cleanliness appropriate for personnel and equipment.

The TSC HVAC system is a PG&E Design Class II system. The pressurization air for the system is supplied by the designated PG&E Design Class I CRPS. The radiological accident mode of operation maintains the TSC area at a positive pressure. The relative humidity of the pressurization air is maintained below 70 percent. The air cleanup portion of the system is equipped with redundant fans and heaters and a power supply that may be manually switched over to a Class 1E bus source. Post-accident dose in the TSC is discussed in Section 15.5.

Section 8.2.1.f - TSC area radiation monitoring instruments provide continuous indication of the general area ambient radiation levels and provide local alarm annunciation in various areas and work spaces. TSC ventilation air monitoring instruments provide continuous sampling of the TSC HVAC return air ducts for detection of airborne radiation and provide for alarm annunciation in the TSC computation center.

The acceptance criteria for the TSC dose is based on Section 8.2.1 (f) of NUREG-0737, Supplement 1, Regulatory Guide 1.183, July 2000, Section 1.2.1, and 10 CFR 50.67. The dose to an operator in the TSC should not exceed 5 rem TEDE for the duration of the accident. TSC shielding and the ventilation system prevent post-accident doses inside the TSC from exceeding 5 rem TEDE for the duration of the accident. Refer to Section 9.4.11.2 for a description of the TSC ventilation system. The adequacy of TSC shielding and ventilation is evaluated for post-LOCA conditions in Section 15.5.

9.4.11.4 Tests and Inspections

Initial checks of the motors, controls, system balance, etc. were made at the time of installation. The system is periodically inspected to ensure that all equipment is functioning properly.

9.4.11.5 Instrumentation Requirements

TSC area radiation monitoring instruments provide continuous indication of the general area ambient radiation levels and provide local alarm annunciation in various areas and work spaces.

TSC ventilation air monitoring instruments provide continuous sampling of the TSC HVAC return air ducts for detection of airborne radioactivity and provide for alarm annunciation in the TSC computation center.

Instrumentation related to post-accident operation includes indication and alarm on low pressurization flow and an alarm for high temperature in the charcoal filter section of the charcoal/HEPA filter bank.

9.4.12 CONTAINMENT PENETRATION AREA GE/GW

The containment penetration area GE/GW ventilation system has the function of maintaining the ambient temperature and pressure of the GE/GW area within acceptable limits during normal operations. The GE/GW area ventilation system is a PG&E Design Class II draw-through type ventilation system. The system consists of both PG&E Design Class I and II components.

The following GE/GW area ventilation system components described in this section are PG&E Design Class:

(1) System isolation dampers, duct, and duct supports from the upstream, isolation damper to the plant vent connection.

(2) Isolation damper actuator, solenoid and control switches. Instrumentation and control components are classified as PG&E Instrument Design Class ID (refer to Section 3.2.2.5).

9.4.12.1 Design Bases

9.4.12.1.1 General Design Criterion 2, 1967 – Performance Standards

The PG&E Design Class I portion of the containment penetration GE/GW area ventilation system is designed to withstand the effects of, or is protected against natural phenomena such as earthquakes, tornadoes, flooding, winds, tsunamis, and other local site effects. In addition, the PG&E Design Class II portion of the containment penetration GE/GW area ventilation system is designed to seismic loads to ensure compliance with the Seismically Induced Systems Interaction Program (refer to Section 3.7.3.13).

9.4.12.1.2 General Design Criterion 17, 1967 – Monitoring Radioactivity Releases

Radiation monitoring instrumentation is provided to monitor radioactive releases in the GE/GW ventilation exhaust via the plant vent.

9.4.12.2 System Description

The GE/GW area ventilation system was designed to:

(1) Provide means of monitoring GE/GW exhaust for airborne radioactivity during normal plant operation by exhausting into the plant vent

(2) Maintain the ambient temperature at maximum average temperature of 104°F during normal plant operation to support environmental requirements of PG&E Design Class I equipment (refer to Section 3.11)

(3) Maintain the GE/GW area at a slight negative pressure with respect to outdoors

(4) Maintain steam relief flow path during HELB accident

(5) Maintain pressure boundary integrity of the plant vent at the interface between the two systems.

The design classification of the containment penetration area GE/GW ventilation system is given in Table 3.2-3. Applicable codes and standards are listed in Table 9.4-8.

One of the two full capacity exhaust fans runs to exhaust air from the GE/GW areas. This running fan maintains the GE/GW areas at a slight negative pressure relative to the outdoors.

The ventilation flow consists of 100 percent outside air drawn through the intake louvers located on the wall of GW area. Four (4) recirculation fans move the entering air to the areas farthest from the intake louver.

The system is shown in Figure 9.4-11.

The ventilation function performed by the containment penetration area GE/GW ventilation system is not a safety-related function, and its complete failure has no safety implication. Two redundant 100 percent capacity fans are provided so that failure of one fan will not result in loss of ventilation for these areas during the normal plant operation.

The containment penetration area GE/GW ventilation system exhausts into the plant vent. Isolation of the Design Class I plant vent from the Design Class II portion of the GE/GW ventilation system is performed by redundant Design Class I isolation dampers. Instrument Class ID instrumentation and controls are seismically designed to ensure damper operation (refer to Section 3.10.2). The plant vent pressure boundary is further maintained by the Design Class I duct and supports that perform the tie-in between the plant vent and the GE/GW ventilation system.

9.4.12.3 Safety Evaluation

9.4.12.3.1 General Design Criterion 2, 1967 – Performance Standards

The containment penetration area GE/GW ventilation system is designed to perform its safety function under the effects of earthquakes (refer to Section 3.10.3.33), winds and tornadoes (refer to Section 3.3), and external missiles (refer to Section 3.5). The GE/GW ventilation system may be vulnerable to tornado effects but does not need to be operational while the plant is brought to a safe operational status.

9.4.12.3.2 General Design Criterion 17, 1967 – Monitoring Radioactivity Releases

The ventilation air flows to the enclosed annular gap on Elevation 140 foot, into the duct connected to the exhaust fan and discharged into the plant vent, where it is monitored for radioactivity.

For a description of the radiological monitoring system refer to Section 11.4.2.1.2.

9.4.12.4 Tests and Inspections

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

Initial checks of the fan housings, bearings, motors, bolts, controls, etc., are made at the time of installation. A system air balance test and adjustment to design conditions are conducted.

The system will be periodically inspected to ensure that all equipment is functioning properly.

9.4.12.5 Instrumentation Applications

Each exhaust fan is manually turned on and off by a switch on the local control panel mounted near the exhaust fans. Switchover to standby unit is done manually. The recirculation fans are initiated by thermostat located in the area where recirculation fan is located. The isolation dampers are operated by their associated pressure differential switches. The shut-off dampers are interlocked to their corresponding exhaust fans "on-off" switches.

9.4.13 REFERENCES

1. Deleted in Revision 8.

2. Deleted in Revision 8.

3. Deleted in Revision 10.

4. Deleted in Revision 10.

5. Recommended Outdoor Design Temperatures, Southern California, Arizona, and Nevada, Third Edition, Southern California Chapter, American Society of Heating, Refrigeration and Air-Conditioning Engineers, Inc., March 1964.

6. Deleted in Revision 22.

7. Deleted in Revision 10.

8. Deleted in Revision 22.

9. R. R. Bellamy, Elemental Iodine and Methyl Iodine Adsorption on Activated Charcoal at Low Concentrations, 13th Air Cleaning Conference, 1974.

10. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended.

11. Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants, USAEC, Revision 0, June 1973.

12. G. W. Kielholts, Filters, Sorbents, and Air Cleaning Systems as Engineered Safeguards in Nuclear Installations, ORNL-NSIC-13, October 1966.

13. NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release," James Wing, USNRC-ONRR, June 1979.

14. Deleted in Revision 10.

15. Deleted in Revision 10.

16. Deleted in Revision 24.

17. Deleted in Revision 8.

18. Deleted in Revision 10.

19. Deleted in Revision 10.

20. Deleted in Revision 10.

21. Branch Technical Position 7-14, Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems, Revision 5, March 2007.

22. Branch Technical Position 7-18, Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems, Revision 5, March 2007.

23. License Amendment Nos. 149 (DPR-80) and 149 (DPR-82), "Issuance of Amendment Re: Elimination of Post-Accident Sampling Requirements," USNRC, July 13, 2001.

24. Regulatory Guide 1.52, Revision 2 - Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants.

9.4.14 REFERENCE DRAWINGS

Figures representing controlled engineering drawings are incorporated by reference and are identified in Table 1.6-1. The contents of the drawings are controlled by DCPP procedures.

9.5 OTHER AUXILIARY SYSTEMS

This section provides information on plant auxiliary systems that do not otherwise apply to the preceding sections of this chapter.

9.5.1 FIRE PROTECTION

The fire protection program is based on the NRC requirements and guidelines, Nuclear Electric Insurance Limited (NEIL) Property Loss Prevention Standards and related industry standards. With regard to NRC criteria, the fire protection program meets the requirements of 10 CFR 50.48(c), which endorses, with exceptions, the National Fire Protection Association's (NFPA) 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition. Diablo Canyon Power Plant (DCPP) Units 1 and 2 has further used the guidance of NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)" as endorsed by Regulatory Guide 1.205," Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants."

Adoption of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition in accordance with 10 CFR 50.48(c) serves as the method of satisfying 10 CFR 50.48(a) and General Design Criterion 3. Prior to adoption of NFPA 805, General Design Criterion 3, "Fire Protection" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," was followed in the design of safety and non-safety related structures, systems and components, as required by 10 CFR 50.48(a).

NFPA 805 does not supersede the requirements of GDC3, 10 CFR 50.48(a), or 10 CFR 50.48(f). Those regulatory requirements continue to apply. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements are met may be different than under 10 CFR 50.48(b). Specifically, whereas GDC 3 refers to SSCs important to safety, NFPA 805 identifies fire protection systems and features required to meet the Chapter 1 performance criteria through the methodology in Chapter 4 of NFPA 805. Also, under NFPA 805, the 10 CFR 50.48(a)(2)(iii) requirement to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is satisfied by meeting the performance criteria in Section 1.5.1 of NFPA 805.

A Safety Evaluation was issued on April 14th, 2016 by the NRC, that transitioned the existing fire protection program to a risk-informed, performance-based program based on NFPA 805, in accordance with 10 CFR 50.48(c).

9.5.1.1 Design Bases

Calculation FP-805 demonstrates how the Diablo Canyon Power Plant Units 1 and 2 meet the requirements of 10 CFR 50.48(a) and 10 CFR 50.48(c). This design basis document satisfies the methodology and documentation requirements of NFPA 805 (2001 Edition), Section 2.7.1.2.

9.5.1.1.1 Defense-in-Depth

The fire protection program is focused on protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations. The fire protection program is based on the concept of defense-in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:

(1) Preventing fires from starting,
(2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage,
(3) Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

9.5.1.1.2 NFPA 805 Performance Criteria

The design basis for the fire protection program is based on the following nuclear safety and radiological release performance criteria contained in Section 1.5 of NFPA 805:

- Nuclear Safety Performance Criteria. Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met:

  (a) Reactivity Control. Reactivity control shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting shall occur rapidly enough such that fuel design limits are not exceeded.

  (b) Inventory and Pressure Control. With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling coolant level such that subcooling is maintained such that fuel clad damage as a result of a fire is prevented for a PWR.

  (c) Decay Heat Removal. Decay heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.

(d) Vital Auxiliaries. Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.

(e) Process Monitoring. Process monitoring shall be capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have been achieved and are being maintained.

- Radioactive Release Performance Criteria. Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR Part 20 Limits.

Chapter 2 of NFPA 805 establishes the process for demonstrating compliance with NFPA 805.

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features.

Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the nuclear safety performance criteria outlined above. The methodology shall be permitted to be either deterministic or performance-based. Deterministic requirements shall be “deemed to satisfy” the performance criteria, defense-in-depth, and safety margin and require no further engineering analysis. Once a determination has been made that a fire protection system or feature is required to achieve the nuclear safety performance criteria of Section 1.5, its design and qualification shall meet the applicable requirement of Chapter 3.

9.5.1.1.3 Codes of Record

The codes and standards used for the design and installation of plant fire protection systems are contained in Appendix C to Calculation FP-805.

9.5.1.2 System Description

9.5.1.2.1 Required Systems

Nuclear Safety Capability Systems, Equipment, and Cables

Section 2.4.2 of NFPA 805 defines the methodology for performing the nuclear safety capability assessment. The systems, equipment, and cables required for the nuclear safety capability assessment are contained in Appendix F to Calculation FP-805.

Fire Protection Systems and Features

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. Compliance with Chapter 3 is documented in Appendix D to Calculation FP-805.

Chapter 4 of NFPA 805 establishes the methodology and criteria to determine the fire protection systems and features required to achieve the nuclear safety performance criteria of Section 1.5 of NFPA 805. These fire protection systems and features shall meet the applicable requirements of NFPA 805 Chapter 3. These fire protection systems and features are documented in Appendix F to Calculation FP-805.

Radioactive Release

The locations of structures, systems, and components relied upon to meet the radioactive release criteria are contained in Section 9 of Calculation FP-805.

9.5.1.2.2 Definition of "Power Block" Structures

Where used in NFPA 805 Chapter 3, the terms "Power Block" and "Plant" refer to structures that have equipment required for nuclear plant operations. For the purposes of establishing the structures included in the fire protection program in accordance with 10 CFR 50.48(c) and NFPA 805, the plant structures considered to be part of the 'power block' are contained in Appendix B to Calculation FP-805.

9.5.1.3 Safety Evaluation

Appendix F of Calculation FP-805 documents the achievement of the nuclear safety and radioactive release performance criteria of NFPA 805 as required by 10 CFR 50.48(c). This document fulfills the requirements of Section 2.7.1.2, "Fire Protection Program Design Basis Document," of NFPA 805. The document contains or identifies the document location of the following:

- Identification of significant fire hazards in the fire area. This is based on NFPA 805 approach to analyze the plant from an ignition source and fuel package perspective.
- Summary of the Nuclear Safety Capability Assessment (at power and non-power) compliance strategies.
    - Deterministic compliance strategies
    - Performance-based compliance strategies (including defense-in-depth and safety margin)
- Summary of the Non-Power Operations Modes compliance strategies.
- Summary of the Radioactive Release compliance strategies.
- Summary of the Fire Probabilistic Risk Assessments.
- Key analysis assumptions to be included in the NFPA 805 monitoring program.

9.5.1.4 Fire Protection Program Documentation, Configuration Control, and Quality Assurance

In accordance with Chapter 3 of NFPA 805, a fire protection plan documented in Program Directive OM8 defines the management policy and program direction and defines the responsibilities of those individuals responsible for the plan's implementation. OM8:

- Designates the senior management position with immediate authority and responsibility for the fire protection program.
- Designates a position responsible for the daily administration and coordination of the fire protection program and its implementation.
- Defines the fire protection interfaces with other organizations and assigns responsibilities for the coordination of activities. In addition, OM8 identifies the various plant positions having the authority for implementing the various areas of the fire protection program.
- Identifies the appropriate authority having jurisdiction for the various areas of the fire protection program.
- Identifies the procedures established for the implementation of the fire protection program, including the post-transition change process and the fire protection monitoring program.
- Identifies the qualifications required for various fire protection program personnel.
- Identifies the quality requirements of the fire protection program.

Detailed compliance with the programmatic requirements of Chapters 2 and 3 of NFPA 805 are contained in Section 9 and Appendix D of Calculation FP-805.

9.5.2 COMMUNICATIONS SYSTEMS

The communications systems include internal (intra-plant) and external communications designed to provide convenient and effective operational communications among various plant locations and between the plant and locations external to the plant.

The communications systems are PG&E Design Class II.

The communications systems also satisfy Emergency Plan and Security Plan requirements, as discussed in Sections 13.3 and 13.7, respectively.

9.5.2.1 Design Bases

9.5.2.1.1 General Design Criterion 3, 1971 – Fire Protection

The communications systems are designed and located to minimize, consistent with other safety requirements, the probability and effects of fires and explosions.

9.5.2.1.2 General Design Criterion 4, 1967 – Sharing of Systems

The communications systems are not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.5.2.1.3 General Design Criterion 11, 1967 – Control Room

The communications systems are designed to support actions to maintain and control the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other cause.

9.5.2.1.4 Communications Systems Safety Function Requirements

(1) <u>Effective Communications</u>

The communications systems are capable of providing effective intra-plant and plant-to-offsite communications during all modes of operation, including accident conditions.

9.5.2.2 System Description

9.5.2.2.1 Intra-Plant System

The communications systems are designed to ensure continuous intra-plant and plant-to-offsite operation by the use of primary, secondary, and tertiary routings as follows:

A direct dial company telephone system is the primary communications facility within the plant. It has conference call features consisting of emergency conference and regular conference circuits. The emergency conference circuit and regular conference circuits will handle a sufficient number of parties. There is a radio paging system for paging plant personnel within the plant and in surrounding communities. Fire alarms may be dialed from any telephone in the plant, and the caller verbally identifies the location of the fire to the control room operator. Response to this alarm is covered by procedures. Where background noise is high, telephones designed especially for use in high noise level areas are provided.

The plant radio system is the secondary communications facility within the plant and operates between the radio control console in the control room, the FHB, and containment building areas.

A manually initiated emergency signal is the tertiary communications facility. It is operated from the main control room control console.

Response to any of the signals mentioned will be governed by procedures.

9.5.2.2.2 Plant-to-Offsite System

Two communications links are utilized between DCPP and the PG&E San Francisco general office (SFGO).

The primary communications link between the plant and the PG&E SFGO is the PG&E West Valley Microwave System. This system transmits administrative and control voice communications, system load dispatch teletype data links, and 500-kV protective relaying tone channels. The system is based on digital equipment and is provided with battery power for emergency operation. The system has dual transmitters/receivers in the event one does not operate. Refer to Figure 9.5-5 for system routing.

The secondary communications link between the plant and the PG&E SFGO is on common carrier facilities. This system transmits simultaneously with the West Valley system all information except 500-kV protective relaying. It is an all-fiber-optic network. Refer to Figure 9.5-6 for system routing.

The tertiary communications link between the plant and the PG&E enterprise network operations center (ENOC) is the public switched telephone network lines that carry voice and the protective relaying circuit tones.

9.5.2.3 Safety Evaluation

9.5.2.3.1 General Design Criterion 3, 1971 – Fire Protection

The communications systems are designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.5.2.3.2 General Design Criterion 4, 1967 – Sharing of Systems

The communications systems are shared between DCPP Unit 1 and Unit 2. The communications systems are PG&E Design Class II. Because the communications systems are PG&E Design Class II, a failure of the system in one unit will not impair safety in the other unit.

The communications room power distribution panels in Unit 1 and Unit 2 are normally fed from their respective Class 1E 480-V bus. Refer to Section 8.3.1.1.4.3.3 for sharing of power sources.

9.5.2.3.3 General Design Criterion 11, 1967 – Control Room

Communications systems are provided to assist operators in maintaining the safe operation of the facility during normal operation, shutdown, and accident conditions. The HSP for each unit is equipped with a communications panel console containing a radio set and a plant telephone.

9.5.2.3.4 Communications Systems Safety Function Requirements

(1) <u>Effective Communications</u>

The communications systems include internal (intra-plant) and external communications designed to provide convenient and effective operational communications among various plant locations and between the plant and locations external to the plant.

All intra-plant systems and all plant-to-offsite communications utilize tertiary backup and multiple power sources, therefore preventing the failure of individual components from causing a discontinuity of communications (refer to Sections 9.5.2.2.1 and 9.5.2.2.2).

Emergency response communications processes and equipment are administratively controlled in accordance with the Emergency Plan (refer to Section 13.3).

Security based communications processes and equipment are administratively controlled in accordance with the Security Plan (refer to Section 13.7).

9.5.2.4 Tests and Inspections

Routine maintenance procedures require testing and lineup of microwave radio multiplex and battery equipment to take place on a regularly scheduled basis. Tests were conducted to ensure that the plant radios do not affect the instrumentation systems for the reactor protection system and ESFs for Unit 1. Those plant areas that were determined to be affected were made radio exclusion zones. Since plant similarity exists for both Unit 1 and Unit 2, plant modifications identified by these tests for Unit 1 were also implemented in Unit 2. All communications equipment serving the plant are continuously operating, thus providing operational quality checks.

9.5.2.5 Instrumentation Applications

Every critical component of the communications systems, such as the microwave radio system, the 48-V batteries, and the associated equipment, has major functions alarmed locally and at the PG&E enterprise network operations center (ENOC).

9.5.3 LIGHTING SYSTEMS

9.5.3.1 Design Bases

9.5.3.1.1 General Design Criterion 3, 1971 – Fire Protection

The lighting systems are designed and located to minimize, consistent with other safety requirements, the probability and effects of fires and explosions.

9.5.3.1.2 Lighting Systems Safety Function Requirements

(1) Normal and Emergency Lighting

Adequate lighting is provided during all plant operating conditions.

(2) Seismic Requirements

The BOLs that are provided to illuminate access and egress routes for operator actions necessary for safe shutdown are designed to withstand the effects of earthquakes.

9.5.3.2 System Description

Normal lighting is operated at 208Y/120-Vac, three-phase on a four-wire solidly grounded system, supplied from the 480-V system through dry type, delta-wye connected three-phase transformers.

The dc emergency lighting is supplied at 125-Vdc from the non-Class 1E station batteries. The dc emergency lighting fixtures are located principally in electrical equipment rooms, stairways, exits and entrances, corridors, passageways, and at lower levels in all other areas.

The ac emergency lighting is supplied from two of the three Class 1E 480-V buses through dry type, single-phase transformers, and is sized to provide a maximum load of 112-kW for Unit 1 and 100-kW for Unit 2. The ac emergency lighting fixtures are located throughout the plant to provide minimum lighting.

The ac emergency lighting in the pipe rack area is powered by an uninterruptible power supply (UPS). This UPS is powered by non-Class 1E power and has an eight hour rated battery.

The ac emergency lighting circuits are routed in separate conduits from the normal ac lighting on the secondary transformer sides to panels and fixtures. On the primary side, the ac power from the Class 1E 480-V buses is run in separate conduits or in respective Class 1E routes. The 208Y/120-Vac circuits are routed in normal power conduits. Lighting circuits for dc emergency lighting are run in their own conduit.

BOLs for areas containing PG&E Design Class I equipment or to enable operator action to meet NFPA 805 safe shutdown requirements are seismically qualified.

After the diesel generators start and the single-phase ac emergency transformers receive power, the dc emergency lights are automatically turned off. The average period of operation of the dc lights is 18 seconds, (approximately 13 seconds for diesel generator loading and 5 seconds for time delay of contactor pick-up in the emergency dc lighting panel). DC lights in the FHB at all elevations will be on for 10 minutes. Safe shutdown equipment areas and various access routes thereto are provided with BOLs or UPS-powered lights capable of providing 8 hours of illumination if ac power to the BOLs or UPS is lost. The batteries are continuously charged with a built-in charger.

Lighting in the containment structure and radiation areas of the SFP area and part of Area K of the auxiliary building is provided by incandescent light fixtures. In the FHB on the 140-foot elevation in the SFP and machine shop areas, the light fixtures are high intensity discharge pulse start metal halide lights.

Security based lighting is administratively controlled in accordance with the Security Plan (refer to Section 13.7).

9.5.3.3 Safety Evaluation

9.5.3.3.1 General Design Criterion 3, 1971 – Fire Protection

The lighting systems are designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

9.5.3.3.2 Lighting Systems Safety Function Requirements

(1) <u>Normal and Emergency Lighting</u>

Normal lighting, dc emergency lighting, ac emergency lighting, and BOLs are provided to ensure that adequate lighting is available for all plant operating conditions, including transients and accident conditions (refer to Section 9.5.3.2).

(2) <u>Seismic Requirements</u>

BOLs that are necessary to perform post-seismic operator actions or to illuminate access/egress routes to the operator action location are seismically qualified. However, those units that are not necessary to perform post-seismic operator actions or to illuminate access/egress routes to the operator action location are exempt from the requirement.

The safety function of the BOLs is to provide eight (8) hours of lighting if normal ac power is lost (for BOLs that energize on loss of normal ac power) or Class 1E ac power is lost (for BOLs that energize on loss of Class 1E ac power.

9.5.4 DIESEL GENERATOR FUEL OIL STORAGE AND TRANSFER SYSTEM

The diesel generator fuel oil system, shown in Figures 3.2-21, 9.5-8, and 9.5-9, maintains adequate storage of diesel fuel oil and supplies it to the six EDGs. The following subsections provide information on (a) design bases, (b) system description, and (c) safety evaluation for the system.

9.5.4.1 Design Bases

9.5.4.1.1 General Design Criterion 2, 1967 - Performance Standards

The EDG fuel oil storage and transfer system is designed to withstand the effects of or is protected against natural phenomena, such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.5.4.1.2 General Design Criterion 3, 1971 - Fire Protection

The EDG fuel oil storage and transfer system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.5.4.1.3 General Design Criterion 4, 1967 - Sharing of Systems

The EDG fuel oil storage and transfer system or components are not shared by the DCPP units unless it is shown safety is not impaired by the sharing.

9.5.4.1.4 General Design Criterion 11, 1967 - Control Room

The EDG fuel oil storage and transfer system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.5.4.1.5 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain EDG fuel oil storage and transfer system variables within prescribed operating ranges.

9.5.4.1.6 General Design Criterion 17, 1971 - Electric Power Systems

The EDG fuel oil storage and transfer system is designed with sufficient capacity, capability, independence, redundancy, and testability to perform its safety function assuming a single failure.

9.5.4.1.7 General Design Criterion 21, 1967 - Single Failure Definition

The EDG fuel oil storage and transfer system is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event shall be treated as a single failure.

9.5.4.1.8 Emergency Diesel Generator Fuel Oil Storage and Transfer System Safety Function Requirements

(1) Protection from Missiles

The PG&E Design Class I portion of the EDG fuel oil storage and transfer system is designed to be protected against missiles that might result from plant equipment failure and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(2) Protection Against High Energy Pipe Rupture Effects

The PG&E Design Class I portion of the EDG fuel oil storage and transfer system is designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(3) Protection from Moderate Energy Pipe Rupture Effects

The PG&E Design Class I portion of the EDG fuel oil storage and transfer system is designed to be protected against the effects of moderate energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(4) Protection from Flooding Effects

The PG&E Design Class I portion of the EDG fuel oil storage and transfer system is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

9.5.4.1.9 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The EDG fuel oil storage and transfer system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.5.4.1.10 Regulatory Guide 1.137, Revision 1, October 1979 – Fuel-Oil Systems for Standby Diesel Generators

For proper operation of the EDGs, it is necessary to ensure the proper quality of the fuel oil. It is also necessary to adhere to recommended fuel oil practices of checking and removing accumulated water from the day tank and main storage tank, respectively, as well as draining the fuel stored in the fuel storage tanks to remove accumulated sediments and to clean the tanks.

9.5.4.1.11 10 CFR 50.63 – Loss of All Alternating Current Power

The EDG fuel oil storage and transfer system is required to (1) maintain adequate storage of diesel fuel oil, and (2) transfer fuel oil from the storage tanks to the day tank associated with the EDG (AAC source) supplying the blacked-out unit in the event of a station blackout (SBO)

9.5.4.2 System Description

The diesel generator fuel oil system diagram is shown in Figure 3.2-21. The physical arrangement of the engine generator units is shown in Figures 9.5-10 and 9.5-11 for Unit 1; the arrangement is similar for Unit 2. Figure 9.5-12 shows the outline of the Unit 1 engine generators; the arrangement is similar for the Unit 2 generators with the exception of EDG 2-3, which is slightly different. The design data is given in Table 9.5-2. The system consists of the following major components and features:

(1) Two underground diesel fuel oil storage tanks, each with a storage capacity of 50,000 gallons.

(2) Two diesel fuel oil transfer pumps located below ground level, each adjacent to a storage tank but in separate compartments. Each transfer pump delivers more than 55 gpm at a discharge pressure of approximately 50 psig. Pumps are of the positive displacement rotary screw type with 5-hp motors. One pump is more than adequate to supply the six diesel generators of Unit 1 and Unit 2 running at rated load. A duplex-type strainer is installed upstream of each fuel oil transfer pump to protect the pump from particles that could damage it. A cartridge-type fuel oil filter is located at the discharge of the fuel oil transfer pumps to prevent any fuel oil contamination from reaching the engine-base-mounted diesel fuel oil tanks. The diagram of the engine fuel oil transfer system is shown in Figure 3.2-21. The physical arrangement of the fuel oil transfer system is shown in Figures 9.5-8 and 9.5-9.

(3) Two diesel fuel oil supply headers to each unit routed in separate trenches.

The motor controllers for the two transfer pumps are located inside the auxiliary building. Manual pump control stations and manual controls for the valves are located near each diesel generator set.

The other engine generator auxiliary systems and accessories are essentially as provided on all engine generator units of this size. Each engine generator unit is equipped with a skid-mounted fuel oil tank that has a capacity of 550 gallons, which provides about 2-1/2 hours of full load operation before fuel oil must be transferred from the underground storage tanks. Fuel is transferred to each diesel generator skid-mounted fuel oil tank via two LCVs (and two associated upstream isolation valves) per diesel generator. Each of the two LCVs and associated upstream isolation valves on each diesel generator is associated with a separate diesel fuel oil transfer system train; however, the LCVs, the isolation valves immediately upstream from the LCVs, and the skid-mounted fuel oil tank are part of the associated diesel generator rather than the diesel fuel oil transfer system. The fuel oil transfer pumps start automatically on low level in the engine generator unit tanks. The two 50,000-gallon storage tanks provide a 7-day supply of fuel.

Each EDG day tank has the capacity for approximately 2.5 hours of continuous full-load operation. The fuel oil transfer system is designed to replenish the day tanks from the underground fuel oil storage tanks to ensure onsite power is available following a design-basis accident. Each day tank has two associated redundant fuel oil transfer system LCVs that automatically open to replenish the tank. The LCVs are air-operated by the associated diesel starting air receiver tanks. To ensure the starting air system is capable of supporting the required automatic LCV operation, the leakage of each diesel starting air system is verified periodically (Reference 4).

Experience with the PG&E transmission system indicates that in the event of complete loss of offsite power, restoration of normal power sources could be accomplished within a few hours. However, 7 days of onsite power generation has been used as a conservative upper limit for design and safety evaluations of fuel storage capacity for the tanks, even though it is highly improbable that the diesel generators would be required to furnish plant auxiliary power for this long a period.

9.5.4.3 Safety Evaluation

9.5.4.3.1 General Design Criterion 2, 1967 - Performance Standards

The EDG units and their associated auxiliary systems, as shown for Unit 1 in Figures 9.5-8 through 9.5-10, and similarly for Unit 2, are installed in separate compartments that are protected from fires, flooding, and external missiles.

The system valves, fittings, and piping are fabricated and inspected to ANSI Code for Pressure Piping B31.1 and B31.7 where applicable. The diesel fuel oil storage tanks are designed and fabricated to UL Standard 58. Seismic effects on the buried tanks are determined by a soil structure interaction analysis. Seismic effects are combined with gravity effects, including those resulting from the weight of the tanks and their contents, and the weight of the soil overburden.

Protection against corrosion problems is provided for underground portions of the system. Tank supports have firm foundations to minimize uneven settling and prevent seismic damage.

The exterior surfaces of the tanks have a corrosion-resistant fiberglass reinforced plastic wrap. Cathodic protection was provided to protect the tanks and underground piping from corrosion in case of damage to the coating. This cathodic protection system provided minimal protection, is no longer functioning, and has been abandoned in place (Reference 5). Most jurisdictions (including San Luis Obispo County) do not require cathodic protection for fiberglass reinforced plastic clad double wall tanks. California Code of Regulations 23 CCR 2635 does not require cathodic protection for this type of tank construction. ANSI/ANS 59.51-1997 states the use of a double wall tank design is an adequate means of meeting corrosion requirements.

Underground steel piping is provided protection from the effects of long term corrosion by coating or wrapping, in accordance with NACE Standard Practice SP0169. The transfer and vent piping is electrically isolated from the tanks in accordance with NACE Standard Practice SP0169. Magnesium anodes are utilized to protect the tank hold down straps.

A calculation was performed to determine the maximum ambient temperature inside the pump vaults due to heat output of fuel oil transfer pump motor (Reference 5). The motor is rated to withstand this maximum temperature. The design outdoor design temperature is based on 9 years of onsite hourly data, and the minimum recorded onsite outdoor temperature during this 9-year period is 39°F (refer to Section 9.4). Based on the results of the calculations, and the fact that the extreme low temperature recorded at the DCPP site was above freezing point, it is concluded that neither heating nor cooling is required to perform safe operation of the fuel oil pumps. However, for personnel protection, temporary portable ventilation equipment will be used as required to restore the confined space for habitability inside the vaults during periodic inspection and maintenance, in accordance with the plant administrative procedures.

9.5.4.3.2 General Design Criterion 3, 1971 - Fire Protection

The EDG areas are designed to meet the requirements of 10 CFR 50.48(a) and (c) (refer to Section 9.5.1).

Refer to Section 8.3.1.4.9 for further discussion on fire barriers and separation. To ensure good fire protection practice, the diesel engine generator installation was designed in accordance with NFPA Standard No. 37, Standard for the Installation and Use of Stationary Combustion Engines and Gas Turbines. This standard permits a maximum capacity of 660 gallons (550 imperial gallons) for an integral fuel oil day tank. It also provides sufficient time for all necessary operator actions to ensure that diesel generator operation is not interrupted in the event of any malfunctions in the system that transfers fuel oil from the underground storage tanks to the day tanks.

The two diesel fuel oil storage tanks are buried, and are designed to the criteria of Part 2, Bulk Underground Storage, of NFPA Standard No. 30, Standard for the Storage, Handling and Use of Flammable Liquids. The National Fire code does not require any specific fire protection system for this type of underground storage tank. Physical separation of the two tanks precludes the possibility of a fire in one storage tank from spreading to another tank. Fire risk is further minimized by the fact that the only source of oxygen to support a fire is through the 4 inch tank vent. If sufficient heat were generated to damage and collapse the storage tank, dirt would cave in and help smother the fire. Two yard hose reel stations are available for fighting any above ground fires at the tank location.

9.5.4.3.3 General Design Criterion 4, 1967 - Sharing of Systems

The diesel generator fuel oil system is provided to supply diesel oil to the emergency diesel engine generators for Unit 1 and Unit 2. The design classification of this system is given in Table 3.2-3. The fuel storage capacity provides 7 days of onsite power generation in order to operate (a) the minimum required ESF equipment following a LOCA for one unit, and the equipment for the second unit in either the hot or cold shutdown condition, or (b) the equipment for both units in either the hot or cold shutdown condition. The supply of fuel beyond the 7 day period is ensured by the availability of offsite sources and a reliable delivery method.

Safety of the reactor facilities is not impaired by the sharing of the fuel oil systems as any combination of one storage tank and one pump is capable of serving all six-day tanks. Each unit normally supplies power to one transfer pump from one Class 1E 480-V bus (refer to Section 8.3.1.1.4.3.3). Provisions are made to manually switch both pumps to the Class 1E buses of either unit, in case one unit should be placed in a prolonged cold shutdown condition.

9.5.4.3.4 General Design Criterion 11, 1967 - Control Room

System instrumentation and control is provided on the tanks and pumps as follows:

(1) The base-mounted day tanks have two separate redundant transfer pump start-stop level switches. Each level switch starts a transfer pump and opens the supply header solenoid valve corresponding to the respective transfer pump, A or B. The start setting for the header A level switches is slightly different from those for header B, allowing one to be a backup.

(2) The start of transfer pump A or B is indicated both locally and in the control room.

(3) Local controls at each diesel generator and manual crosstie valving between headers allow manual starting of either transfer pump and filling of the base-mounted day tanks from either header system A or B.

(4) High- and low-level alarm switches are installed on all base-mounted day tanks that activate alarms both locally and in the control room to alert the operators.

(5) High- and low-level alarm switches are installed on both fuel oil storage tanks that will activate alarms in the control room. Additionally, dipstick-type indicators and a local level indicator are provided for each storage tank.

Additionally, a monitoring system to detect the leakage of oil from the fuel oil transfer system piping in the fuel oil transfer trenches is provided to comply with California Underground Storage Tank regulations.

9.5.4.3.5 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Refer to Section 9.5.4.3.4 for discussion related to instrumentation and control.

9.5.4.3.6 General Design Criterion 17, 1971 - Electric Power Systems

The EDG system has sufficient capacity, capability, independence, redundancy, and testability to perform its safety function assuming a single failure. Refer to Sections 9.5.4.2 and 9.5.4.3.7 for details on the system.

9.5.4.3.7 General Design Criterion 21, 1967 - Single Failure Definition

The diesel generator fuel oil system is designed to remain operable after sustaining a single failure of either an active or a passive component. The capability to meet the single failure criterion is met by providing redundancy in tanks, pumps, valves, piping, and power supplies. The system arrangement provides sufficient separation of the tanks and their associated transfer pumps so that the possibility of damage to both simultaneously as a result of a single event is considered highly unlikely. The fuel oil transfer piping, transfer pumps, and tank manifolding are arranged so a single failure of any pipe, valve, tank, or pump will not disable the system.

The design incorporates sufficient redundancy so that a malfunction or failure of either an active or a passive component will not impair the ability of the system to supply fuel oil.

As discussed in the preceding paragraph, the diesel fuel oil system transfer components and power sources are redundant up to and including fill valves and connections on the engine day tanks, so that a single malfunction will not prevent the transfer of oil. In the unlikely event of malfunctions in both redundant fuel oil headers, such as a pump failure in one and piping blockage in the other, low level will be alarmed when sufficient fuel oil remains in the base-mounted day tank for a nominal one hour period of operation of the engine at full load. This nominal one hour period is adequate for an operator (a) to correct a malfunction on one of the two redundant transfer headers, or (b) to line up manually the valves of the two headers into one path that will transfer oil. All the valves necessary for this action are readily accessible in the compartments for the diesel fuel oil transfer pumps.

Each diesel generator has a dedicated shaft driven fuel pump, priming tank and day tank along with instrumentation, and fuel injectors such that failure of this engine mounted fuel equipment affects its associated diesel generator only.

9.5.4.3.8 Emergency Diesel Generator Fuel Oil Storage and Transfer System Safety Function Requirements

(1) <u>Protection from Missiles</u>

The EDG units and their associated auxiliary systems, as shown for Unit 1 in Figures 9.5-8 through 9.5-10, and similarly for Unit 2, are installed in separate compartments that are protected from internal missiles.

The provisions taken to protect the PG&E Design Class I, EDG fuel oil storage and transfer system from missiles resulting from plant equipment failures and from events and conditions outside the plant are discussed in Sections 3.5.

(2) <u>Protection Against High Energy Pipe Rupture Effects</u>

The provisions taken to protect the PG&E Design Class I portion of the EDG fuel oil storage and transfer system from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Section 3.6.

(3) <u>Protection from Moderate Energy Pipe Rupture Effects</u>

The provisions taken to provide protection of the PG&E Design Class I portion of the EDG fuel oil storage and transfer system from the effects of moderate energy pipe failure are discussed in Section 3.6.

(4) <u>Protection from Flooding Effects</u>

The provisions taken to provide protection of the PG&E Design Class I portion of the EDG fuel oil storage and transfer system from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

The design considerations to prevent water from flooding or groundwater from entering the fuel oil storage tanks, concrete vaults, and pipe trenches were:

(1) As discussed in Section 2.4, the risk of surface water flooding at this site is essentially zero. No groundwater has been encountered at or below the buried tanks, pump vaults, or pipe trenches. Therefore, the source potential for water flooding the fuel oil system is negligible. In addition, the below-ground system is completely sealed with the vent line extending approximately 2 feet above ground.

(2) Fuel oil tank vent lines running above ground are protected by concrete boxes and are surrounded by warning posts to alert any vehicular traffic. The concrete boxes will protect redundant vent lines from any credible common-mode failure.

(3) The two transfer pumps are in separate, underground, reinforced concrete vaults with solid covers protected from surface runoff due to their location inside the west buttress and condensate polishing system structure. The vault's manway hatch covers are made of steel and are provided with concrete curbing to prevent water intrusion into the vaults. These vaults are drained to the building sump and are protected with backwater valves. Flooding of the transfer pump vaults is alarmed at the TBS local annunciator and on the Unit 1 main annunciator.

(4) The two redundant fuel oil supply headers are in separate, below-ground reinforced concrete pipe trenches with solid steel or concrete covers that are generally flush with the adjacent ground level, except as noted below. Since the trenches collect water from surface runoff, drainage is provided through floor drains to manholes, which are pumped to the TBSs or standpipes that can be connected to portable pumps. Portions of the fuel oil supply header trenches routed in the rooms housing the EDGs are provided with metal grating. The grating provides physical protection of the headers, allows for visual inspection, and provides access for manual control of the fuel oil LCVs using a wrench. Because the grating is located indoors, the potential for flooding is extremely small. The trenches, with the exception of that in the room housing EDG 2-3, drain back to the turbine building sump. The trench in the room housing EDG 2-3 main header does not drain to a sump and must be pumped out manually.

(5) The design classification for the diesel fuel oil piping within the trenches is given in Table 3.2-3.

The possibility of flooding in the turbine building and in the diesel generator compartments is discussed in Section 8.3.1.1.6.3.9.

9.5.4.3.9 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The EDG system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition (refer to Section 9.5.1).

9.5.4.3.10 Regulatory Guide 1.137, Revision 1, October 1979 – Fuel-Oil Systems for Standby Diesel Generators

There is no specific DCPP commitment to use Regulatory Guide 1.137, Revision 1, guidance to establish fuel oil quantity requirements for 7 day EDG operation.

The Technical Specification required fuel oil quantity for 7 day EDG operation is based on the calculated fuel oil consumption necessary to support the operation of the EDGs to power the minimum ESF systems required to mitigate a design basis LOCA in one unit and those minimum required systems for a concurrent non-LOCA safe shutdown in the other unit (both units initially in Mode 1 operation).

Proper operation of the EDGs requires following recommended fuel oil practices to ensure proper quality. These practices include checking and removing accumulated water from the day tanks and main storage tanks and draining the main fuel storage tanks, removing accumulated sediment, and cleaning the tanks. Regulatory Guide 1.137, Revision 1, recommends surveillance frequencies for these practices as well as assurance of fuel oil quality in accordance with applicable industry standards.

9.5.4.3.1110 CFR 50.63 – Loss of All Alternating Current Power

In the event of a SBO, a diesel fuel oil transfer pump powered from a Class 1E 480-V bus in the non-blacked-out unit takes suction from a diesel fuel oil storage tank and provides fuel oil to the day tank associated with the EDG (AAC source) supplying the blacked-out unit.

Each EDG at DCPP is served by its own separate day tank, which is supplied from a common storage tank. The fuel oil in the common tank is sampled and analyzed in accordance with Technical Specifications to ensure it meets acceptable high quality standards.

Refer to Section 8.3.1.6 for further discussion of station blackout.

9.5.4.4 Tests and Inspections

The diesel fuel supply headers are hydrostatically or pneumatically tested during construction and all active system components, pumps, valves, and controls are functionally tested during startup and periodically thereafter. The diesel fuel oil in storage will be periodically tested for any possible contamination or deterioration. The underground Diesel Fuel Oil and Transfer System is demonstrated operable as required by the Technical Specifications and by:

(1) Draining each fuel oil storage tank, removing the accumulated sediment, and cleaning the tank using a sodium hypochlorite or equivalent solution at least once per ten years, and

(2) Performing a visual examination of accessible piping in accordance with the Buried Piping and Tanks Program.

9.5.4.5 Instrumentation Application

Refer to Section 9.5.4.3.4 for discussion related to instrumentation application.

9.5.5 DIESEL GENERATOR COOLING WATER SYSTEM

The diesel generator cooling water system is shown schematically in Figure 3.2-21. The physical arrangement of the EDG units is shown in Figures 9.5-10 and 9.5-11 for Unit 1; the arrangement is similar for Unit 2. Figure 9.5-12 shows the outline of the Unit 1 EDGs. The arrangement is similar for the Unit 2 EDGs with the exception of EDG 2-3, which is slightly different.

9.5.5.1 Design Bases

9.5.5.1.1 General Design Criterion 2, 1967 - Performance Standards

The EDG cooling water system is designed to withstand the effects of, or is protected against natural phenomena such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.5.5.1.2 General Design Criterion 3, 1971 - Fire Protection

The EDG cooling water system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.5.5.1.3 General Design Criterion 11, 1967 - Control Room

The EDG cooling water system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.5.5.1.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain EDG cooling water system variables within prescribed operating ranges.

9.5.5.1.5 General Design Criterion 21, 1967 - Single Failure Definition

The EDG cooling water system is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event shall be treated as a single failure.

9.5.5.1.6 Emergency Diesel Generator Cooling Water System Safety Function Requirements

(1) Protection from Missiles

The PG&E Design Class I portion of the EDG cooling water system is designed to be protected against missiles that might result from plant equipment failure and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(2) Protection Against High Energy Pipe Rupture Effects

The PG&E Design Class I portion of the EDG cooling water system is designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(3) Protection from Moderate Energy Pipe Rupture Effects

The PG&E Design Class I portion of the EDG cooling water system is designed to be protected against the effects of moderate energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(4) Protection from Flooding Effects

The PG&E Design Class I portion of the EDG cooling water system is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

9.5.5.1.7 10 CFR 50.55a(g) - Inservice Inspection Requirements

Applicable EDG cooling water system components must meet the requirements of 10 CFR 50.55a(g). The EDG jacket water cooling system is the only EDG component included in the DCPPISI Program.

9.5.5.1.8 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The EDG cooling water system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.5.5.2 System Description

A closed loop jacket water-cooling system is provided for each of the six diesel engines. The engine generator skid has an integrally mounted radiator with a direct engine-driven fan for cooling the engine jacket water. Cooling air is largely outside ambient air, drawn by the fan from outside the building through the tornado missile shield into the radiator-fan portion of the engine generator compartment, and then through the radiator core. The diesel generator ventilation system is described in Section 9.4.7. This closed system allows the diesel generator unit to function in a self-contained manner, independent of outside cooling water systems and electric motor-driven fans.

The radiator is a jacket water-to-air heat exchanger of all copper and brass construction. Makeup to the jacket water cooling system can be added through a fill line after removing the radiator cap, labeled RV-71 in Figure 3.2-21 (Sheets 13, 14, 14A, and 14B). A low jacket water level alarm notifies the operator that makeup is required.

Engine jacket water is circulated by a jacket water pump directly driven by the engine. A three-way temperature regulating valve bypasses a portion of engine jacket cooling water around the radiator to maintain proper system temperature. Lubricating oil is cooled by a shell and tube heat exchanger using the jacket water as the coolant. Thermostatically controlled immersion heaters keep the jacket water warm for fast starting while the engine is in a shutdown condition.

The generators on these units are air-cooled by a shaft-mounted blower. The cooling system is shown schematically in Figure 9.4-6.

9.5.5.3 Safety Evaluation

9.5.5.3.1 General Design Criterion 2, 1967 - Performance Standards

Refer to Section 8.3.1.1.6.3.1 for discussion regarding the protection of the EDGs, and their associated auxiliary systems from flooding and external missiles.

9.5.5.3.2 General Design Criterion 3, 1971 - Fire Protection

Refer to Section 8.3.1.1.6.3.2 for discussion regarding the design of the fire protection system for the EDGs and their associated auxiliary systems.

9.5.5.3.3 General Design Criterion 11, 1967 - Control Room

Refer to Section 8.3.1.1.6.3.4 for discussion regarding the design of the EDGs and their associated auxiliary systems to support safe shutdown and to maintain safe shutdown from the control room or from an alternate location if the control room access is lost due to fire or other causes.

9.5.5.3.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Refer to Sections 8.3.1.1.6.3.5 and 8.3.1.1.6.5 for discussions regarding the instrumentation and control for the EDGs and their associated auxiliary systems provided as required to monitor and maintain their variables within prescribed operating ranges.

9.5.5.3.5 General Design Criterion 21, 1967 - Single Failure Definition

Each diesel generator has a dedicated radiator/fan set, water pump and expansion tank such that failure of the cooling equipment affects its associated diesel generator only.

Refer to Section 8.3.1.1.6.3.8 for discussion regarding single failure criterion.

9.5.5.3.6 Emergency Diesel Generator Cooling Water System Safety Function Requirements

(1) <u>Protection from Missiles</u>

The EDG units and their associated auxiliary systems, as shown for Unit 1 in Figures 9.5-8 through 9.5-10, and similarly for Unit 2, are installed in separate compartments that are protected from internal missiles. Because EDG units are separated from each other by the concrete walls of the compartments, the units are protected from postulated internal missiles. Any missile created by an explosion within a compartment would remain in that compartment.

The provisions taken to protect the PG&E Design Class I, EDG cooling water system from missiles resulting from plant equipment failures and from events and conditions outside the plant are discussed in Sections 3.5.

(2) <u>Protection Against High Energy Pipe Rupture Effects</u>

The provisions taken to protect the PG&E Design Class I portion of the EDG cooling water system from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Section 3.6.

(3) <u>Protection from Moderate Energy Pipe Rupture Effects</u>

The provisions taken to provide protection of the PG&E Design Class I portion of the EDG cooling water system from the effects of moderate energy pipe failure are discussed in Section 3.6.

(4) Protection from Flooding Effects

The provisions taken to provide protection of the PG&E Design Class I portion of the EDG cooling water system from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

The possibility of flooding in the turbine building and in the diesel generator compartments is discussed in Section 8.3.1.1.6.3.9.

9.5.5.3.7 10 CFR 50.55a(g) - Inservice Inspection Requirements

Refer to Section 8.3.1.1.6.3.10 for discussion regarding compliance to the requirement of 10 CFR 50.55a(g).

9.5.5.3.8 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805 The EDG system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition (refer to Section 9.5.1).

9.5.5.4 Tests and Inspections

Refer to Section 8.3.1.1.6.4 for discussion regarding the inspection and testing associated with the EDGs and their associated auxiliary systems.

9.5.5.5 Instrumentation Applications

Instrumentation application for this system is discussed in Section 8.3.1.1.6.5.

9.5.6 DIESEL GENERATOR STARTING SYSTEM

The diesel generator starting system is shown schematically in Figure 3.2-21. The physical arrangement of the engine generator units is shown in Figures 9.5-10 and 9.5-11 for Unit 1; the arrangement is similar for Unit 2. Figure 9.5-12 shows the outline of the Unit 1 engine generators. The arrangement is similar for the Unit 2 generators with the exception of EDG 2-3, which is slightly different.

Each diesel generator is designed with two starting control circuits, one field flashing circuit, and one sensing circuit. These circuits receive Class 1E dc control power through a manual transfer switch. Normal Class 1E dc power is from the same train as the diesel generator. In the event of failure of dc power to these control circuits, an alarm appears on the main annunciator. The manual transfer switch, located near the control panel at the diesel generator can be used to transfer to backup Class 1E dc power.

9.5.6.1 Design Bases

9.5.6.1.1 General Design Criterion 2, 1967 - Performance Standards

The EDG starting system is designed to withstand the effects of, or is protected against natural phenomena such as earthquakes, tornadoes, flooding, winds, and tsunamis other local site effects.

9.5.6.1.2 General Design Criterion 3, 1971 - Fire Protection

The EDG starting system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.5.6.1.3 General Design Criterion 11, 1967 - Control Room

The EDG starting system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.5.6.1.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain EDG starting system variables within prescribed operating ranges.

9.5.6.1.5 General Design Criterion 21, 1967 - Single Failure Definition

Refer to Section 8.3.1.1.6.3.8 for discussion regarding single failure criterion.

9.5.6.1.6 Emergency Diesel Generator Starting System Safety Function Requirements

(1) <u>Protection from Missiles</u>

The PG&E Design Class I portion of the EDG starting system is designed to be protected against missiles that might result from plant equipment failure and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(2) <u>Protection Against High Energy Pipe Rupture Effects</u>

The PG&E Design Class I portion of the EDG starting system is designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(3) Protection from Moderate Energy Pipe Rupture Effects

The PG&E Design Class I portion of the EDG starting system is designed to be protected against the effects of moderate energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(4) Protection from Flooding Effects

The PG&E Design Class I portion of the EDG starting system is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

9.5.6.1.7 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The EDG starting system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.5.6.2 System Description

Each diesel engine of the six engine generator sets is provided with two separate air-start systems. Normally two trains, four starting air motors, operate together in combination with the turbo-charger air assist system to ensure that the engine generator set starts and accelerates to rated speed and to minimum bus voltage in less than 10 seconds. Each of the two air-starting systems consists of an air receiver; a non-Class 1E electric motor-driven air compressor to charge the receiver; two air-starting motors that engage and turn the engine flywheel; and all piping, valves, and controls necessary to provide a complete system. Each of the two redundant PG&E Design Class II air compressors for each engine generator unit is fed from different Class 1E buses so that the possibility of simultaneous loss of both air compressors is minimized. The diesel engine vendor sizing criteria for each air start receiver is to provide sufficient capacity for three consecutive 15-second cranking cycles. Additional cranking cycles can be made as the PG&E Design Class II air compressors replenish their air receivers.

In addition to the above described two air-start systems, each diesel engine is equipped with an engine turbocharger boost system. The turbocharger boost system serves two functions: it aids in acceleration of the large rotating mass of the turbocharger, and it provides extra air to the engine to improve combustion during acceleration. The additional air is necessary since the turbocharger is inefficient at low speeds. The turbocharger boost system consists of one PG&E Design Class II air compressor, one air receiver, and all piping, valves, and controls necessary to provide a complete system. The turbocharger boost system with two air-starting motors is capable of starting and accelerating the engine generator set to rated speed and to minimum bus voltage in 10 seconds. The starting air system and the turbocharger boost systems are shown schematically in Figure 3.2-21 (Sheets 3 through 6B). The physical arrangement of those systems is shown in Figures 9.5-10 and 9.5-11.

9.5.6.3 Safety Evaluation

9.5.6.3.1 General Design Criterion 2, 1967 - Performance Standards

Refer to Section 8.3.1.1.6.3.1 for discussion regarding the protection of the EDGs and their associated auxiliary systems from flooding and external missiles.

9.5.6.3.2 General Design Criterion 3, 1971 - Fire Protection

Refer to Section 8.3.1.1.6.3.2 for discussion regarding the design of the fire protection system for the EDGs and their associated auxiliary systems.

9.5.6.3.3 General Design Criterion 11, 1967 - Control Room

Refer to Section 8.3.1.1.6.3.4 for discussion regarding the design of the EDGs and their associated auxiliary systems to support safe shutdown and to maintain safe shutdown from the control room or from an alternate location if the control room access is lost due to fire or other causes.

9.5.6.3.4 General Design Criterion 12, 1967 - Instrumentation and Control

Refer to Sections 8.3.1.1.6.3.5 and 8.3.1.1.6.5 for discussions regarding the instrumentation and control for the EDGs and their associated auxiliary systems provided as required to monitor and maintain their variables within prescribed operating ranges.

9.5.6.3.5 General Design Criterion 21, 1967 - Single Failure Definition

Each diesel generator has dedicated air start receiver tanks (redundant), and a turbocharger receiver tank along with solenoid valves such that failure of this air start equipment affects its associated diesel only.

Refer to Section 8.3.1.1.6.3.8 for discussion regarding single failure criterion.

9.5.6.3.6 Emergency Diesel Generator Starting System Safety Function Requirements

(1) <u>Protection from Missiles</u>

The EDG units and their associated auxiliary systems, as shown for Unit 1 in Figures 9.5-8 through 9.5-10, and similarly for Unit 2, are installed in separate compartments that are protected from internal missiles. Because EDG units are separated from each other by the concrete walls of the compartments, the units are protected from postulated internal missiles. Any missile created by an explosion within a compartment would remain in that compartment.

The provisions taken to protect the PG&E Design Class I, EDG starting system from missiles resulting from plant equipment failures and from events and conditions outside the plant are discussed in Sections 3.5.

(2) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect the PG&E Design Class I portion of the EDG starting system from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Section 3.6.

(3) Protection from Moderate Energy Pipe Rupture Effects

The provisions taken to provide protection of the PG&E Design Class I portion of the EDG starting system from the effects of moderate energy pipe failure are discussed in Section 3.6.

(4) Protection from Flooding Effects

The provisions taken to provide protection of the PG&E Design Class I portion of the EDG starting system from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

The possibility of flooding in the turbine building and in the diesel generator compartments is discussed in Section 8.3.1.1.6.3.9.

9.5.6.3.7 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The EDG starting system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition (refer to Section 9.5.1).

9.5.6.4 Tests and Inspections

Refer to Section 8.3.1.1.6.4 for discussion regarding the inspection and testing associated with the EDGs and their associated auxiliary systems.

9.5.6.5 Instrumentation Applications

Instrumentation application for this system is discussed in Section 8.3.1.1.6.5.

9.5.7 DIESEL GENERATOR LUBRICATION SYSTEM

The diesel generator lubrication system is shown schematically in Figure 3.2-21. The physical arrangement of the engine generator units is shown in Figures 9.5-10 and 9.5-11 for Unit 1; the arrangement is similar for Unit 2. Figure 9.5-12 shows the outline of the Unit 1 engine generators. The arrangement is similar for the Unit 2 generators with the exception of EDG 2-3, which is slightly different.

9.5.7.1 Design Bases

9.5.7.1.1 General Design Criterion 2, 1967 - Performance Standards

The EDG lubrication system is designed to withstand the effects of, or is protected against natural phenomena such as earthquakes, tornadoes, flooding, winds, tsunamis and other local site effects.

9.5.7.1.2 General Design Criterion 3, 1971 - Fire Protection

The EDG lubrication system is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

9.5.7.1.3 General Design Criterion 11, 1967 - Control Room

The EDG lubrication system is designed to or contains instrumentation and controls that support actions to maintain the safe operational status of the plant from the control room or from an alternate location if control room access is lost due to fire or other causes.

9.5.7.1.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Instrumentation and controls are provided as required to monitor and maintain EDG lubrication system variables within prescribed operating ranges.

9.5.7.1.5 General Design Criterion 21, 1967 - Single Failure Definition

The EDG lubrication system is designed to remain operable after sustaining a single failure. Multiple failures resulting from a single event are treated as a single failure.

9.5.7.1.6 Emergency Diesel Generator Lubrication System Safety Function Requirements

(1) Protection from Missiles

The PG&E Design Class I portion of the EDG lubrication system is designed to be protected against missiles that might result from plant equipment failure and conditions outside the plant to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(2) Protection Against High Energy Pipe Rupture Effects

The PG&E Design Class I portion of the EDG lubrication system is designed and located to accommodate the dynamic effects of a postulated high-energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(3) Protection from Moderate Energy Pipe Rupture Effects

The PG&E Design Class I portion of the EDG lubrication system is designed to be protected against the effects of moderate energy pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

(4) Protection from Flooding Effects

The PG&E Design Class I portion of the EDG lubrication system is designed to be protected from the effects of internal flooding to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

9.5.7.1.7 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The EDG lubrication system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition.

9.5.7.2 System Description

The lubricating oil system for each engine is entirely contained on that engine's baseplate. The system is schematically shown in Figure 3.2-21 (Sheets 11, 11A, 11B and 12, 12A, 12B). During engine operation, all required lubricating oil is drawn from the engine crankcase through a shaft-mounted oil pump to a lubricating oil filter with a built-in pressure relief device to bypass lubricating oil flow in the event that the filter becomes excessively dirty. The oil is then cooled in the jacket water-cooled heat exchanger and returned to the engine bearings through a duplex strainer. During normal operation, engine cooling water will not leak into the lubricating oil due to a leak in the lubricating oil heat exchanger since the operating pressure of the lubricating oil system is above 75 psig, and the operating pressure of the cooling water system is about 35 psig. Instrumentation provided in the lubricating oil circuit is described in Section 8.3.1.1.6.5.

While the engine is idle, oil is continually circulated by means of a small precirculating pump. A thermostatically controlled immersion heater on the outlet of the precirculating pump maintains a 90-110°F oil temperature to ensure rapid and safe startup of the engine at any time.

9.5.7.3 Safety Evaluation

9.5.7.3.1 General Design Criterion 2, 1967 - Performance Standards

Refer to Section 8.3.1.1.6.3.1 for discussion regarding the protection of the EDGs and their associated auxiliary systems from flooding, and external missiles.

9.5.7.3.2 General Design Criterion 3, 1971 - Fire Protection

Refer to Section 8.3.1.1.6.3.2 for discussion regarding the design of the fire protection system for the EDGs and their associated auxiliary systems.

9.5.7.3.3 General Design Criterion 11, 1967 - Control Room

Refer to Section 8.3.1.1.6.3.4 for discussion regarding the design of the EDGs and their associated auxiliary systems to support safe shutdown and to maintain safe shutdown from the control room or from an alternate location if the control room access is lost due to fire or other causes.

9.5.7.3.4 General Design Criterion 12, 1967 - Instrumentation and Control Systems

Refer to Sections 8.3.1.1.6.3.5 and 8.3.1.1.6.5 for discussions regarding the instrumentation and control for the EDGs and their associated auxiliary systems provided as required to monitor and maintain their variables within prescribed operating ranges.

9.5.7.3.5 General Design Criterion 21, 1967 - Single Failure Definition

Each diesel generator has a dedicated lubrication oil tank, cooler, recirculation pump, and pre-circulation pump such that failure of this equipment affects the associated diesel generator only.

Refer to Section 8.3.1.1.6.3.8 for discussion regarding single failure criterion.

9.5.7.3.6 Emergency Diesel Generator Lubrication System Safety Function Requirements

(1) <u>Protection from Missiles</u>

The EDG units and their associated auxiliary systems, as shown for Unit 1 in Figures 9.5-8 through 9.5-10, and similarly for Unit 2, are installed in separate compartments that are protected from internal missiles. Because EDG units are separated from each other by the concrete walls of the compartments, the units are protected from postulated internal missiles. Any missile created by an explosion within a compartment would remain in that compartment.

The provisions taken to protect the PG&E Design Class I, EDG lubrication system from missiles resulting from plant equipment failures and from events and conditions outside the plant are discussed in Sections 3.5.

(2) Protection Against High Energy Pipe Rupture Effects

The provisions taken to protect the PG&E Design Class I portion of the EDG lubrication system from damage that might result from dynamic effects associated with a postulated rupture of high-energy piping are discussed in Section 3.6.

(3) Protection from Moderate Energy Pipe Rupture Effects

The provisions taken to provide protection of the PG&E Design Class I portion of the EDG lubrication system from the effects of moderate energy pipe failure are discussed in Section 3.6.

(4) Protection from Flooding Effects

The provisions taken to provide protection of the PG&E Design Class I portion of the EDG lubrication system from flooding that might result from the effects associated with a postulated rupture of piping are discussed in Section 3.6.

The possibility of flooding in the turbine building and in the diesel generator compartments is discussed in Section 8.3.1.1.6.3.9.

9.5.7.3.7 10 CFR 50.48(c) – National Fire Protection Association Standard NFPA 805

The EDG system is designed to meet the nuclear safety and radioactive release performance criteria of Section 1.5 of NFPA 805, 2001 Edition (refer to Section 9.5.1).

9.5.7.4 Tests and Inspections

Refer to Section 8.3.1.1.6.4 for discussion regarding the inspection and testing associated with the EDGs and their associated auxiliary systems.

9.5.7.5 Instrumentation Applications

Instrumentation application for this system is discussed in Section 8.3.1.1.6.5.

9.5.8 REFERENCES

1. Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Issuance of Amendments Regarding Transition to a Risk–Informed Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48 (c) (CAC Nos. MF2333 and MF2334), April 14, 2016.

2. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended.

3. Diablo Canyon Engineering Calculation HVAC 83-11, Pacific Gas and Electric Company, Nuclear Power Generation files.

4. Pacific Gas and Electric Company Probabilistic Risk Assessment Calculation File No. PRA04-11, Potential Loss of DFO Day Tank LCVs Following a Seismically Induced LOOP, July 5, 2005.

5. Pacific Gas and Electric Company Design Change Package DDP: 1000000205, Diesel Fuel Oil Storage Tank Cathodic Protection System Abandonment, April 2009

6. Calculation FP-805, NFPA 805 Design Basis Document, SAPN No. 9000041689

9.5.9 REFERENCE DRAWINGS

Figures representing controlled engineering drawings are incorporated by reference and are identified in Table 1.6-1. The contents of the drawings are controlled by DCPP procedures.

TABLE 9.1-1

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM ANALYSIS DATA[a]

Spent Fuel Pool Permanent Storage Capacity	1,324 assemblies[b]
Spent Fuel Pool Water Volume, cubic feet	47,215
Minimum Boron Concentration of the Spent Fuel Pool Water, ppm	2,000 (Up to 2600 during MPC Loading)*
Partial core off-load (Case 1): 96 fuel assemblies from current refueling assumed to have 52,000 MWD/MTU burnup. Conservative assumptions have been used to minimize decay time and maximize base decay hear load.	
Spent fuel pool HX CCW inlet temp, °F	75
Decay heat production, Btu/hr	22.92×10^6
Spent fuel pool water temperature, °F	≤ 127
Full core off-load (Case 2): 193 fuel assemblies from current refueling separated into two burnup groups: 101 assemblies at 52,000 MWD/MTU and 92 assemblies at 25,000 MWD/MTU. Conservative assumptions have been used to minimize decay time and maximize base decay heat load.	
Spent fuel pool HX CCW inlet temp, °F	75
Decay heat production, Btu/hr	36.67×10^6
Spent fuel pool water temperature, °F	≤ 157
Emergency full core off-load (Case 3): 193 assemblies from current refueling after 36 days of operation at full power. Fuel assemblies are separated into two burnup groups: 113 assemblies at 40,000 MWD/MTU and 80 assemblies at 3,000 MWD/MTU. Conservative assumptions have been used to minimize decay time and maximize base decay heat load.	
Spent fuel pool HX CCW inlet temp, °F	75
Decay heat production, Btu/hr	38.71×10^6
Spent fuel pool water temperature, °F	≤ 162
Spent Fuel Pool Water Heat Inertia	
Time to heat from 127 to 212°F for partial offload Case 1 above and no heat loss, hr	11.27
Time to heat from 157 to 212°F for full offload Case 2	4.35

TABLE 9.1-1

above, hr	
Time to heat from 162 to 212°F for emergency offload Case 3 above, hr	3.76

* Refer to Diablo Canyon ISFSI Technical Specifications

(a) DCPP thermal-hydraulic analyses have been changed per LAR 04-07, Enclosure 1, Section 4.3 as part of the temporary cask pit spent fuel storage rack project

(b) The maximum number of irradiated fuel assemblies stored in the SFP during normal plant operations is limited to 1433 fuel assemblies per DCPP TS LCO 3.7.17.C.

TABLE 9.1-2

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM DESIGN AND OPERATING PARAMETERS

	Pumps 1-1/2-1	Pumps 1-2/2-2
Spent Fuel Pool Pump		
Number, per unit	1	1
Design pressure, psig	150	100
Design temperature, °F	225	225
Design flow, gpm	2,300	3,000
Material	Stainless steel	Stainless steel
Spent Fuel Pool Skimmer Pump		
Number, per unit	1	
Design pressure, psig	50	
Design temperature, °F	200	
Design flow, gpm	100	
Material	Stainless steel	
Refueling Water Purification Pump		
Number, per unit	1	
Design pressure, psig	150	
Design temperature, °F	200	
Design flow, gpm	400	
Material	Stainless steel	
Spent Fuel Pool Heat Exchanger		
Number, per unit	1	
Design heat transfer, Btu/hr	11.95×10^6	
	Shell	Tube
Design pressure, psig	150	150
Design temperature, °F	250	250
Design flow, lb/hr	1.49×10^6	1.14×10^6
Inlet temperature, °F	95	120
Outlet temperature, °F	103	109.5
Fluid circulated	Component cooling water	Spent fuel pool water
Material	Carbon steel	Stainless steel

Spent Fuel Pool Demineralizer	
Number, per unit	1
Design pressure, psig	200
Design temperature, °F	250
Design flow, (min. to max.), gpm	27 to 109
Resin volume, ft^3	39
Material	Stainless steel
Spent Fuel Pool Filter	
Number, per unit	1
Design pressure, psig	200
Design temperature, °F	250
Design flow, gpm	150
Filtration requirement	98% retention of particles above 5 microns
Material, vessel	Stainless steel
Spent Fuel Pool Resin Trap Filter	
Number, per unit	1
Design pressure, psig	200
Design temperature, °F	250
Design flow, gpm	150
Filtration requirement	98 percent retention of particles above 5 microns
Material, vessel	Stainless steel
Spent Fuel Pool Skimmer Filter	
Number, per unit	1
Design pressure, psig	200
Design temperature, °F	250
Rated flow, gpm	150
Filtration requirement	98% retention of particles above 5 microns
Material, vessel	Stainless steel
Refueling Water Purification Filter	
Number, per unit	1
Design pressure, psig	200
Design temperature, °F	250
Design flow, gpm	400
Filtration requirement	98% retention of particles above 5 microns
Demineralizer Resin Trap	
Number, per unit	1
Slot opening, inches	0.010
Material	Stainless steel

TABLE 9.1-2

Spent Fuel Pool Strainer	
Number, per unit	1
Rated flow, gpm	2300
Perforation, inches	Approximately 0.2
Material	Stainless steel
Spent Fuel Pool Skimmer Strainer	
Number, per unit	1
Rated flow, gpm	100
Design pressure, psig	50
Design temperature, °F	200
Perforation, inches	1/8
Material	Stainless steel
Spent Fuel Pool Skimmers	
Number, per unit	2
Design flow, gpm	50
Piping and Valves	
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless steel

TABLE 9.2-1

AUXILIARY SALTWATER SYSTEM
COMPONENT DESIGN DATA

Auxiliary Saltwater Pumps

Number, per unit	2
Type	Vertical, wet pit, single stage
Rated capacity, gpm	11,000
Rated head, feet of sea water	125
Efficiency, %	83
Motor horsepower	465[(a)]
Speed, rpm	900
Column and discharge head material exposed to salt water	316L stainless steel
Design temperature, °F	120
Casing rated pressure, psig	100
Submergence required, ft	3
Minimum flow to prevent overheating, gpm	1,400

(a) Nameplate rating is 400 HP; analyzed to be capable of operating at 465 brake HP

TABLE 9.2-2

AUXILIARY SALTWATER SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Consequences and Comments
1. Auxiliary saltwater pump	Pump casing rupture	The casing is designed for 100 psig, which exceeds maximum operating conditions. Pump is inspectable. Standby pump provides redundancy.
2. Auxiliary saltwater pump	Pump fails to start	One operating pump supplies either normal or minimum postaccident flow. Second pump provides full redundancy. Indication in control room shows pump status.
3. Auxiliary saltwater pump	Manual valve at discharge closed or check valve sticks closed	This will be prevented by prestartup and operational checks. Indication in control room shows position of remotely operated valves and whether flow is established.
4. Component cooling water heat exchanger	Tube or channel rupture	Rupture is considered highly unlikely because of low operating pressures and use of corrosion resistant materials. However, the leaking heat exchanger can be identified by sequential isolation or visual inspection. If the leak is in the on-line heat exchanger, the standby heat exchanger would be placed in service and the leaking exchanger isolated.
5. Piping and general	Any leakage from system via open vent or drain valve, ruptured heat exchanger tube, or other malfunction of equipment served by the system	The leaking component can be identified by sequential isolation or visual inspection. Rupture in a main cooling header would be indicated in the control room as low differential pressure across the heat exchanger and high component cooling water temperature. The operator can remotely or manually transfer flow to the redundant supply header restoring the system's cooling function before there is a significant temperature rise in the component cooling water system.
6. Motor	Pipe rupture at pump-motor enclosure	Rupture is considered highly unlikely because of low operating pressures and use of corrosion-resistant materials in contact with the saltwater. However, each pump is located in an isolated enclosure. Rupture could flood one compartment but the other motor would be protected. Level transmitters in pump enclosures alarm to the control room when flooding is detected.
7. Traveling Screen	Screen clogging	Clogging of the ASW common traveling screen with debris is discussed in Section 9.2.7.2.3, "Intake Structure and Equipment."

TABLE 9.2-3

COMPONENT COOLING WATER SYSTEM COMPONENT DESIGN DATA

Component Cooling Water Pumps	
Number, per unit	3
Type	Horizontal centrifugal
Rated capacity, gpm	9200
Rated head, ft	145
Motor horsepower	400
Casing material	ASTM A-216 Gr. WCB carbon steel
Design pressure, psig	150
Design temperature, °F	300
Component Cooling Water Heat Exchangers	
Number, per unit	2
Type	Shell and tube
Heat transferred, Btu/hr	258.8×10^6
Shell-side	
Component cooling water outlet temperature, °F	125.0
Component cooling water inlet temperature, °F	171.7
Component cooling water flow rate, gpm	11,210
Design pressure, psig	150
Design temperature, °F	300
Tube side	
Auxiliary saltwater inlet temperature, °F	70
Auxiliary saltwater outlet temperature, °F	120
Auxiliary saltwater flow rate, gpm	10,580
Design pressure, psig	100
Design temperature, °F	200
Tube material	ASME SB-111 and/or ASME SB-543, 90-10 CuNi
Shell material	ASME SA 515 Gr. 70
Component Cooling Water Surge Tank	
Number, per unit	1
Type	Horizontal cylindrical with elliptical head: approximately 8 ft diameter x 30 ft long, internally baffled
Volume	
Total, gal	10,750
Normal operating, gal	Greater than 4,000
Design pressure, psig	150
Design temperature, °F	300
Material	ASME SA-285 Gr C

TABLE 9.2-4

COMPONENTS COOLED BY THE COMPONENT COOLING WATER SYSTEM

	Loop A	Loop B	Loop C
Containment fan coolers	2	3	
Residual heat removal heat exchangers	1	1	
Residual heat removal pump seal water coolers	1	1	
Centrifugal charging pumps CCP1 and CCP2 oil coolers	1	1	
Safety injection pump oil and seal water coolers	1	1	
Component cooling water pump oil coolers and stuffing boxes	2	1	
Post-LOCA sampling cooler	1		
Spent fuel pool heat exchanger			1
Seal water heat exchanger			1
Letdown heat exchanger			1
Excess letdown heat exchanger			1
NSSS sample heat exchangers – Unit 1			5
NSSS sample heat exchangers – Unit 2			4
Steam generator blowdown sample heat exchangers			5
Reactor coolant pump thermal barriers and motor oil coolers			4
Boric acid evaporator condenser, distillate cooler, vent condenser, and sample cooler (abandoned in place)			
Auxiliary steam drain receiver vent condenser (abandoned in place)			1
Waste gas compressors			2
Reactor vessel support coolers			4
Sample panel coolers			1

TABLE 9.2-5

COMPONENT COOLING WATER SYSTEM NOMINAL FLOWS[a]
(in gpm)

	Normal	LOCA[b]	Cooldown[c]
Containment fan coolers and motors	10,250	4,560	10,250
RHR heat exchangers	-	5,000	10,000
RHR pumps[f]	20	10	20
Centrifugal charging pumps CCP1 and CCP2[f]	30	15	30
Safety injection pumps[d][f]	48	24	48
Component cooling water pumps[e][f]	30	10	30
Spent fuel pool heat exchanger[i]	3,400	-	3,400
Seal water heat exchanger	210	-	210
Letdown heat exchanger	1,000	-	300
Excess letdown heat exchanger	-	-	-
NSSS sample heat exchanger	50	-	50
CCW sample line	10	-	10
Steam generator blowdown HX	40	-	40
Reactor coolant pumps	780	-	780
Boric acid evaporator package[h]	-	-	-
Waste concentrator package[h]	-	-	-
Auxiliary steam vent condenser[h]	-	-	-
Waste gas compressors	100	-	100
Reactor vessel support coolers	100	-	100
Sample panel coolers	40	-	40
Post-LOCA sample cooler	-	-	-
No. of pumps required	2	2	3[g]
No. of pumps normally in service	2	2 (one for each vital loop	3
No. of pumps installed	3	3	3

(a) Unless noted otherwise, the data contained in this table are nominal flows. For the purpose of design basis analyses, lower assumed flows may be used to evaluate equipment, containment heat removal capability, or different alignments (such as Section XI testing or RHR heat exchanger operation); higher assumed flows may be used to evaluate maximum CCW temperatures.

(b) Recirculation phase (flows are for vital header "B" which is greater than those of vital header "A"). The minimum required post-LOCA flow rates are shown. The flow rate for the CFCUs represents a minimum flow of 1490 gpm to the cooling coils and 30 gpm to the motor coolers. Note that during the injection phase the total minimum CFCU flow rate is 1650 gpm.

(c) 20 hours after initiation of reactor cooldown.

(d) For safety injection pumps, the minimum required cooling flow following a LOCA is 24 gpm per pump.

(e) For component cooling water pumps, a minimum flow of 10 gpm per pump is required for stuffing box cooling flushing. Additional flow is required for lube oil cooling and is controlled by throttling to maintain lube oil temperature.

(f) Flow rates are minimum required following a LOCA. Flow rates during normal operation are higher, depending on system alignment.

(g) Three pumps are desirable. Loss of one extends cooldown time but does not create an unsafe condition.

(h) The waste concentrator package, boric acid evaporator package, and auxiliary steam vent condenser have been abandoned in place and are currently isolated from the CCW system.

(i) During core offload

TABLE 9.2-6

COMPONENTS WITH A SINGLE BARRIER BETWEEN COMPONENT COOLING WATER AND REACTOR COOLANT WATER

	Barrier Design Temperature, °F	Barrier Design Pressure, psig	Temperature Range of Reactor Coolant Water, °F	Pressure Range of Reactor Coolant Water, psig
RHR heat exchangers	400	630	≤ 350	≤ 600
RHR pumps (seal coolers)	800	5140	≤ 350	≤ 600
Reactor coolant pumps(thermal barrier cooling coils)	650	2485	≤ 600	≤ 2485
Letdown heat exchanger	400	600	≤ 380	≤ 600
Excess letdown heat exchanger	650	2485	≤ 600	≤ 2485
Seal water heat exchanger	250	150	≤ 200	≤ 150
Sample heat exchangers	680	2485	≤ 652.7	≤ 2485

TABLE 9.2-7

COMPONENT COOLING WATER SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Consequences and Comments
1. Component cooling water pumps	Pump casing rupture	The casing is designed for 150 psig and 300°F which exceeds maximum operating conditions. Pump is inspectable and protected against credible missiles. Rupture is not considered credible. Standby pump provides redundancy.
2. Component cooling water pumps	Pump fails to start	Two operating pumps supply normal flow. Third standby provides redundancy.
3. Component cooling water pumps	Manual valve at pump suction or discharge closed or check valve sticks closed	This will be prevented by prestartup and operational checks. Further, periodic checks during normal operation would show that a valve was closed.
4. Component cooling water heat exchanger	Tube or shell rupture	Because of low operating pressures, rupture is considered incredible in normal or postaccident injection phase service; however, a leaking exchanger could be ascertained by sequential isolation or visual inspection. If a leak is in the on-line component cooling water heat exchanger, the standby exchanger would be put on stream and the leaking exchanger isolated and repaired. The leaking exchanger could be left in service with leakage up to the capacity of the makeup line to the system from the condensate storage tank, which is 250 gpm. During long-term postaccident recirculation the component cooling water system (CCWS) may be operated as two separate loops, either of which could sustain this failure with the other providing the minimum required engineered safety features.

Component	Malfunction	Consequences and Comments
5. General	Any leakage from system via open or drain valve severed loop piping, ruptured heat exchanger tube or other malfunction of equipment served by the system	Leakage from the CCWS can be detected by a falling level in the component cooling water surge tank. Observation of the time for the water level to fall a given amount and the area of the water surface in the tank will permit a determination of the leakage rate. The leaking component can be ascertained by sequential isolation or visual inspection of equipment in the system or complete isolation of a header. The 4000 gallons remaining in the surge tank after a makeup valve open alarm and the makeup flow will provide time for the closure of valves external to the containment for the isolation of all but the largest leaks. A 200 gpm maximum leak or rupture is postulated, so the operator has at least 20 minutes to isolate the leak before the surge tank is empty. The period is extended if the automatically operated, Design Class II, normal makeup path functions as designed and adds makeup water to the system. If the operator observes a rapidly falling surge tank level, he can elect to align/start a backup makeup water flow to the system before trying to isolate the leak (a backup source of water to the CCWS is provided from the condensate storage tank). The 250 gpm, Design Class I, makeup water flowpath, described in Section 9.2.3.3.7, can be started within 10 minutes. Because the makeup rate is greater than the postulated leak, the sequential isolation of components to stop the leak can proceed in an orderly manner. The 200 gpm leak is within the capacity of the auxiliary building drainage system. The complete rupture of large pipes or equipment with rapid loss of water is considered highly unlikely in normal operation or postaccident injection phase operation since all vital system piping and heat exchangers are located in the missile protected area of the containment. However, any of the three headers can be isolated. The two vital headers may be separated in accordance with EOP E-1.4 based on plant conditions during long-term postaccident operation and a passive failure in either would not impair the minimum engineered safety features supported by the other. Leaks into the CCW system can result from heat exchanger tube failure. For the RCP thermal barrier, the system design is to contain the in-leakage to the CCW system within the containment structure. The outboard containment isolation valve returning CCW from all reactor coolant pump thermal barriers closes on a high flow signal indicating in-leakage from the higher pressure RCS. All piping and valves

required to contain this in-leakage are designed for an RCS design pressure of 2485 psig. Should a coolant leak develop from the postulated failure mode that does not result in automatic flow isolation, the corresponding increase in CCW volume is accommodated by the relief valve on the CCW surge tank. The four relief valves on the CCW returns from thermal barriers are sized to relieve volumetric expansion and are set to relieve at RCS design pressure.

Tube failure in components with design pressures / temperatures less than RCS design condition can also initiate a leak into the CCW system. The radioactivity associated with the reactor coolant would actuate the CCW system radiation monitor (see Section 9.2.2.2.12). The monitor in turn would annunciate in the control room and close the vent valve located just upstream of the CCW surge tank back-pressure regulator to prevent the regulator from venting after sensing high radiation. The operator would also receive high level and high-pressure alarms from the surge tank as it filled. Continued in-leakage would increase the pressure in the surge tank until the high surge tank pressure alarm is actuated and the relief valve setpoint is reached. The relief valve on the surge tank protects the system from overpressurization. The maximum postulated in-leakage into the CCW system is based on an RHR heat exchanger tube rupture. The relief valve can accommodate this flow. Relief valve discharge from the CCW system surge tank is routed to the skirted area under the surge tank where it enters a floor drain routed to the auxiliary building sump.

TABLE 9.2-9 Sheet 1 of 3

MAKEUP WATER SYSTEM
EQUIPMENT DESIGN AND OPERATING PARAMETERS

Lime Tank[(a)]

Coagulant Tank[(a)]

Chemical Feed[(a)]

Mixed Bed Demineralizers Regeneration System

Caustic day tank

Number, total	1
Lining	Thermo-setting resin
Size, in.	42 (dia) x 48 (straight side)
Material	Carbon steel

Caustic storage tank[(a)]

Acid storage tank[(a)]

Miscellaneous equipment

Caustic dilution water heater	Electric, 125 kW @ 460 V
Caustic Feed pump	Diaphragm, 98 gph @ l30 psig

Reverse Osmosis (RO) System[(a)]

RO Auxiliary Equipment[(a)]

RO booster pumps (total)	2

RO prefilters

Hypochlorite feed

Acid feed

Hexametaphosphate feed

Dechlorination system

Makeup Water Degassifier[(a)]

Makeup Water Storage Tanks

Primary water storage tank

Number, per unit	1
Type	Vertical diaphragm sealed tank
Capacity, gal.	200,000
Size, ft	30 (dia) x 40 (straight side)
Material	304L stainless steel

Condensate storage tank

Number, per unit	1
Type	Vertical tank with floating roof
Capacity, gal.	425,000
Size, ft	40 (dia) x 47 (straight side)
Material	Carbon steel with reinforced concrete wall

Fire water and transfer storage tank - serves both units

Number	1
Type	Vertical dual compartment tank
Capacity - fire water, gal.	300,000
transfer, gal.	150,000
Size - fire water, ft	33 (dia) x 51 (straight side)
transfer, ft	40 (dia) x 51 (straight side)
Material	Carbon steel Inner tank - carbon steel Outer tank - carbon steel with reinforced concrete wall

Primary water makeup pumps

Number, per unit	2
Design pressure, psia	160
Design temperature, °F	70
Operating flow, gpm	150
Operating head, ft	222
Fluid	Water
Driver, hp	15
Material	316 stainless steel
Type	Vertical in-line

Makeup water transfer pumps

Number, shared	2
Design pressure, psia	210
Design temperature, °F	70
Operating flow, gpm	250
Operating head, ft	240
Fluid	Water
Driver, hp	30

TABLE 9.2-9 Sheet 3 of 3

Material	Class 30 cast iron
Type	Horizontal end suction

(a) These items are abandoned in place per administrative procedure. DCPP uses the MWTS.

TABLE 9.3-1

COMPRESSED AIR SYSTEM EQUIPMENT

A. INSTRUMENT AIR SYSTEM

Compressors	
Number	4
Type	Oil-free, single-stage, double-acting, reciprocating
Capacity	334 scfm each
Pressure	100 psig
Number	2
Type	Oil-free, two stage, water cooled, rotary screw
Capacity	650 scfm each
Pressure	110 psig
Number	1
Type	Oil-free, two stage, air cooled, rotary screw
Capacity	650 scfm each
Pressure	110 psig
Air Dryers	
Number	2
Type	Adsorbent, heat-regenerative (1) and heatless (1)
Capacity	1500 scfm each
Inlet pressure	100 psig
Inlet temperature	100°F
Exit dew point	-40°F at 100 psig and inlet air 100°F sat.
Pre-Filters	
Number	2
Type	Positive-seal
Capacity	2800 scfm each
Filtration	0.6 microns
After Filters	
Number	2
Type	Positive-seal
Capacity	3600 scfm each
Filtration	1 micron
Receivers	
Number	2
Capacity	650 cu ft each
Pressure	120 psig
Material	ASME SA 515 Gr. 70

B. SERVICE AIR SYSTEM

Compressors	
Number	1
Type	Oil-free, two stage, air-cooled, rotary screw
Capacity	650 scfm
Pressure	125 psig
Number	1
Type	Oil-free, two stage, air-cooled, rotary screw
Capacity	1050 scfm each
Pressure	125 psig
Air Dryers	
Number	3
Type	Heatless (2) and heat of compression (1)
Capacity	1400 scfm (each heatless), 1500 scfm (heat of compression)
Inlet pressure	100 psig
Exit dew point	-40°F
Pre-Filters	
Number	2
Type	Positive-seal
Capacity	2100 scfm
Filtration	1 micron
After Filters	
Number	2
Type	Positive-seal
Capacity	2400 scfm
Filtration	1 micron
Receiver	
Number	1
Capacity	750 cu ft
Pressure	125 psig

TABLE 9.3-2

NUCLEAR STEAM SUPPLY SYSTEM
SAMPLING SYSTEM COMPONENT DESIGN DATA

Sample Heat Exchanger

Number, per unit	3	
Design heat transfer rate (duty for 652.7°F sat. steam to 127°F liquid), each, Btu/hr	2.12×10^5	
	Shell	Tube
Design pressure, psig	150	2485
Design temperature, °F	350	680
Design flow, gpm	14.1	0.42
Temperature, in, °F	95	652.7 (max)
Temperature, out, °F	125	127 (max)
Fluid	Component cooling water	Sample
Material	Carbon steel	Austenitic stainless steel

Sample Pressure Vessel

Number, per unit	4
Volume, ml	150
Design pressure, psig	2485
Design temperature, °F	680
Material	Austenitic stainless steel

TABLE 9.3-5

CHEMICAL AND VOLUME CONTROL SYSTEM DESIGN DATA

General:	
Seal water supply flow rate, four pumps, nominal, gpm	32
Seal water return flow rate, four pumps, nominal, gpm	12
Letdown flow:	
Normal, gpm	75
Minimum, gpm	45
Maximum, gpm	120
Charging flow (excludes seal water):	
Normal, gpm	55
Minimum, gpm	25
Maximum, gpm	100
Temperature of letdown reactor coolant entering system, °F	545
Temperature of charging flow directed to reactor coolant system, °F	495
Centrifugal charging pump (CCP1 and 2) bypass flow (each), gpm	60
Centrifugal charging pump (CCP3) bypass flow (each), gpm	50
Amount of 4% boric acid solution required to meet design basis shutdown requirements, gal. (fuel cycle dependent)	14,042
Maximum pressurization required for hydrostatic testing of reactor coolant system, psig	3107

TABLE 9.3-6

CHEMICAL AND VOLUME CONTROL SYSTEM
PRINCIPAL COMPONENT DATA SUMMARY

Centrifugal Charging Pumps CCP1 and CCP2	
Number, per unit	2
Design pressure, psig	2800
Design temperature, °F	300
Design flow, gpm	150
Design head, ft	5800
Material	Austenitic stainless steel
Centrifugal Charging Pump CCP3	
Number, per unit	1
Design pressure, psig	3200
Design temperature, °F	300 [(a)]
Design flow, gpm	150
Design head, ft	5700
Material	Austenitic stainless steel
Boric Acid Transfer Pump	
Number, per unit	2
Design pressure, psig	150
Design temperature, °F	250
Design flow, gpm	75
Design head, ft	235
Material	Austenitic stainless steel
Gas Stripper Feed Pumps	
Number, per unit	2
Design pressure, psig	150
Design temperature, °F	200
Design flow, gpm	30
Design head, ft	320
Material	Austenitic stainless steel
Liquid Holdup Tank Recirculation Pump	
Number, per unit	1
Design pressure, psig	75
Design temperature, °F	200
Design flow, gpm	500
Design head, ft	100
Material	Austenitic stainless steel

Boric Acid Reserve Tank Pumps	
Number, shared	2
Design pressure, psig	150
Design temperature, °F	200
Design flow, gpm	150
Design head, ft	200
Material	Austenitic stainless steel
Boric Acid Reserve Tank Recirculation Pump	
Number, shared	2 per tank
Design pressure, psig	300
Design temperature, °F	400
Design flow, gpm	25
Design head, ft.	100
Material	Austenitic stainless steel
Concentrates Holding Tanks 0-1,0-2 Transfer Pumps (abandoned in place)	
Regenerative Heat Exchanger	
Number, per unit	1
Heat transfer rate at design conditions, Btu/hr	10.3×10^6
Shell side	
Design pressure, psig	2485
Design temperature, °F	650
Fluid	Borated reactor coolant
Material	Austenitic stainless steel
Tube side	
Design pressure, psig	2735
Design temperature, °F	650
Fluid	Borated reactor coolant
Material	Austenitic stainless steel
Shell side (Letdown)	
Flow, lb/hr	37,050
Inlet temperature, °F	545
Outlet temperature, °F	290
Tube Side (Charging)	
Flow, lb/hr	27,170
Inlet temperature, °F	130
Outlet temperature, °F	495

Letdown Heat Exchanger

Number, per unit	1	
Heat transfer rate at design conditions, Btu/hr	14.8×10^6	
Shell side		
Design pressure, psig	150	
Design temperature, °F	250	
Fluid	Component cooling water	
Material	Carbon steel	
Tube side		
Design pressure, psig	600	
Design temperature, °F	400	
Fluid	Borated reactor coolant	
Material	Austenitic stainless steel	
Shell side	Design	Normal
Flow, lb/hr	492,000	203,000
Inlet temperature, °F	95	95
Outlet temperature, °F	125	125
Tube Side (Letdown)		
Flow, lb/hr	59,280	37,050
Inlet temperature, °F	380	290
Outlet temperature, °F	127	127

Excess Letdown Heat Exchanger

Number, per unit	1	
Heat transfer rate at design conditions, Btu/hr	4.61×10^6	
	Shell Side	Tube Side
Design pressure, psig	150	2485
Design temperature, °F	250	650
Design flow, lb/hr	115,000	12,380
Inlet temperature, °F	95	545
Outlet temperature, °F	135	195
Fluid	Component cooling water	Borated reactor coolant
Material	Carbon steel	Austenitic stainless steel

Seal Water Heat Exchanger

Number, per unit	1	
Heat transfer rate at design conditions, Btu/hr	2.49×10^6	
	Shell side	Tube Side
Design pressure, psig	150	150
Design temperature, °F	250	250
Design flow, lb/hr	99,500	160,600
Inlet temperature, °F	95	143
Outlet temperature, °F	120	127
Fluid	Component cooling water	Borated reactor coolant
Material	Carbon steel	Austenitic stainless steel

Volume Control Tank

Number, per unit	1
Volume, ft^3	400
Design pressure, psig	75
Design temperature, °F	250
Spray nozzle flow (maximum), gpm	120
Material	Austenitic stainless steel

Boric Acid Tank

Number, per unit	2
Capacity, gal.	8,060
Design pressure	Atmospheric
Design temperature, °F	180
Material	Austenitic stainless steel

Boric Acid Batching Tank

Number, shared	1
Capacity, gal.	800
Design pressure	Atmospheric
Design temperature, °F	300
Material	Austenitic stainless steel

Chemical Mixing Tank

Number, per unit	1
Capacity, gal.	5
Design pressure, psig	150
Design temperature, °F	200
Material	Austenitic stainless steel

Liquid Holdup Tanks

Number, shared	5
Volume, gal.	83,220
Design pressure, psig	15
Design temperature, °F	200
Material	Austenitic stainless steel

Boric Acid Reserve Tanks

Number, shared	2
Capacity, gal.	24,610
Design pressure	Atmospheric
Design temperature, °F	150
Material	Austenitic stainless steel with membrane seal

Concentrates Holding Tank (abandoned in place)

Mixed Bed Demineralizers

Number, per unit	2
Design pressure	200
Design temperature, °F	250
Design flow, gpm	120
Normal resin volume, each, ft^3	30
Maximum resin volume, each, ft^3 (flush only)	39
Material	Austenitic stainless steel

Boric Acid Reserve Tank Recirculation Heater

Number, shared	1 per tank
Design pressure, psig	150
Design temperature, °F	500
Design flow, gpm	25
Material	Austenitic stainless steel

Cation Bed Demineralizer

Number, per unit	1
Design pressure, psig	200
Design temperature °F	250
Design flow, gpm	120
Resin volume, each, ft^3	27
Material	Austenitic stainless steel

Deborating Demineralizers

Number, per unit	2
Design pressure	200
Design temperature, °F	250
Design flow, gpm	120
Normal resin volume, each, ft^3	30
Maximum resin volume, each, ft^3 (flush only)	39
Material	Austenitic stainless steel

Evaporator Feed Ion Exchangers

Number, per unit	4
Design pressure, psig	200
Design temperature, °F	250
Design flow, gpm	30
Resin volume (maximum), ft^3	16
Material	Austenitic stainless steel

Evaporator Condensate Demineralizers (abandoned in place)

Reactor Coolant Letdown Filters

Number, per unit	2
Design pressure, psig	200 (1-1, 2-1), 300 (1-2, 2-2)
Design temperature, °F	250
Design flow, gpm	150 (1-1, 2-1), and 250 (1-2, 2-2)
Particle retention	98% of 25 micron size 100% of 50 micron size
Material, vessel	Austenitic stainless steel

<u>Seal Water Injection Filters</u>

Number, per unit	2
Design pressure, psig	2,735
Design temperature, °F	200
Design flow, gpm	80
Particle retention	98% of 5 micron size
Material, vessel	Austenitic stainless steel

<u>Seal Water Return Filter</u>

Number, per unit	1
Design pressure, psig	200
Design temperature, °F	250
Design flow, gpm	150
Particle retention	98% of 25 micron size
Material, vessel	Austenitic stainless steel

<u>Boric Acid Filter</u>

Number, per unit	1
Design pressure, psig	200
Design temperature, °F	250
Design flow, gpm	150
Particle retention	5 micron or less
Material, vessel	Austenitic stainless steel

<u>Ion Exchanger Filter</u>

Number, per unit	1
Design pressure, psig	200
Design temperature, °F	250
Design flow, gpm	35
Particle retention	98% of 25 micron size
Material, vessel	Austenitic stainless steel

<u>Condensate Filter</u> (abandoned in place)

Concentrates Filter (abandoned in place)

Boric Acid Blender

Number, per unit	1
Design pressure, psig	150
Design temperature, °F	250
Material	Austenitic stainless steel

Letdown Orifice	45 gpm	75 gpm
Number, per unit	1	2
Design flow, lb/hr	22,230	37,050
Differential pressure at design flow, psia	1,900	1,900
Design pressure, psig	2,500	2,500
Design temperature, °F	650	650
Material	Austenitic stainless steel	Austenitic stainless steel

Gas Stripper - Boric Acid Evaporator Package (abandoned in place)

(a) The design temperature of 300°F is for the structural integrity of the pump but not the limitation for operation.

TABLE 9.3-7

NITROGEN REQUIREMENTS

Equipment	Supply Pressure, psig	Supply Flow, scfm	Comments
Pressurizer relief tank	3	8.5	Initial gas cover
Unit 1 Reactor coolant drain tank	<15	20 (27 max.)	Gas cover/purge (refueling outage)
Unit 2 Reactor coolant drain tank	<15	20 (40 max.)	Gas cover/purge
Volume control tank	18	8.3	Initial gas cover (2950 scf total for all 3 tanks above, per refueling cycle)
Spray additive tank	5	5	
Gas decay tanks	5	15	
Waste concentrator condenser	2	10	(abandoned in place)
Boric acid concentrator	2	10	(abandoned in place)
Boric acid tanks	1	10	
Concentrates holding tank 0-1	1	6	(abandoned in place)
Concentrates holding tank 0-2	1	8	(abandoned in place)
Liquid holdup tank	3	4 (70 max.)	
Accumulators	650	22	103,000 scf initial fill, based on 700 psig and 40% gas space per refueling cycle
Boric acid reserve tanks	1.5	1	
Various replenishment requirements			Degasification purging during cold shutdown, 8400 scf per cold shutdown
Nitrogen layup system for steam generators			400 lb of nitrogen gas required to purge 4 main steam lines and provide initial charge of nitrogen on steam generators at 5 psig

Equipment	Supply Pressure, psig	Supply Flow, scfm	Comments
Nitrogen layup for 12 feedwater heaters			Same requirement as for steam generators except need 760 lb of nitrogen
CCW surge tank pressurization with N_2	20	25 (max)	Intermittent (to maintain surge tank pressure)
Condenser	30 (max)	20	Continuous
Zinc Tank	2” H_2O	<1	Intermittent (to maintain N_2 blanket in the zinc tank)

TABLE 9.3-8

HYDROGEN REQUIREMENTS

Equipment	Supply Pressure, psig	Supply Flow, scfm	Comments
Generator	75 (max.)	200 (makeup)	Initial fill of 34,390 scf
Volume control tank	15 to 26	8.3	960 scf required for purge and initial fill

TABLE 9.4-1 Sheet 1 of 2

CONTROL ROOM VENTILATION SYSTEM (CRVS) COMPONENT DESIGN DATA

Filter Unit (per unit)

	No.	Size, in.	Efficiency, %[d]	Rated Air Flow Per Filter, cfm
Roughing filters	2	24 x 24 x 6[g]	80-85[a]	1000 @ 0.30"SP
HEPA filters	2	24 x 24 x 11-1/2	99.97[b]	1150 @ 1.00"SP
Charcoal filters (cells or trays)	6	26 x 26 x 6	99/85[c]	333 @ 1.3"SP

Charcoal Filter Humidity Control Heater (per unit)

	No.	Capacity, kW	Voltage, V
Electric heater	2	5.0	480

Fans (per unit)

	No.	Static Pressure, in H_2O	Air Flow, cfm	Motor, hp
Main supply	2	1.75	7800	7.5
Filter booster	2	5.40	2100	3
Exhaust	1	1.00	1700	3/4
Pressurization[e]	2	5.00	2100	7.5

Air Conditioning Units (per unit)

Two full-capacity air conditioning units are provided, each consisting of a reciprocating compressor, cooling coil, and fan air-cooled condenser. The refrigeration capacity for each condenser is 31.0 tons using Freon 22 at 37°F suction temperature and 107°F condenser temperature. The cooling coils are each capable of handling 7800 cfm of air at 400 fpm face velocity. The air conditions for the cooling coil are:

Entering air: 84.0°F D.B./64.8°F W.B.
Leaving air: 54°F D.B./53°F W.B.
Friction drop through the coil: 0.41 in. H_2O

Airborne Contaminant Detectors (per unit)

	No.	Location	Sensitivity
Smoke	2	Return air duct	Trace amounts of combustion products
	2	Control room normal intake	Trace amounts of combustion products
Radioactivity	2	Control room normal intake	1 x 10^{-2} to 10^{+3} mR/hr
	2	Pressurization intakes	10^{-2} mR/hr (Sensitivity range is 10^{-2} to 10^{+4} mR/hr with setting point of 2 mR/hr)
Chlorine(f)	-	-	-

(a) Minimum efficiency requirements. Efficiency based on American Filter Institute (AFI) dust spot test

(b) Based on standard DOP test 0.3 micron particles

(c) Radioactive elemental iodine and radioactive iodide as methyl iodide, respectively (Efficiency rates are for filters as originally specified. Replacement filters shall comply with the requirements of the Ventilation Filter Test Program.

(d) Efficiency rates for HEPA and charcoal filters are for individual components (for filter efficiency rates during a DBA, refer to the respective accident analysis in Chapter 15)

(e) All four fans are common to both Unit 1 and Unit 2

(f) Chlorine monitors are abandoned in place or removed as there is no bulk chlorine onsite

(g) Table shows original filter size. Replacement filter depth dimension may vary.

COMPLIANCE WITH REGULATORY GUIDE 1.52, JUNE 1973
DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR ATMOSPHERE CLEANUP SYSTEM
AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER COOLED NUCLEAR POWER PLANTS

Regulatory Position	Compliance	Reasons or Comments
2. SYSTEM DESIGN CRITERIA		
a. Redundant systems and required components	Yes, except:	
	(1) No demisters.	(1) Demisters are not provided since they are not required for this filtration system design. In the control room, the outside air intake plenum is designed to present a tortuous path for flow in order to remove entrained water. At the time of operating the hydrogen vent system, several weeks after the DBA, no moisture will be entrained in the containment. Filters in the fuel handling building and the auxiliary building are not in direct contact with outside air.
	(2) No HEPA after-filters.	(2) Charcoal is thoroughly cleaned before insertion in filter trays. Formation of charcoal fines is not considered significant.
	(3) Auxiliary building ventilation system does not have redundant charcoal filters.	(3) As discussed in Section 15.5.17, failure of an RHR pump seal was already assumed as the single failure, an additional failure of the auxiliary building charcoal filter need not be postulated. Therefore, the charcoal filters need not be redundant.
	(4) Auxiliary building ventilation system does not have redundant electric heaters. Fuel handling building ventilation system does not have electric heaters (refer to 3.b below).	(4) Installation of one electric heater for the auxiliary building ventilation system is consistent with the commitments made as discussed in Section 15.5.17.

Regulatory Position	Compliance	Reasons or Comments
2. SYSTEM DESIGN CRITERIA		
b. Physical separation, including missile protection	Yes, except some ventilation systems in the plant are not specifically designed against tornadoes (refer to Section 3.3.2) or against local damage of ducting by missiles. However, the control room positive pressurization system is specifically designed for protection against tornado missiles (Refer to Section 9.4.1.3.1). The remainder of the control room ventilation system is enclosed within reinforced concrete structures.	No damage to components essential to safe plant shutdown or to protection of the public.
d. Protection against pressure surges	Not applicable.	Auxiliary building ventilation system not specifically designed for pressure surge due to failure of gas decay tank. Since insignificant amounts of radioiodine are contained in the tank, loss of this system will not affect offsite exposures.
f. Maximum air flow rate per train and preferred filter array	Air flow rate for the fuel handling building ventilation system is 35,750 cfm and auxiliary building ventilation system is 73,500 cfm versus recommended 30,000 cfm.	In either case, filters are easily replaced. Structural platforms and traveling hoists are permanently installed in filter rooms. In only one case are filters aligned more than three high above the floor or a platform.
g. Flow and ΔP signal, alarm and record in control room	No control room recorders or alarms.	Systems are tested at least once each 31 days when required to be operable per Technical Specifications. Fan status and damper position indicator lights are provided in the control room for the auxiliary and fuel handling buildings ventilation systems. The control room HVAC system is provided with a system trouble alarm in the control room.
j. Total enclosure and intact replacement	Not replaceable intact.	Each filter train consists of a totally enclosed unit, which cannot be replaced intact due to the as-built dimensions of the filter train room and access doors. Filter train components can be individually replaced.

Regulatory Position	Compliance	Reasons or Comments
3. COMPONENT DESIGN CRITERIA & QUALIFICATION TESTING		
a. Demisters quality to MSAR 71-45 & UL Class I	Not applicable.	No demisters.
b. Heaters to reduce RH to 70% under DBA	The fuel handling building ventilation system does not have electric heaters. Heaters are installed for the control room and the auxiliary building ventilation systems.	Heaters not required for the fuel handling building, control room, and auxiliary building ventilation systems. Charcoal samples are tested to 95% RH. Relative humidity of the exhaust air of these systems is not expected to exceed the tested condition.
4. MAINTENANCE		
i. Entire standby atmosphere cleanup train should be operated at least 10 hours per month, with the heaters on (if so equipped), in order to reduce the buildup of moisture.	No.	Site climate should not lead to an excessive buildup of moisture on the filters; however, at least once per 31 days, when required by the Technical Specifications for operability, flow is initiated through the filters for at least 15 minutes to verify flow (refer to Section 9.4.1.4.2)
5. IN-PLACE TESTING CRITERIA		
b. In-place penetration of HEPA filters should conform to ANSI N101.1-1972.	No. Visual inspection, in-place penetration and bypass leakage testing of HEPA filter performed in accordance with ANSI N510-1980.	The in-place testing criteria are established in the Technical Specifications. The 1980 revision of ANSI N510 encompasses the testing criteria required by ANSI N101.1-1972.
HEPA filter sections should be tested in place initially and semiannually thereafter, to confirm a penetration of less than 0.05% at rated flow.	No. An in place test of the CRVS, ABVS and FHBVS HEPA filters shows a penetration and system bypass < 1.0% when tested at least once per 24 months.	The in-place testing criteria are established in the Technical Specifications.
c. Adsorber banks should be leak tested in accordance with USAEC Report DP-1082.	No. Visual inspection, in-place penetration and bypass leakage testing of adsorber banks for CRVS, ABVS and FHBVS is performed in accordance with ANSI N510-1980.	The in-place testing criteria are established in the Technical Specifications. The 1980 revision of ANSI N510 encompasses the testing criteria required by USAEC Report DP-1082.

Regulatory Position	Compliance	Reasons or Comments
Adsorber leak testing should be conducted whenever DOP testing is done.	No. An in place test of the CRVS, ABVS and FHBVS charcoal adsorbers shows a penetration and system bypass < 1.0% when tested at least once per 24 months.	The in-place testing criteria are established in the Technical Specifications.
6. LABORATORY TESTING CRITERIA FOR ACTIVATED CARBON		
a. Activated carbon adsorber section should be assigned the decontamination efficiencies given in Table 2.	No. The control room ventilation system decontamination efficiencies assumed in the accident analysis are 93% / 93% for elemental and organic iodine respectively. The control room HVAC system charcoal samples are tested at 30°C/95% RH per ASTM D3803-89 with a 2.5% acceptance criteria versus using RDT M16-1T (1972) at DBA conditions.	The laboratory test acceptance criteria are established in the Ventilation Filter Test Program, which is controlled by the Technical Specifications. Testing per RDT M16-1T (1972) has been superseded by the more conservative test requirements in ASTM D3803-1989. In accordance with Generic Letter 99-02, June 1999 a safety factor of 2 is used in determining the charcoal filter efficiency for use in safety analyses.
	The auxiliary building ventilation system decontamination efficiencies assumed in the accident analysis are 88%/88% for elemental and organic iodine, respectively, versus 95%/95% as assigned by Regulatory Guide 1.52, June 1973 Table 2. Additionally, the charcoal samples are tested at 30°C/95% RH per ASTM D3803-89 with a 5% acceptance criterion versus using RDT M16-1T (1972) at DBA conditions.	The decontamination efficiencies used in the accident analysis are more conservative than the values stated in Regulatory Guide 1.52, June 1973. The laboratory test acceptance criteria are established in the Technical Specifications. Testing per RDT M16-1T (1972) has been superseded by the more conservative test requirements in ASTM D3803-1989. In accordance with Generic Letter 99-02, June 1999 a safety factor of 2 is used in determining the charcoal filter efficiency for use in safety analyses.
	The fuel handling building ventilation system decontamination efficiencies are 95%/95% as assigned by Regulatory Guide 1.52, June 1973 Table 2. Refer to Table 15.5-45 for a discussion of assumed efficiencies in the accident	The decontamination efficiencies used in the accident analysis are more conservative than the values stated in Regulatory Guide 1.52, June 1973. The laboratory test acceptance criteria are established in the Technical Specifications. Testing per RDT M16-1T (1972) has been superseded by the more Conservative test requirements in ASTM D3803-1989.

Regulatory Position	Compliance	Reasons or Comments
	analysis. Additionally, the charcoal samples are tested at 30°C/95% RH per ASTM D3803-89 with a 15% acceptance criterion versus using RDT M16-1T (1972) at DBA conditions.	

TABLE 9.4-5

AUXILIARY BUILDING VENTILATION SYSTEM
COMPONENTS DESIGN DATA

Filters, per unit

	No.	Size, in	Efficiency, %[d]	Rated Air Flow Per Filter, cfm
Supply roughing[e]	70	24x24x26	30 (minimum)[e]	2,000 @ 0.17" SP
Exhaust roughing	172	24x24x12	80-85[a]	2,000 @ 0.55" SP
Exhaust HEPA	273	24x24x 11-1/2	99.97[b]	1,150 @ 1.00" SP
Exhaust charcoal (module composed of 3 cells or trays)	77	27-5/32 x 26-3/8 x 28-1/2	99/85[c]	1,000 @ 1.3" SP (Air enters from rear of tray)

Fans, per unit

	No.	Max. Static Pressure H_2O, in	Air Flow, cfm	Motor, hp
Supply (S31 & S32, Unit 1) (S33 & S34, Unit 2)	2	2.80	67,500	60
Exhaust (1E1 & 1E2,Unit 1) (2E1 & 2E2,Unit 2)	2	10.00	73,500	150

Electric Heaters, per unit

	No.	Rating
Exhaust (1EH-30, Unit 1) (2EH-30, Unit 2)	1	54 kW, 480 V, 3d

(a) Minimum efficiency requirements (Efficiency based on National Bureau of Standard (NBS) methods, dust spot test.)

(b) Based on standard DOP test 0.3 micron particles

(c) Radioactive elemental iodine and radioactive iodide as methyl iodide, respectively. Efficiency rates are for filters as originally specified. Replacement filters shall comply with the requirements of Regulatory Guide 1.52, June 1973 and ANSI N509.

(d) Efficiency rates for HEPA and charcoal filters are for individual components.

(e) Average dust spot efficiency requirement (Efficiency based on ASHRAE Standard 52.1 test)

(f) Supply filters are nominal values and may not reflect the as-built conditions.

TABLE 9.4-6

FUEL HANDLING BUILDING VENTILATION SYSTEM
COMPONENTS DESIGN DATA

Filters, per unit

	No.	Size, in.	Efficiency, %	Rated Air Flow Per Filter, cfm
Supply roughing	21	24x24x13	30[a]	2,500 @ 0.35" SP
Exhaust roughing	120	24x24x12	80-85[a]	2,000 @ 0.55" SP
Exhaust HEPA	107	24x24x 11-1/2	99.97[b]	1,150 @ 1.00" SP
Exhaust charcoal (module composed of 3 cells or trays)	72	27-5/32 x 26-3/8 x 28-1/2	99/85[c]	1,000 @ 1.3" SP (Air enters from rear of tray)

Fans, per unit

	No.	Max. Static Pressure H_2O, in.	Air Flow, cfm	Motor, hp
Supply (S1 & S2)	2	2.90-3.30	23,300	25
Exhaust (E4)	1	6.75	35,750	75
(E5 & E6)	2	7.55	35,750	75

Heating Coils, per unit

Heating Capacity, Btu/hr	Area, sq ft	Air Flow, cfm	Entering Temp., °F	Leaving Temp., °F	Steam Press., psig
1,400,000	61.8	23,300	35	70	15

(a) Minimum efficiency requirements (Efficiency based on National Bureau of Standards (NBS) methods, dust spot test.)

(b) Based on standard DOP test 0.3 micron particles

(c) Radioactive elemental iodine and radioactive iodide as methyl iodide, respectively (Efficiency rates are for filters as originally specified. Replacement filters shall comply with the requirements of Regulatory Guide 1.52 and ANSI N509.)

TABLE 9.4-7

DESIGN VALUES FOR TURBINE BUILDING (GENERAL AREA) VENTILATION SYSTEM,

Turbine building volume, ft^3	5,125,000
15 Supply fans rating, cfm/fan	28,000
4 Exhaust fans rating, cfm/fan	40,000

TABLE 9.4-8

DESIGN CODES AND STANDARDS FOR VENTILATION SYSTEMS

Building or Area	Code
Control room	State of California, Industrial Safety Orders, Title 8, Sub-Chapter 7
	American Society of Heating, Refrigerating and Air Conditioning Engineers (ASHRAE) Guide
	Sheet Metal, Air Conditioning Contractors' National Association (SMACNA) Code
	Air Movement and Control Association (AMCA) - Standards for Air Moving Devices
Auxiliary building (excluding fuel handling area)	State of California, Industrial Safety Orders, Title 8, Sub-Chapter 7
	ASHRAE Guide
	SMACNA Code
	AMCA - Standards for Air Moving Devices
	American Conference of Governmental Industrial Hygienists - Industrial Ventilation Manual
Turbine building	State of California, Industrial Safety Orders, Title 8, Sub-Chapter 7
	ASHRAE Guide
	SMACNA Code
	AMCA - Standards for Air Moving Devices
Fuel handling area of the auxiliary building	State of California, Industrial Safety Orders, Title 8, Sub-Chapter 7
	ASHRAE Guide
	SMACNA Code
	AMCA - Standards for Air Moving Devices

Building or Area	Code
Containment	State of California, Industrial Safety Orders, Title 8, Sub-Chapter 7
	ASHRAE Guide
	SMACNA Code
	AMCA - Standards for Air Moving Devices
Auxiliary saltwater pump vaults	State of California, Industrial Safety Orders, Title 8, Sub-Chapter 7
	ASHRAE Guide
	SMACNA Code
	AMCA - Standards for Air Moving Devices
125-Vdc/480-V switchgear area	State of California, Industrial Safety Orders, Title 8, Sub-Chapter 7
	ASHRAE Guide
	SMACNA Code
	AMCA - Standards for Air Moving Devices
4.16-kV Switchgear room	State of California, Industrial Safety Orders, Title 8, Sub-Chapter 7
	ASHRAE Guide
	SMACNA Code
	AMCA - Standards for Air Moving Devices
Post-accident sample room	ASHRAE Guide
	SMACNA Code
	AMCA - Standards for Air Moving Devices

TABLE 9.4-8

Building or Area	Code	
Technical support center	ASHRAE Guide	
	SMACNA Code	
	AMCA - Standards for Air Moving Devices	
Containment Penetration Area GE/GW	AMCA 99 (1972) -	Standards Handbook
	AMCA 210 (1974) -	Test Code For Air Moving Devices
	AMCA 500 (1975) -	Test Methods for Louvers, Dampers and Shutters
	ANSI N509-1980 -	Nuclear Power Plants Air Cleaning Units and Components
	SMACNA -	HVAC Duct Construction Standards 1985
	SMACNA -	Round Industrial Duct Construction Standard

TABLE 9.4-9

ESTIMATED CONTROL ROOM AREA HEAT LOADS[a]

(NORMAL OPERATING CONDITIONS - MODE 1)

Area	Btu/hr	
Control Room		
Floor and wall	14,990	
Occupants	3,200	
Lighting	35,495	
Annunciators, control boards, and nuclear instrumentation	109,365	
Computer Room		
Floor and wall	8,900	
Occupants	1,280	
Lighting	8,765	
Computers and analog control system	3,865	67,580[b]
Record Storage and Office		
Floor and wall	3,450	
Occupants	1,150	
Lighting	5,300	
Safeguard Room		
Floor and walls	2,545	
Occupants	640	
Lighting	3,030	
Solid state protection system	18,770	
Control Room Area Outside Air	2,835	
Total Heat Loads	223,580	
Total Tonnage - 19.0 Tons		

(a) All loads are given for Unit 1. Unit 2 heat loads are considered the same. The information contained in this table is "representative" of Units 1 and 2. Refer to the applicable design calculations for the current heat loads. This table will not be revised to reflect current design bases heat loads.

(b) Unit 1 computer room cooling load handled by supplemental computer room air conditioning units.

TABLE 9.4-10 Sheet 1 of 2

DESIGN VALUES FOR AUXILIARY BUILDING VENTILATION SYSTEM[(a)]

Auxiliary Building Ventilation System (per unit)

Mode 1: Building Ventilation

1 Supply fan	Rating, cfm/fan	67,500
1 Exhaust fan	Rating, cfm/fan	73,500

Item	Heat Load, Btu/hr
Lighting	286,500
Equipment, piping, and cables	780,200
Electric motors	526,300
TOTAL	1,593,000

Mode 2: Building and Engineered Safety Ventilation

2 Supply fans	Rating, cfm/fan	67,500
2 Exhaust fans	Rating, cfm/fan	73,500

Item	Heat Load, Btu/hr
Lighting	286,500
Equipment, piping, and cables	1,194,000
Electric motors	1,271,200
TOTAL	2,751,700

Mode 3: Engineered Safety Ventilation

1 Supply fan	Rating, cfm/fan	67,500
1 Exhaust fan	Rating, cfm/fan	73,500

Item	Heat Load, Btu/hr
Lighting	286,500
Equipment, piping, and cables	1,194,000
Electric motors	1,103,400
TOTAL	2,583,900

Fuel Handling Building Ventilation System (per unit)

Normal and Iodine Removal Modes (not including auxiliary feed pumps)

1 Supply fan	Rating, cfm/fan	23,300
1 Exhaust fan	Rating, cfm/fan	35,750
Item		Heat Load, Btu/hr
Lighting		114,700
Equipment, piping, and cables		245,500
Electric motors		209,000
TOTAL		569,200

Normal and Iodine Removal Modes (including auxiliary feed pumps)

1 Supply fan	Rating, cfm/fan	23,300
1 Exhaust fan	Rating, cfm/fan	35,750
Item		Heat Load, Btu/hr
Lighting		114,700
Equipment, piping, and cables		245,500
Electric motors		352,900
TOTAL		713,100

(a) The information contained in this table is “representative” of Units 1 and 2 and is provided for information only. Refer to the applicable design calculations for the current heat loads. This table will not be revised to reflect current design bases heat loads.

TABLE 9.4-11

ESTIMATED NSSS HEAT LOSSES INSIDE CONTAINMENT

Piping	Btu/hr (x 10^6)
Reactor Coolant System	0.09
Other Piping	0.04
Equipment	
Reactor Vessel[(a)]	
Above seal	0.037
Below seal	0.125
Reactor Coolant Pumps[(b)]	3.600
Steam Generators	0.800
Pressurizer[(c)]	0.133
Control Rod Drive Mechanisms[(d)]	2.260
Pressurizer Relief Tank	0.022
Primary Concrete Shield	0.015
Regenerative Heat Exchanger	0.022
Excess Letdown Heat Exchanger	0.010
Total	7.154
Contingency	1.106
Total[(e)]	8.260

(a) Does not include supports

(b) Each pump: motor - 0.75 x 10^6 Btu/hr; uninsulated section - 0.15 x 10^6 Btu/hr

(c) Includes supports and heater terminal connections

(d) Includes heat losses from control rod drive mechanisms, and control rod penetration housings

(e) The following heat losses have not been included

Heat losses from piping and equipment not considered part of NSSS
Daily and seasonal ambient temperature changes
Solar heat load
Fan input power

TABLE 9.4-12

ESTIMATED TOTAL HEAT SOURCES INSIDE CONTAINMENT

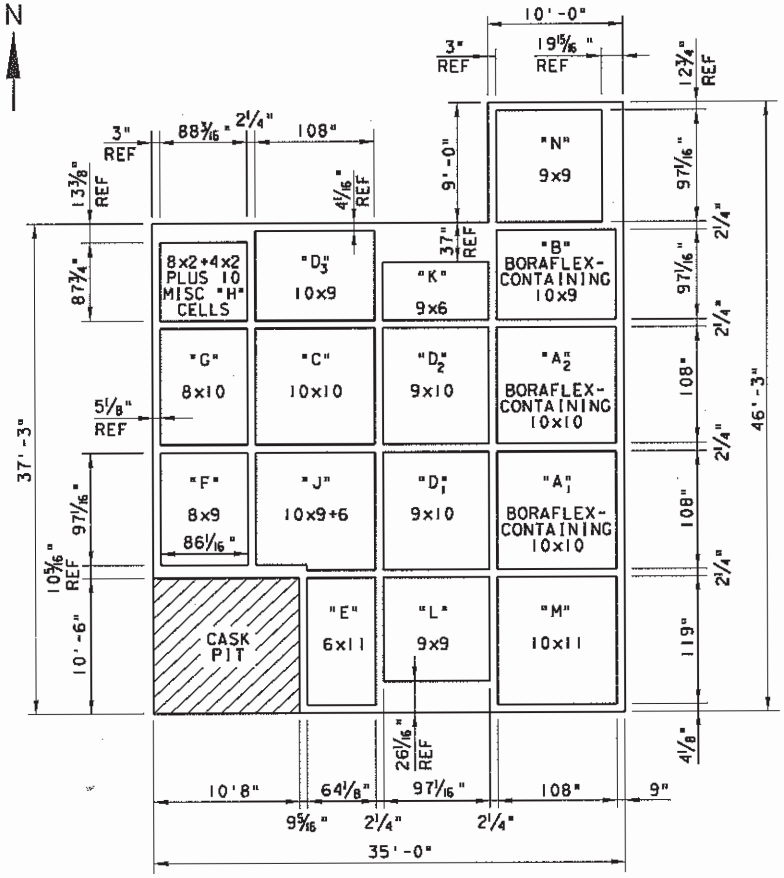
Heat Sources	Btu/hr (x 10^6)
a. Steam and feedwater lines	0.30
b. Solar	0.10
c. Control rod drive fans	0.35
d. Ventilation fans (4 running)	2.94
e. NSSS[(a)]	8.26
f. Contingency for heat sources a, b, c, & d	0.37
Total	12.32

(a) The assumed NSSS sources of heat losses are shown in Table 9.4-11

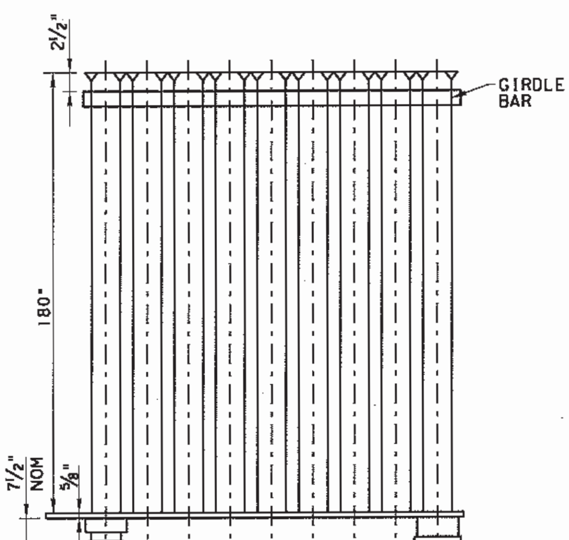
TABLE 9.5-2

DIESEL GENERATOR FUEL OIL SYSTEM COMPONENT DESIGN DATA

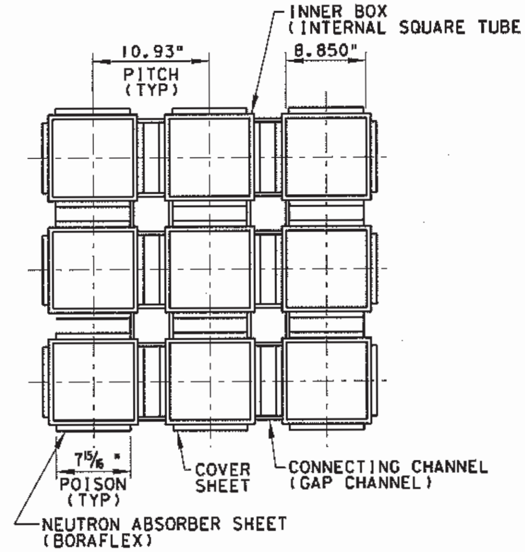
Storage Tanks	
Number, shared	2
Type	Horizontal, underground
Capacity, gal	50,000
Pressure	Atmospheric
Temperature	Ambient ground temperature
Material	Carbon steel/fiberglass
Transfer Pumps	
Number, shared	2
Type	Rotary, two-screw
Viscosity range, ssu	35 - 100
Rated capacity, gpm	58
Discharge pressure, psig	50
Suction pressure, in Hg	20
Casing material	Nodular iron
Motor horsepower	5



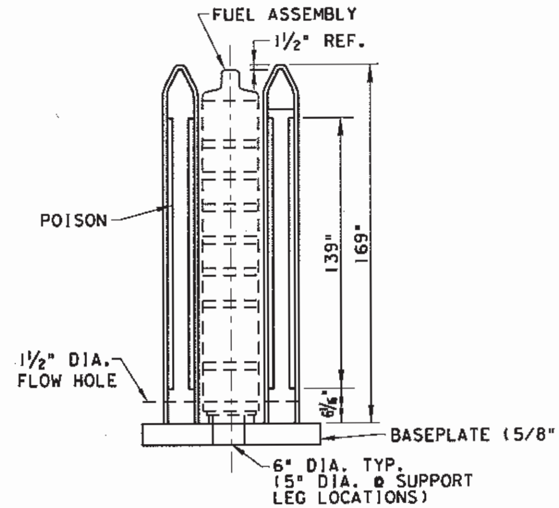
POOL LAYOUT
UNIT 1



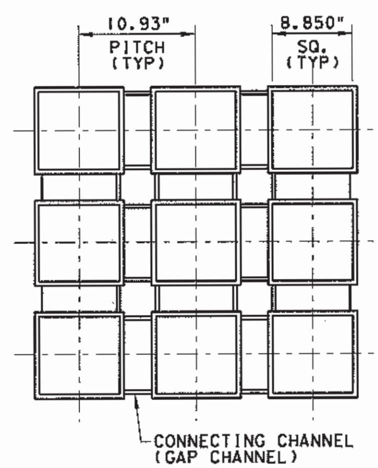
TYPICAL RACK ELEVATION



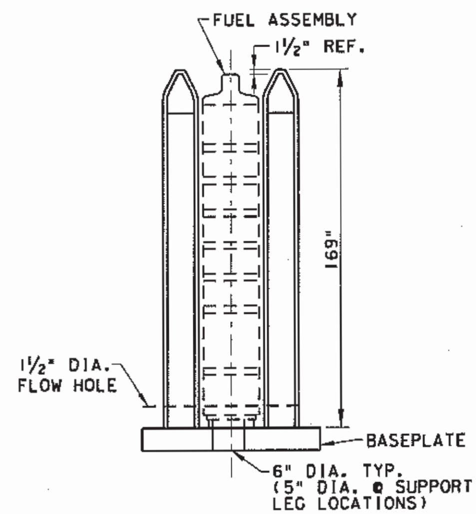
3 x 3 TYPICAL ARRAY
BORAFLEX - CONTAINING
(POISONED CELLS)



TYPICAL CELL ELEVATION
BORAFLEX - CONTAINING
(POISONED CELLS)



TYPICAL ARRAY
NON BORAFLEX - CONTAINING
(UNPOISONED CELLS)



TYPICAL CELL ELEVATION
NON BORAFLEX - CONTAINING
(UNPOISONED CELLS)

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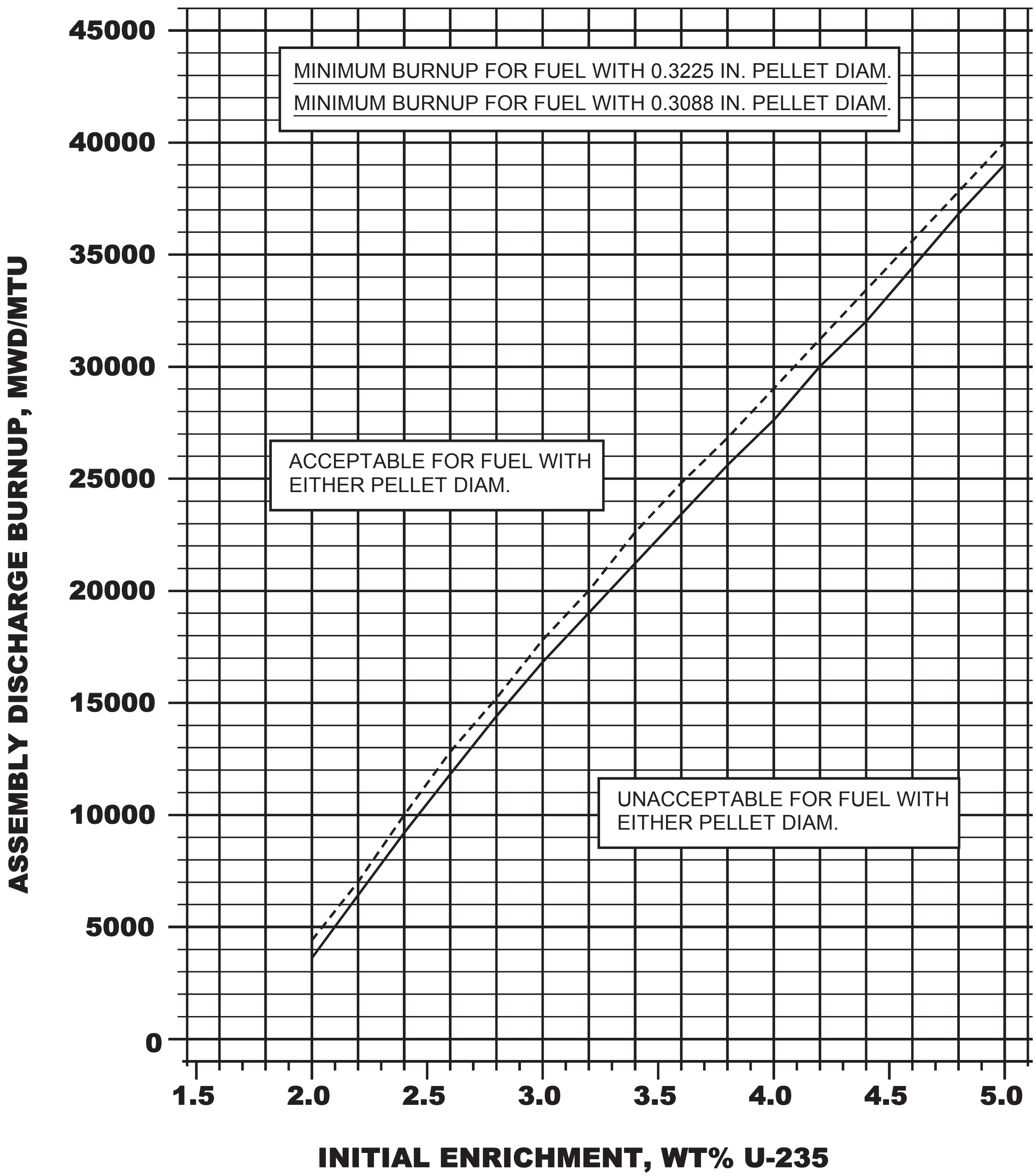
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.1-2
SPENT FUEL STORAGE

September 1988 Revision 4



MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP AS A FUNCTION OF INITIAL ENRICHMENT AND FUEL PELLET DIAMETER FOR AN ALL CELL STORAGE CONFIGURATION.

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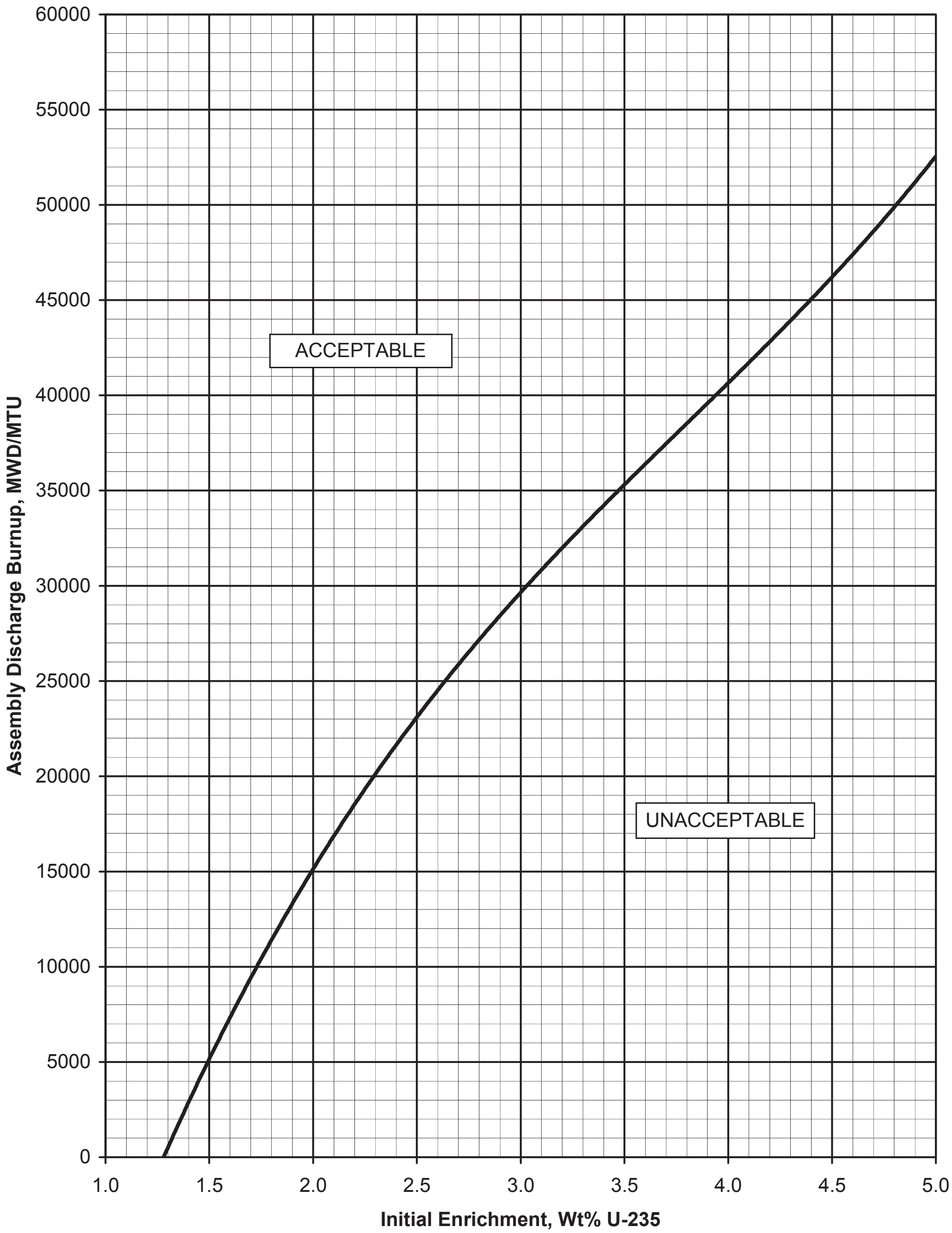
FIGURE 9.1-2A BURNUP VS. ENRICHMENT (ALL CELL)



MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT
FOR A 2X2 ARRAY STORAGE CONFIGURATION

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FIGURE 9.1-2B BURNUP VS. ENRICHMENT (2 X 2 ARRAY)

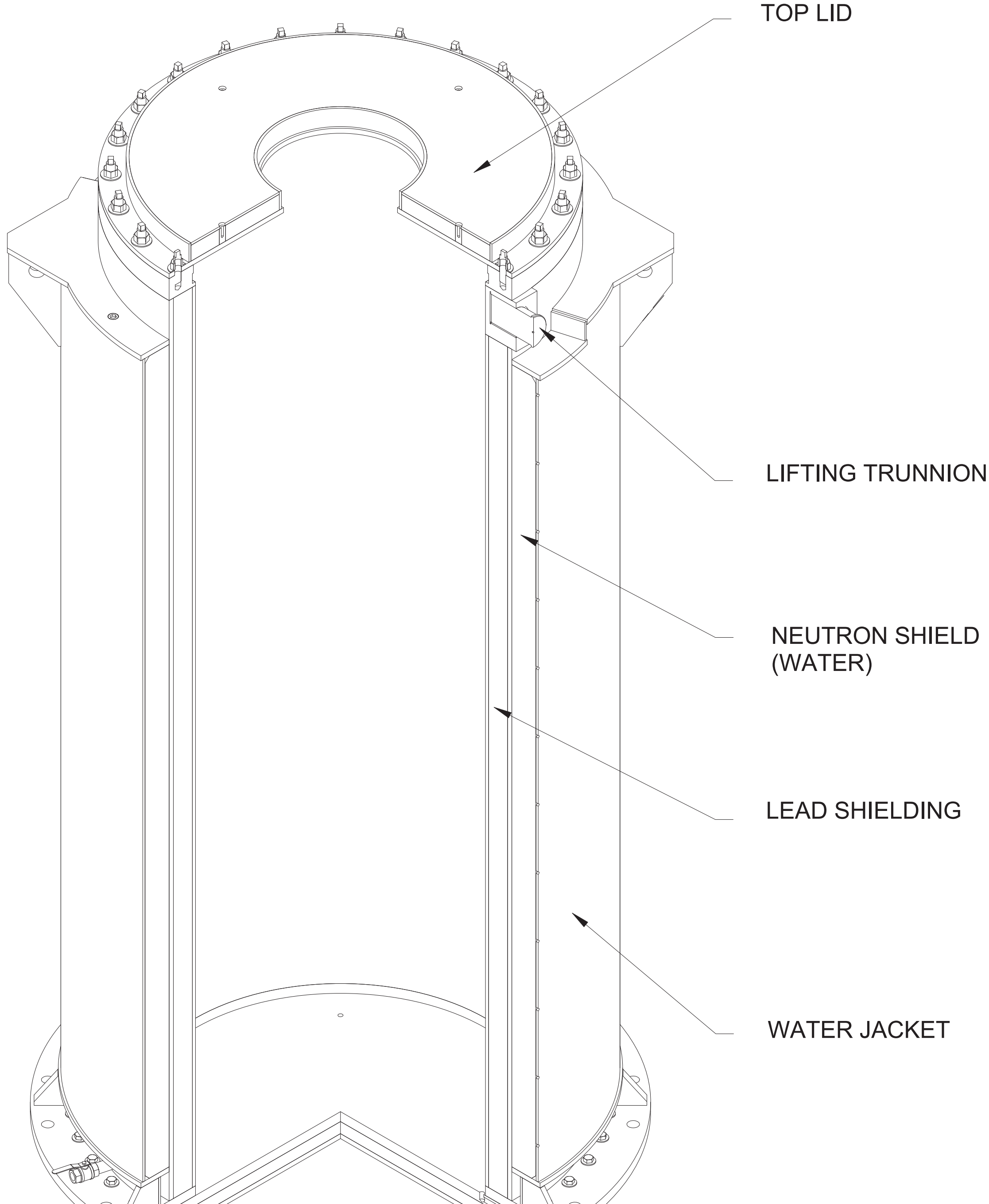

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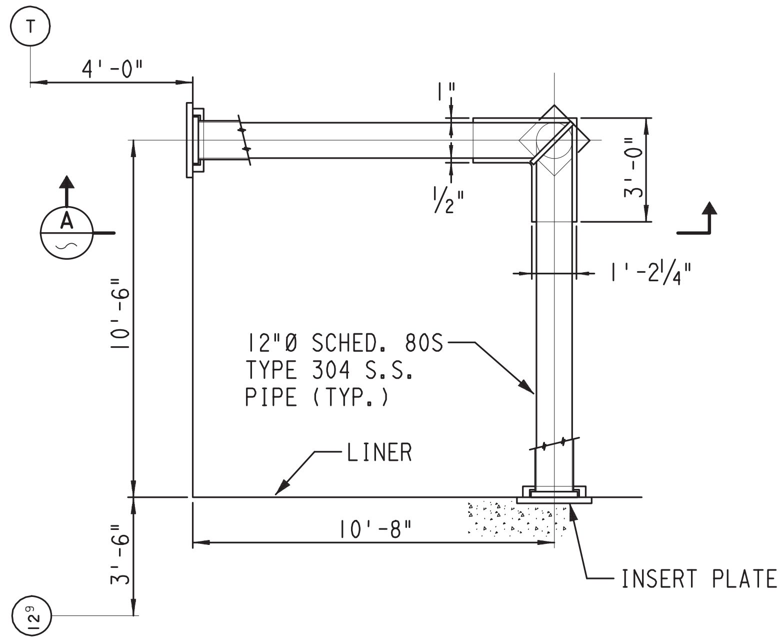
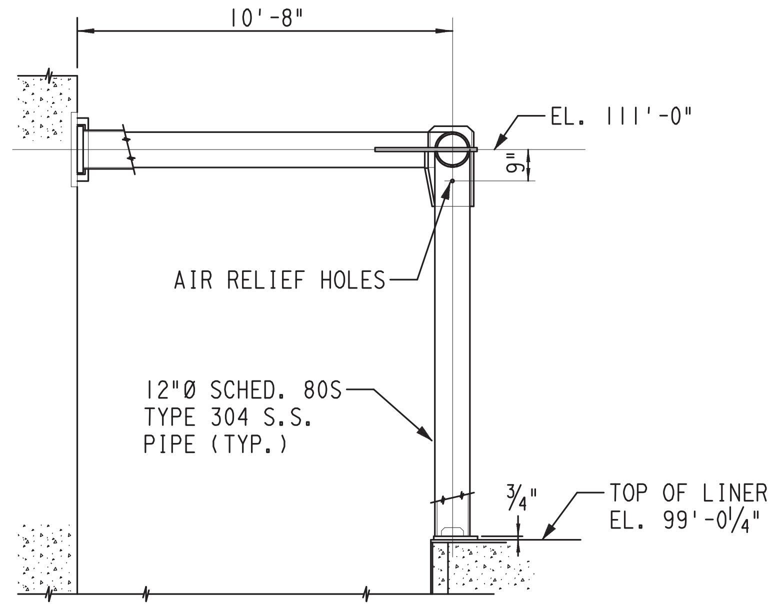
FIGURE 9.1-4 HI-TRAC 125D TRANSFER CASK



PLAN - SPENT FUEL CASK RESTRAINT



SECTION A

NOTE:
THIS FIGURE IS FOR UNIT 1
COLUMN LINES FOR UNIT 2 ARE DIFFERENT

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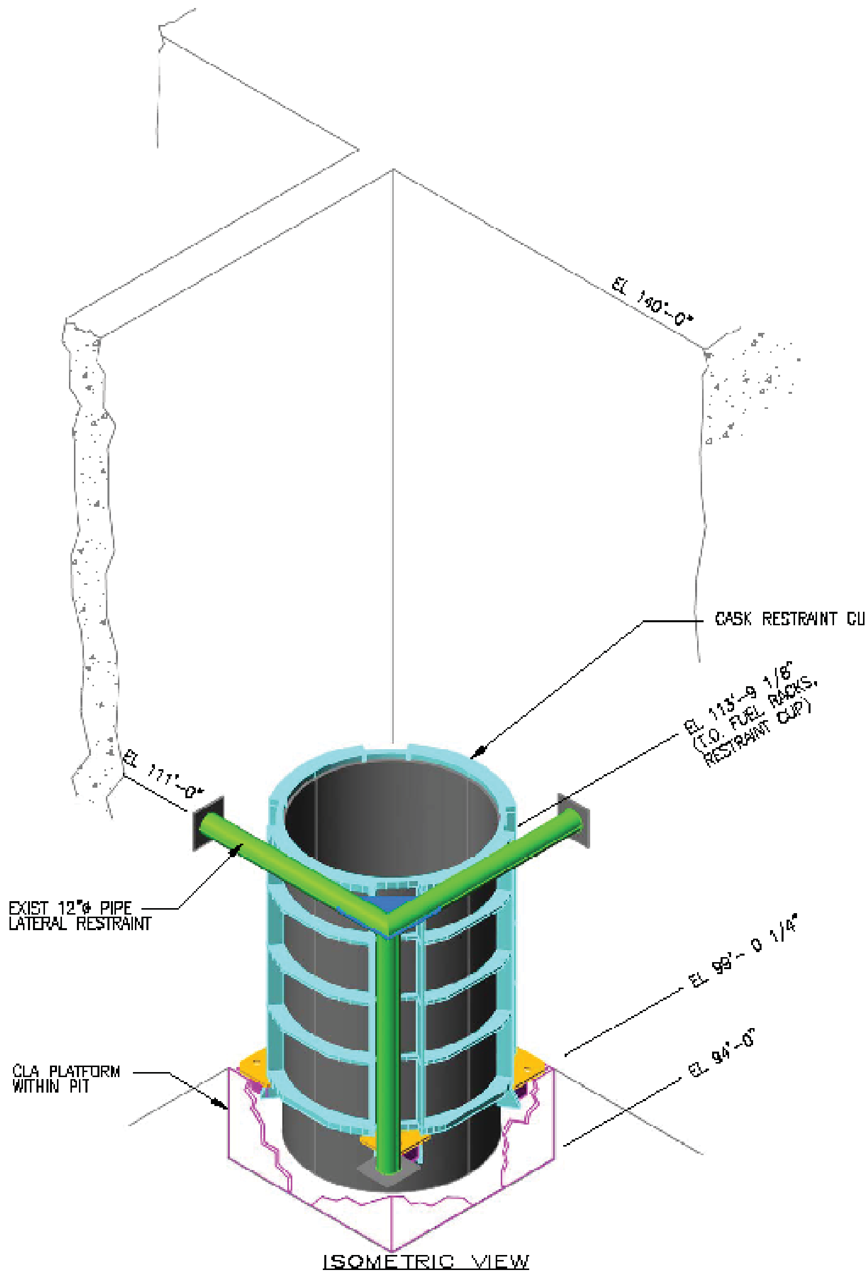
UNITS 1 AND 2 **DIABLO CANYON SITE**
FIGURE 9.1-5 **SPENT FUEL CASK RESTRAINT** **SPENT FUEL POOL**



Dimensions and elevations are estimates

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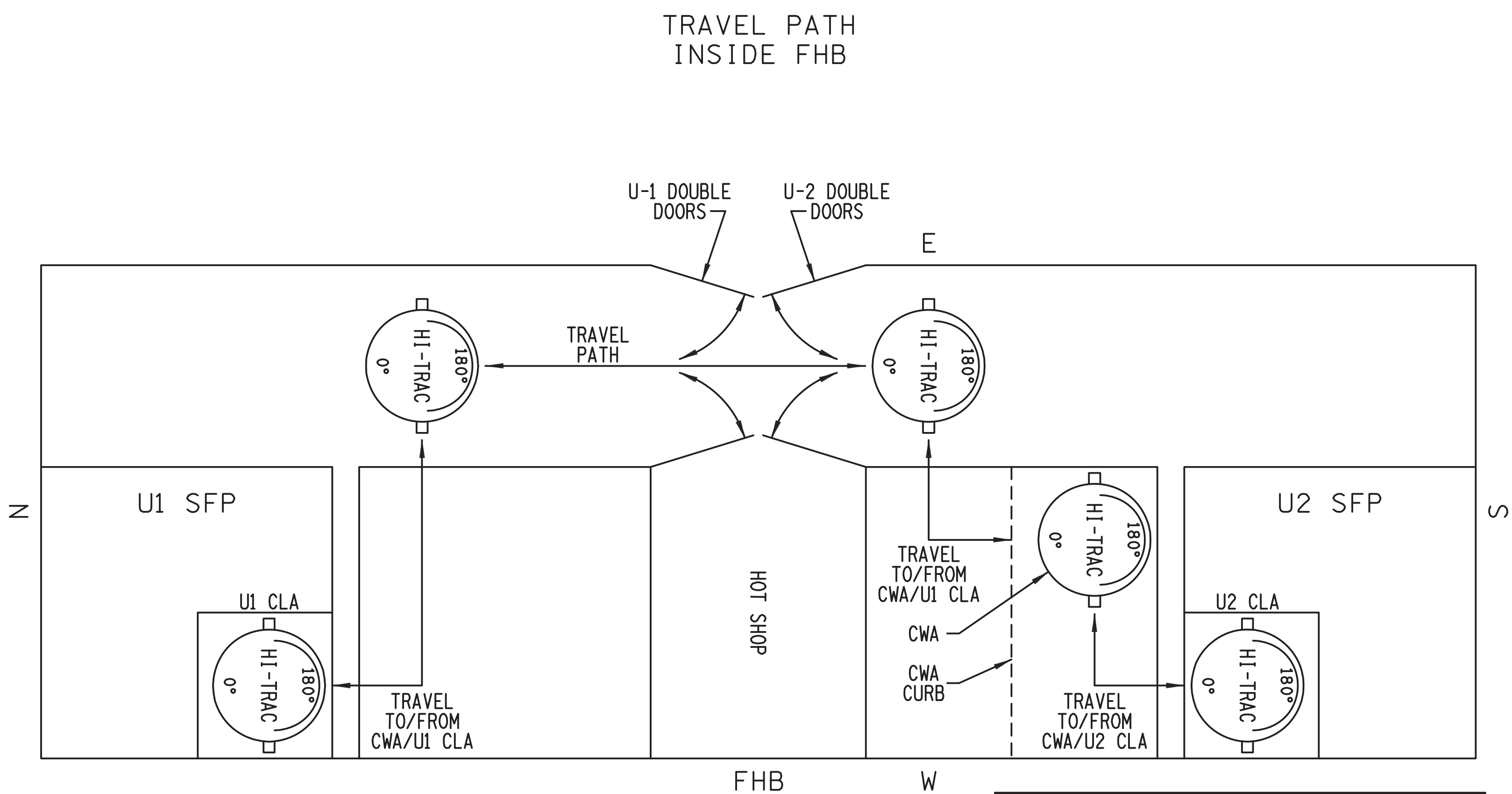
FIGURE 9.1-6 SPENT FUEL POOL TRANSFER CASK RESTRAINT CUP

Revision 19 May 2010



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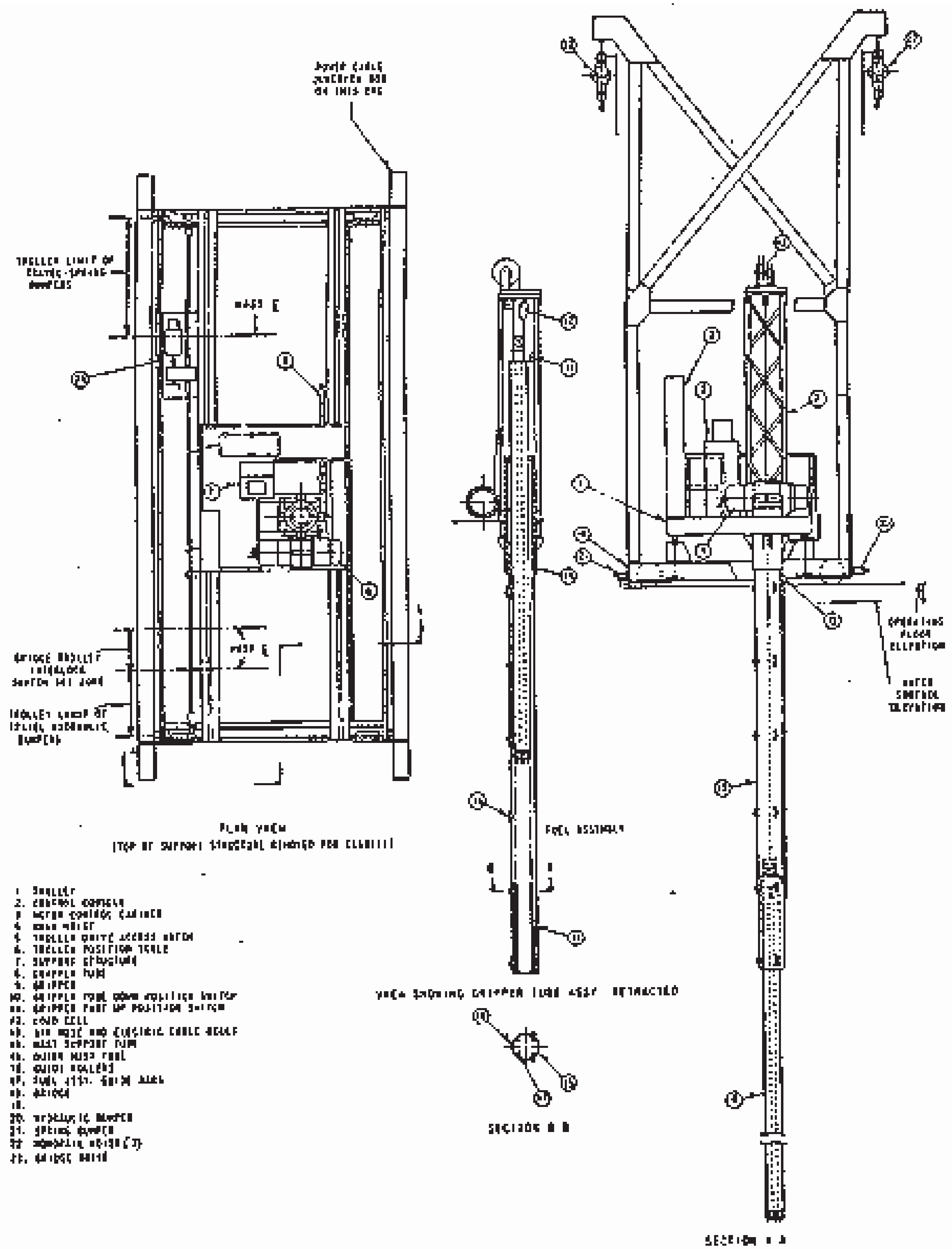
DIABLO CANYON SITE

FIGURE 9.1-7
HEAVY LOAD HANDLING PATHS FOR THE TRANSFER CASK/MPC

Revision 19 May 2010



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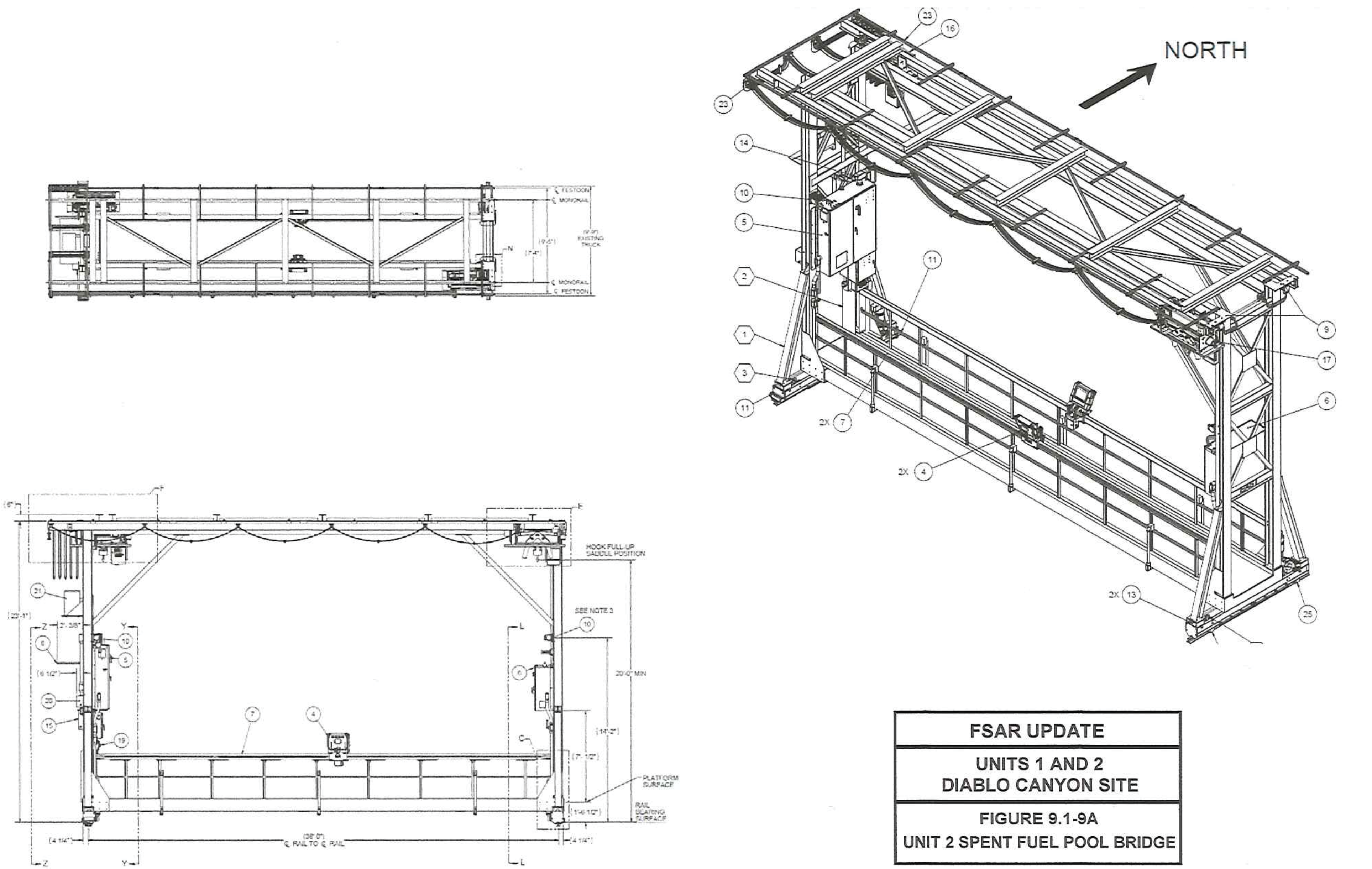
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.1-8

MANIPULATOR CRANE



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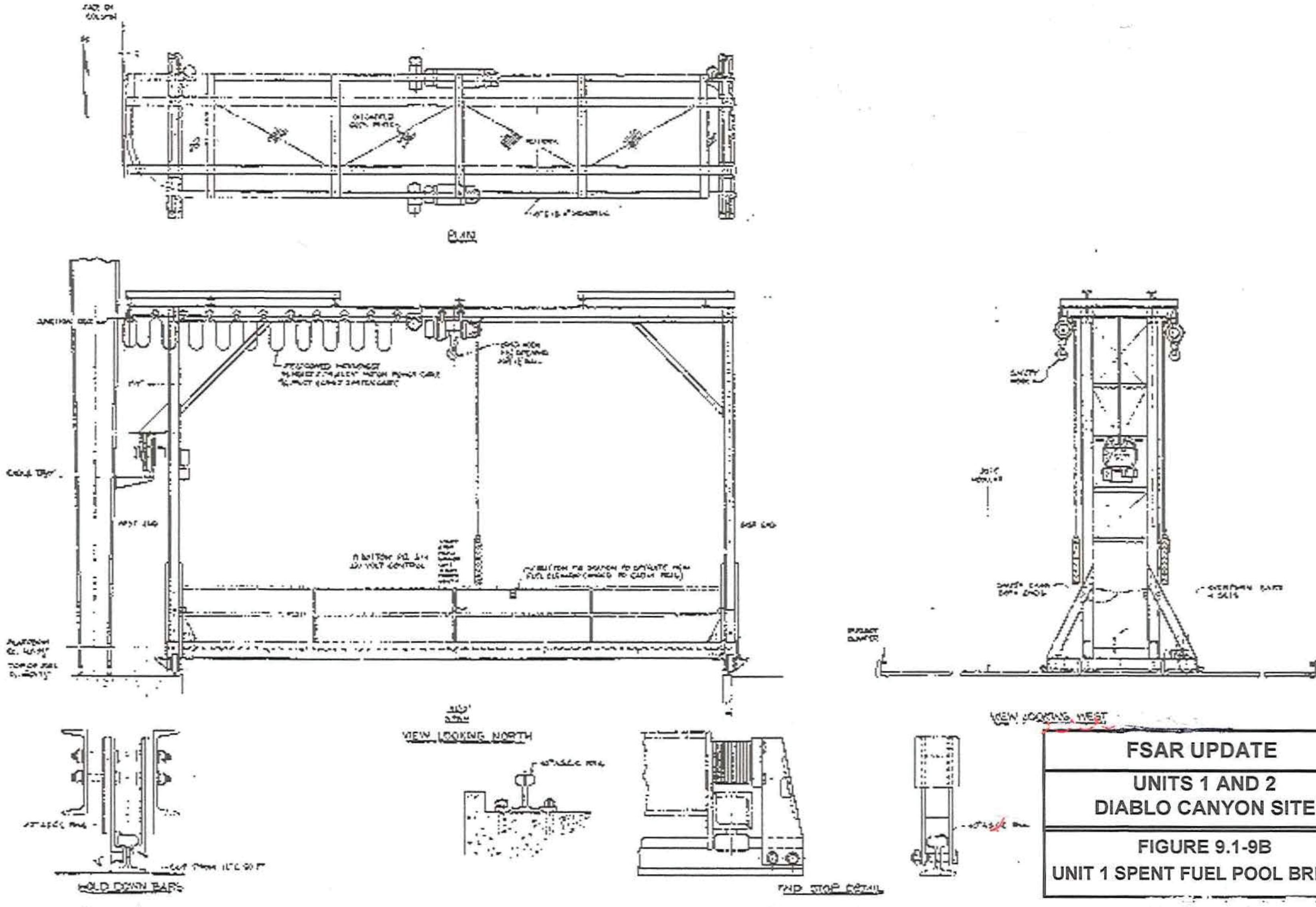
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 9.1-9A UNIT 2 SPENT FUEL POOL BRIDGE



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UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.1-9B
UNIT 1 SPENT FUEL POOL BRIDGE

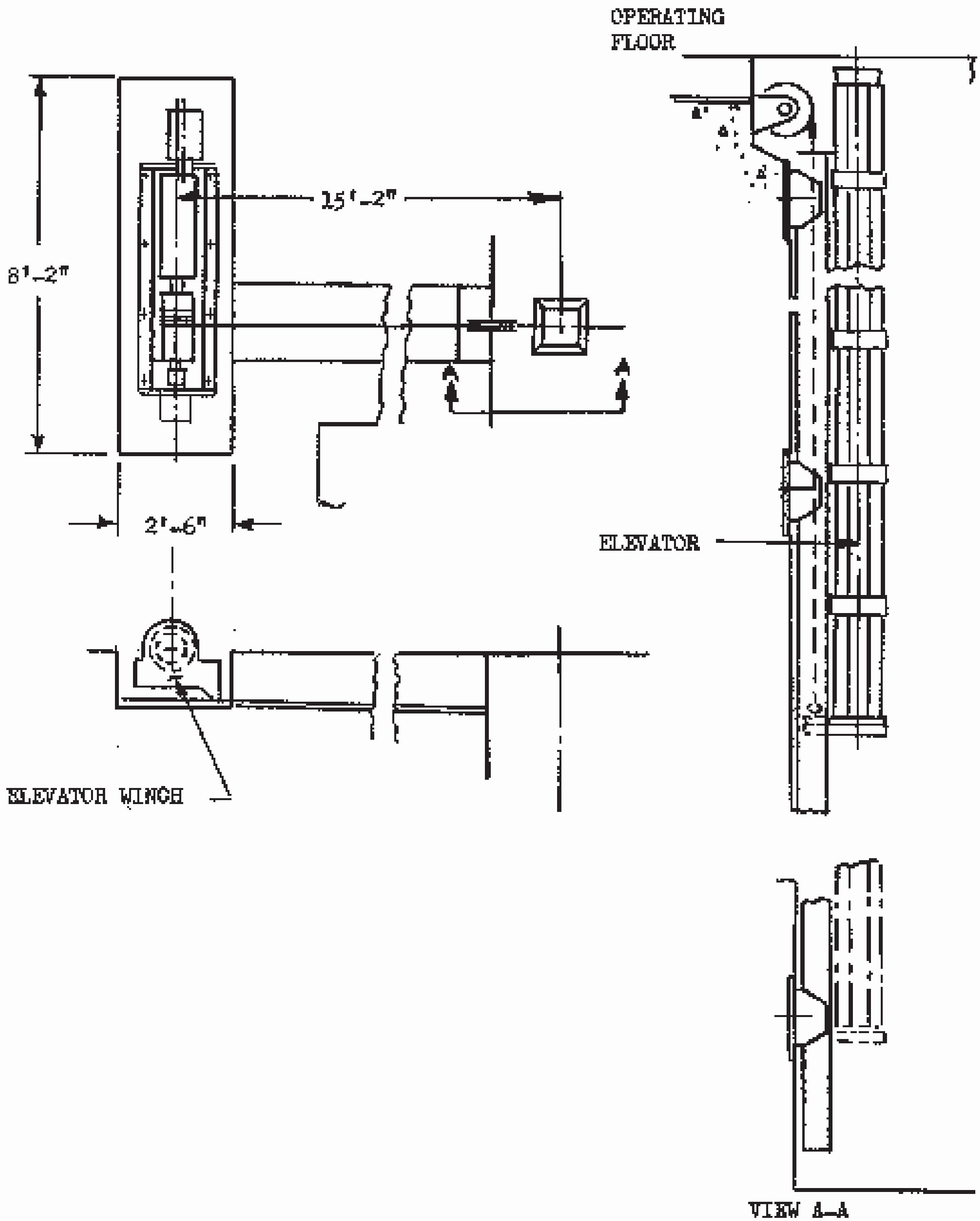

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FIGURE 9.1-10 NEW FUEL ELEVATOR

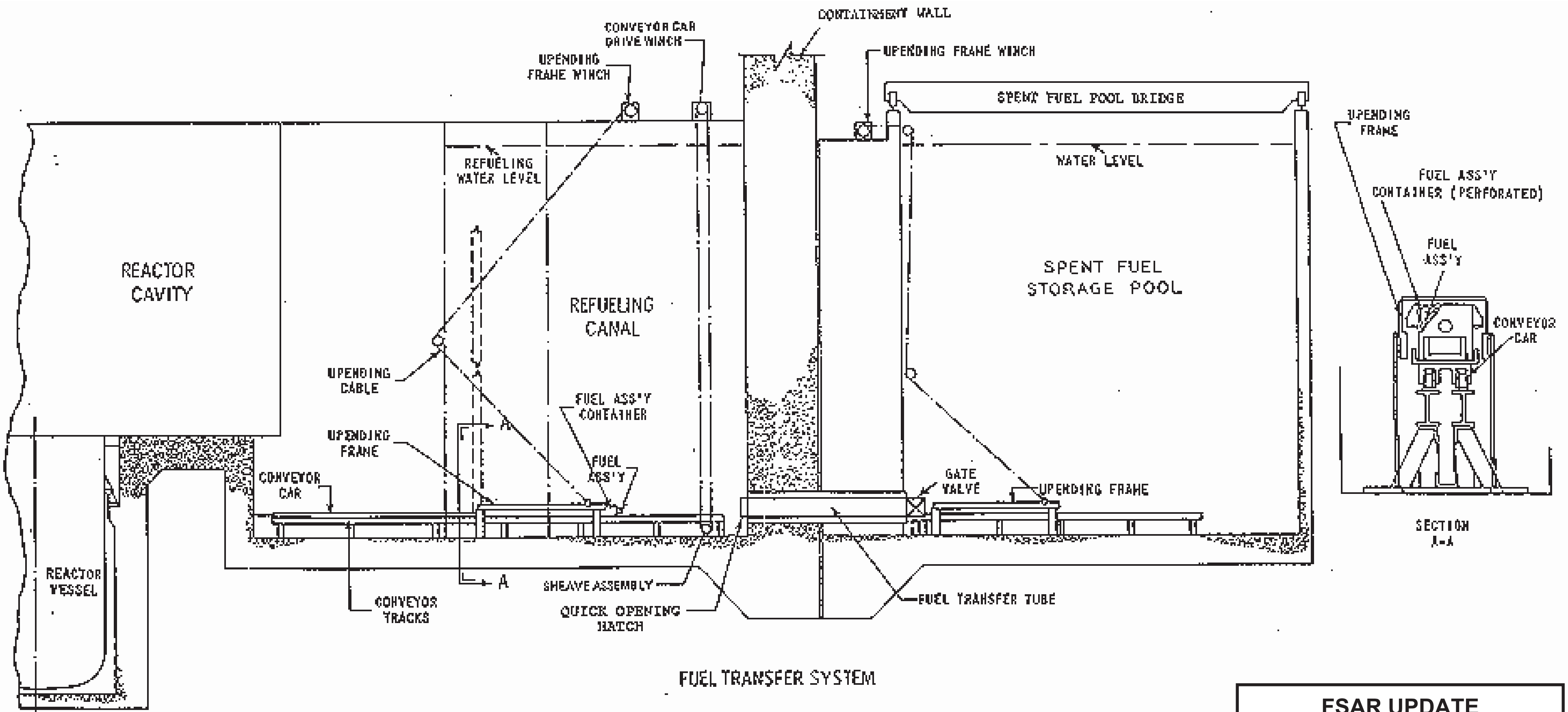

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FIGURE 9.1-11 FUEL TRANSFER SYSTEM

Revision 11 November 1996

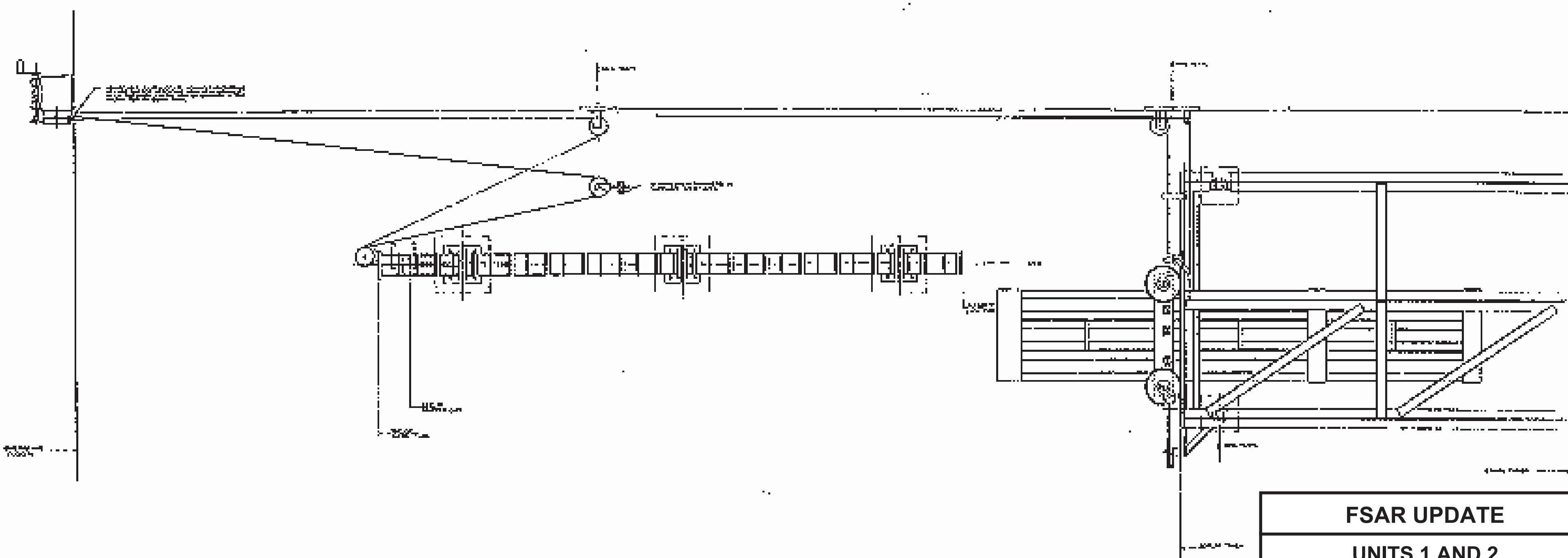

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FIGURE 9.1-12 ROD CLUSTER CONTROL CHANGING FIXTURE

Revision 11 November 1996

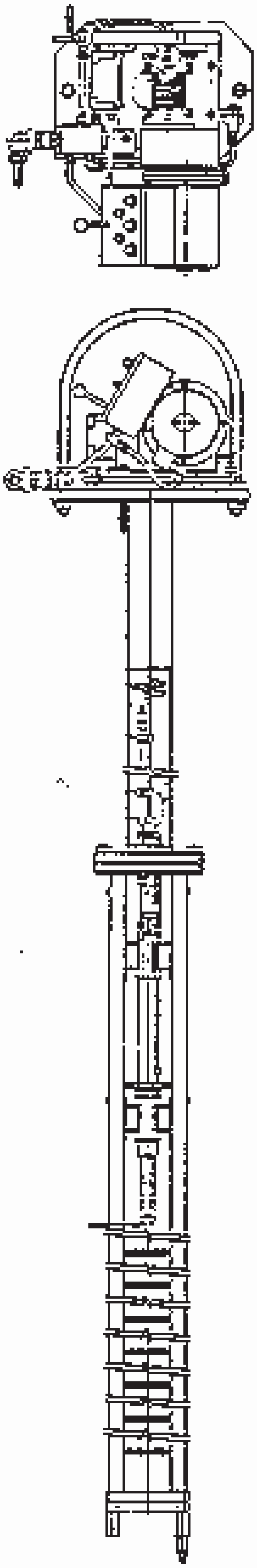

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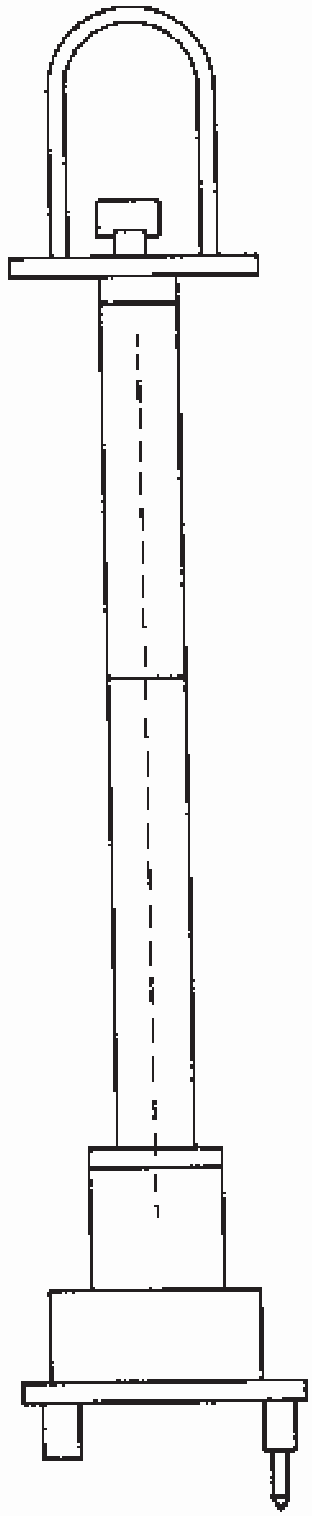
FIGURE 9.1-12a
ROD CLUSTER CONTROL
CHANGING TOOL



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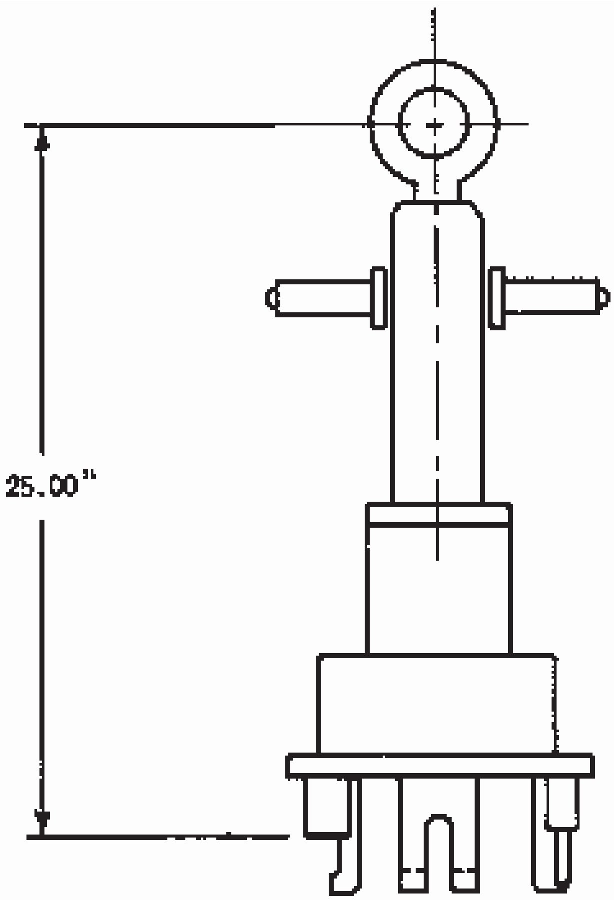
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.1-13
SPENT FUEL HANDLING TOOL



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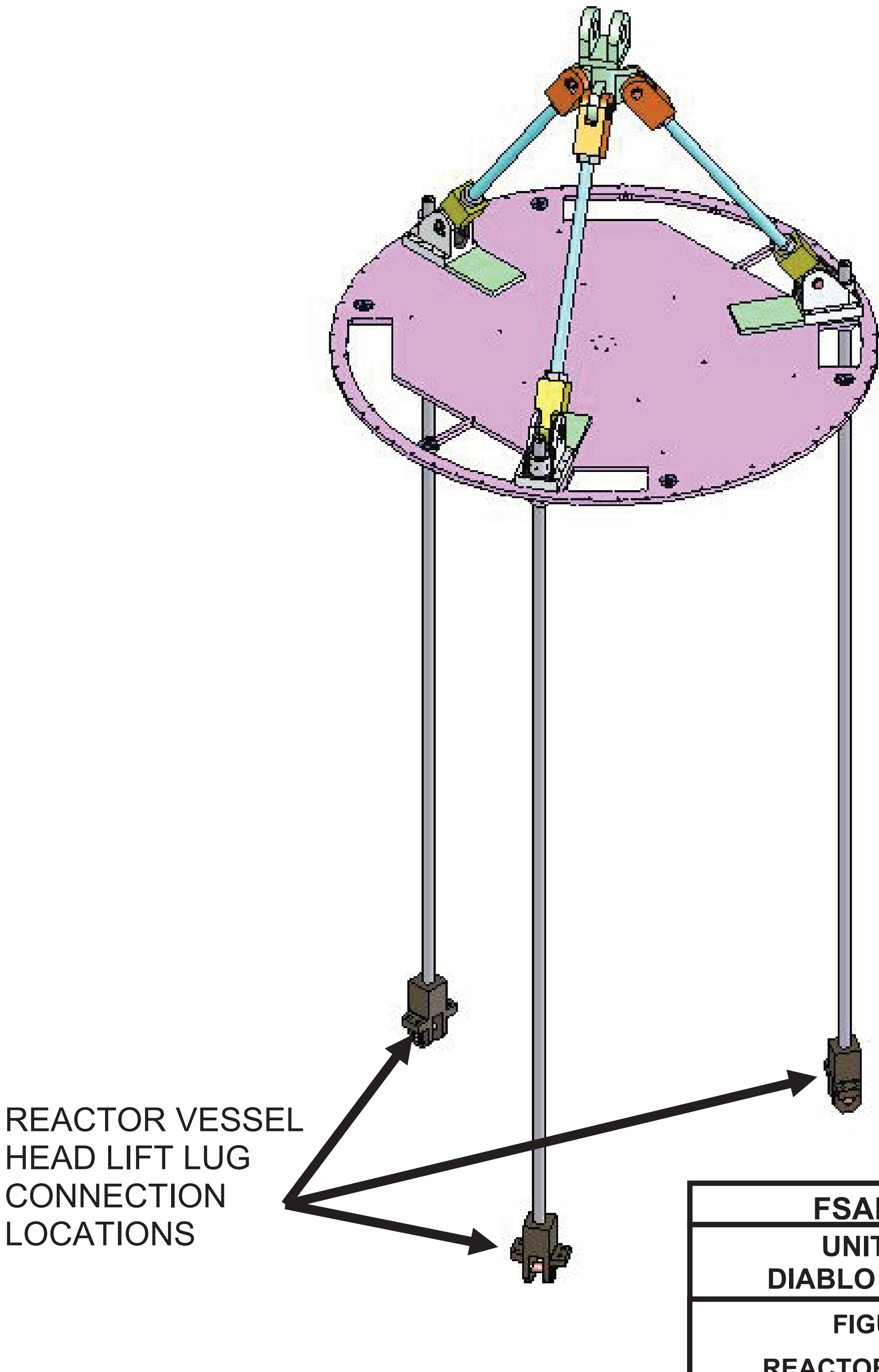
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.1-14
NEW FUEL ASSEMBLY
HANDLING FIXTURE



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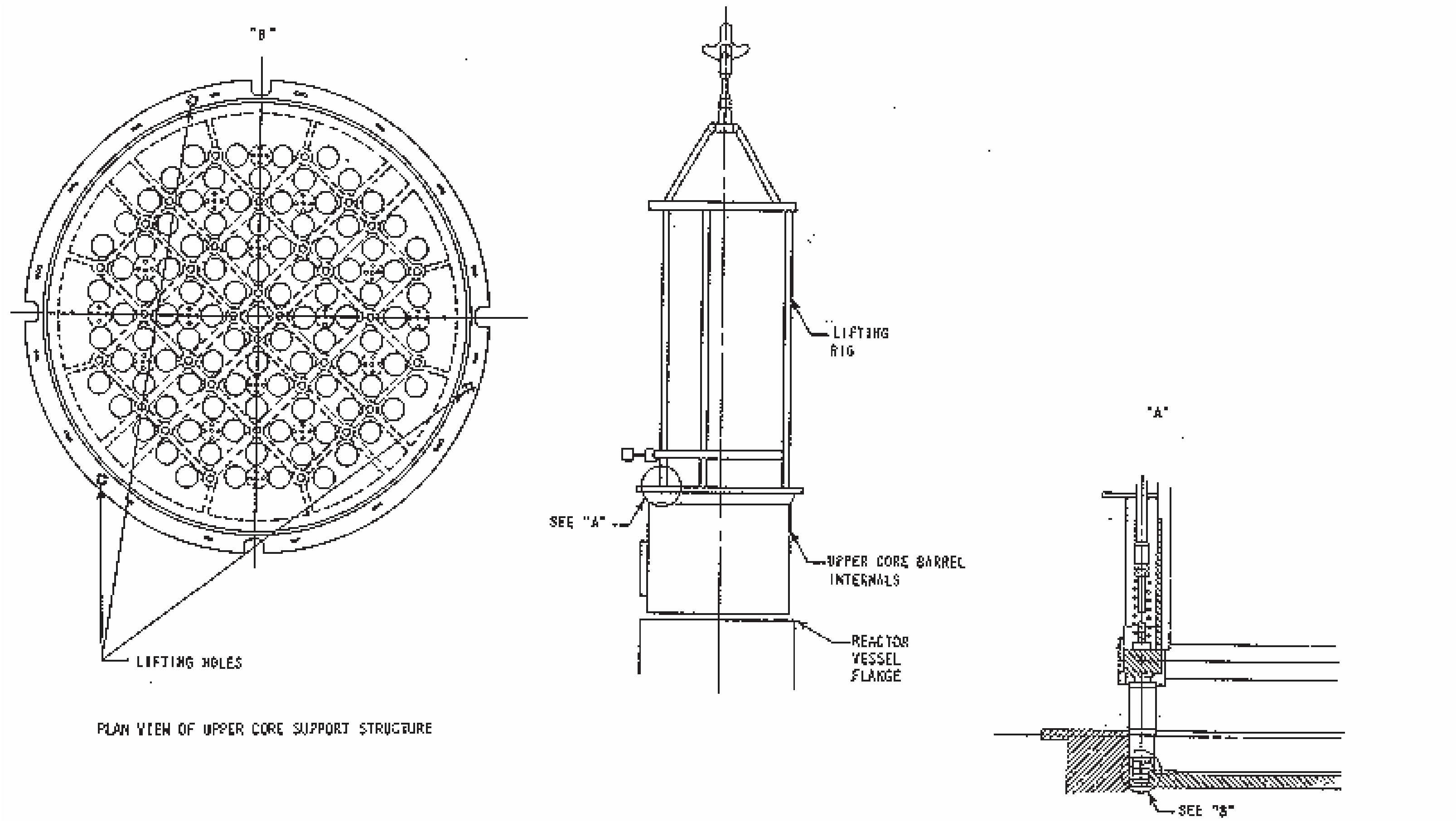
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.1-15

REACTOR VESSEL HEAD
LIFT RIG ASSEMBLY



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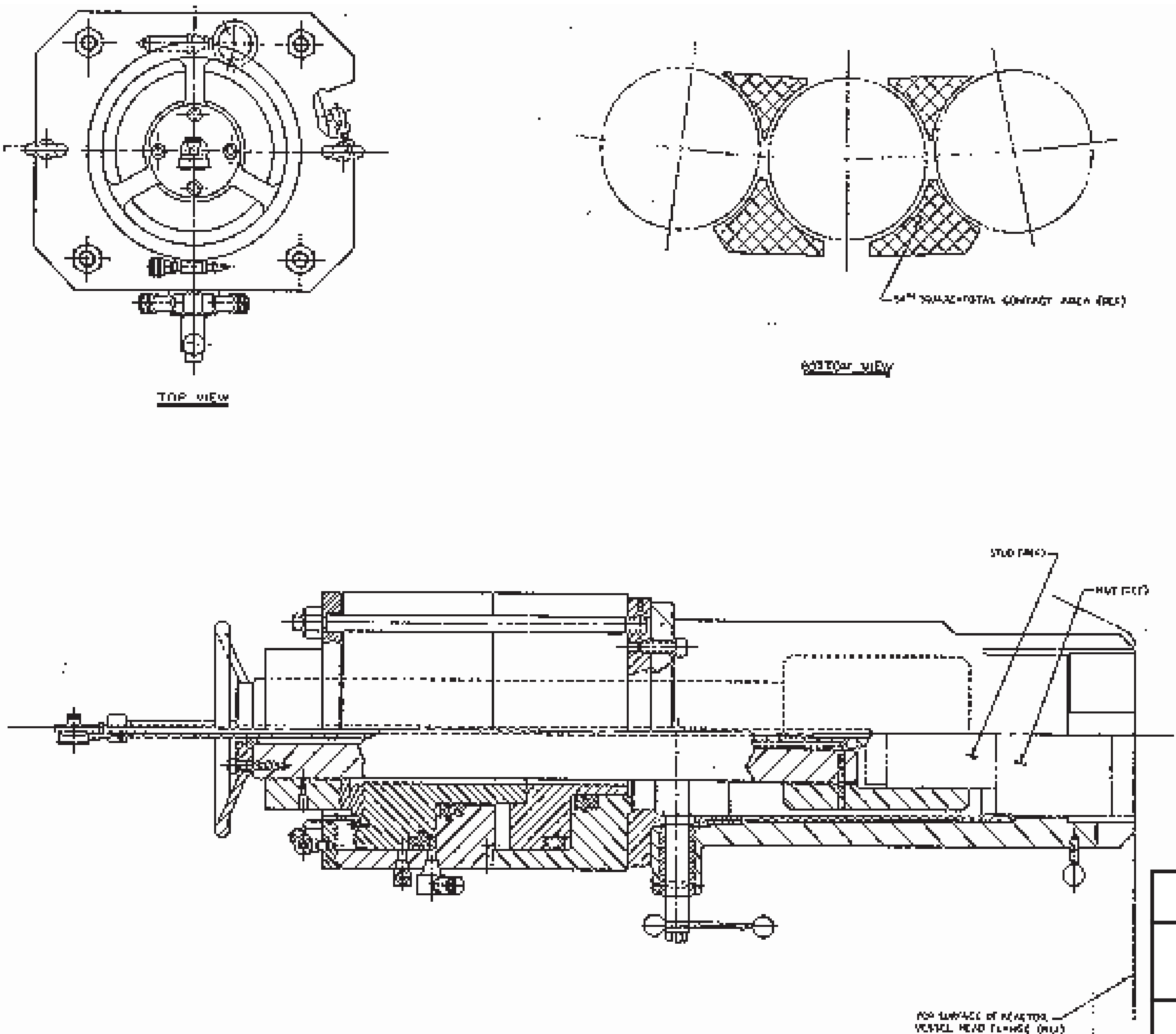
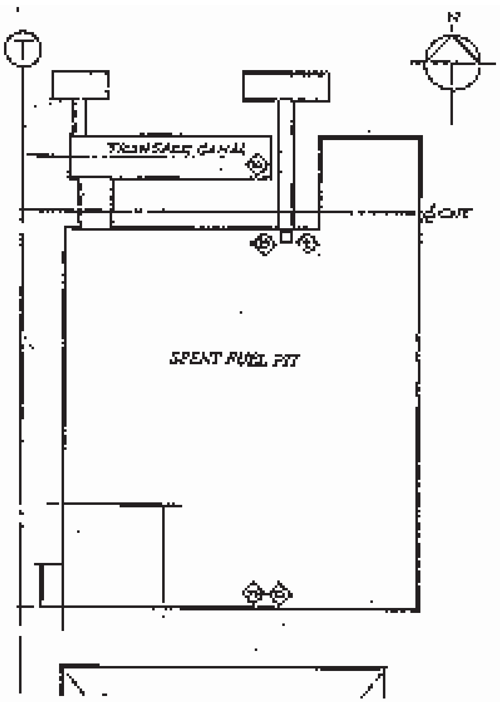
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.1-16
REACTOR INTERNALS LIFTING DEVICE



FSAR UPDATE
UNITS 1 AND 2 **DIABLO CANYON SITE**
FIGURE 9.1-17 **REACTOR VESSEL** **STUD TENSIONER**

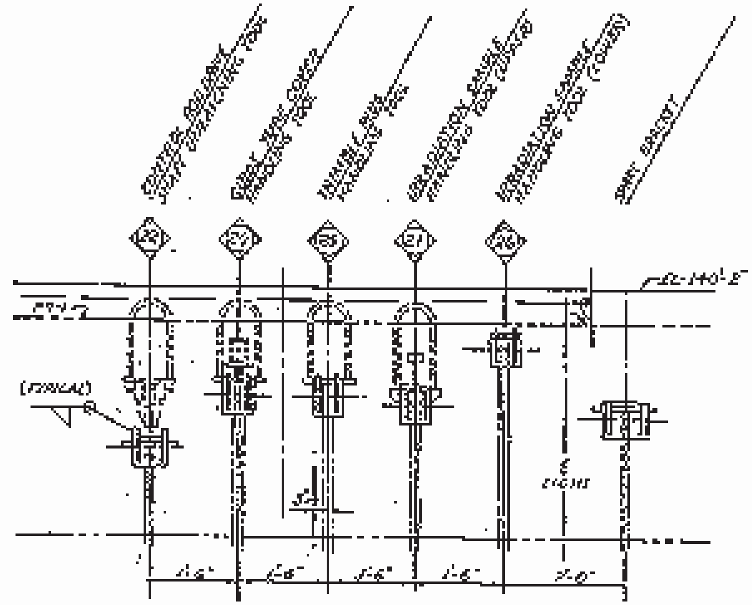
Revision 11 November 1996



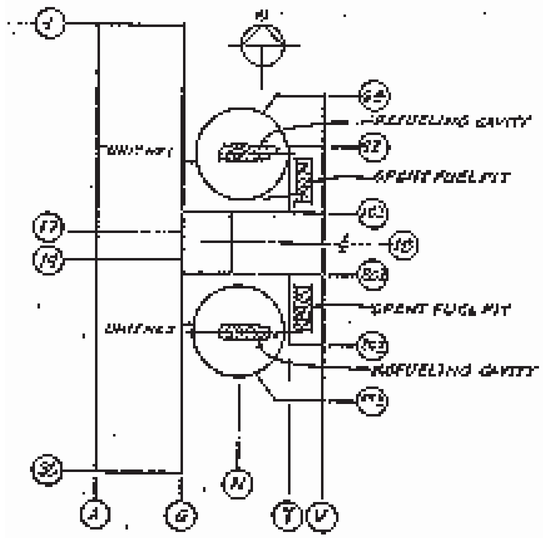
TOOL LOCATION PLAN
FUEL HANDLING BLDG.
UNIT No. 1 SHOWN / UNIT No. 2 OPPOSITE HAND

No.	NAME	LENGTH	WEIGHT
[illegible]	IRRADIATION SAMPLE/UPPER	21'-10" ±	110 LBS
[illegible]	CONT. ROD D.S. UNLATCHING	17'-3" ±	250 LBS
	TOOL DELETED		
[illegible]	GUIDE TUBE COVER	68'-6" ±	185 LBS
[illegible]	THIMBLE PLUG	30'-9" ±	250 LBS
[illegible]	IRRADIATION SAMPLE/LOWER	21'-8" ±	120 LBS
[illegible]	CASK STORAGE RACK LATCHING	16'-6"	10 LBS
[illegible]	17x17 SPENT FUEL ASSEMBLY	30'-4"	386 LBS
[illegible]	BURNABLE POISON ROD	35'-2"	602 LBS.
[illegible]	CASK STOR.-RACK TOP		135 LBS
[illegible]	CASK STOR.-RACK BOTTOM		
[illegible]	GUIDE PIN SUPPORT TOP	19'-6" ±	1500 LBS
[illegible]	GUIDE PIN SUPPORT BOTTOM		
[illegible]	NEW FUEL ASSEMBLY	2'-0" ±	25 LBS
[illegible]	NEW ROD CHANGING FIXTURE	33'-1" ±	1045 LBS
[illegible]	STUD HOLE PLUG	4'-0" ±	
[illegible]	THIMBLE PLUG	35'-0" ±	356 LBS.

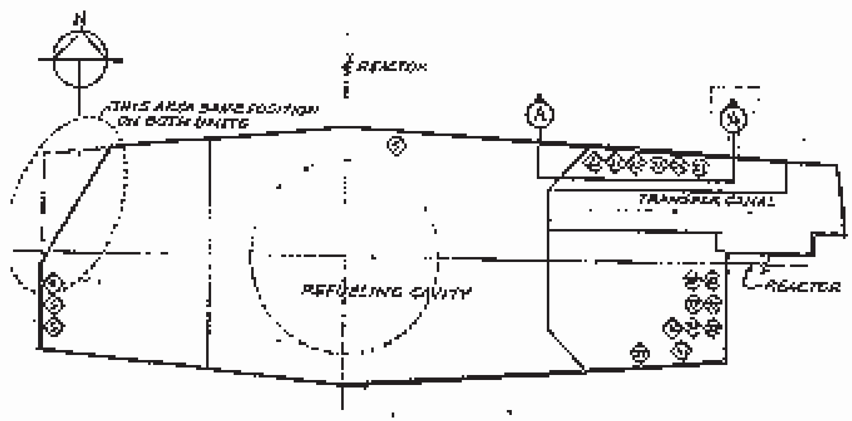
HANDLING TOOL RACK SCHEDULE



HANDLING TOOL - SUPPORT RACKS
SECTION A - A



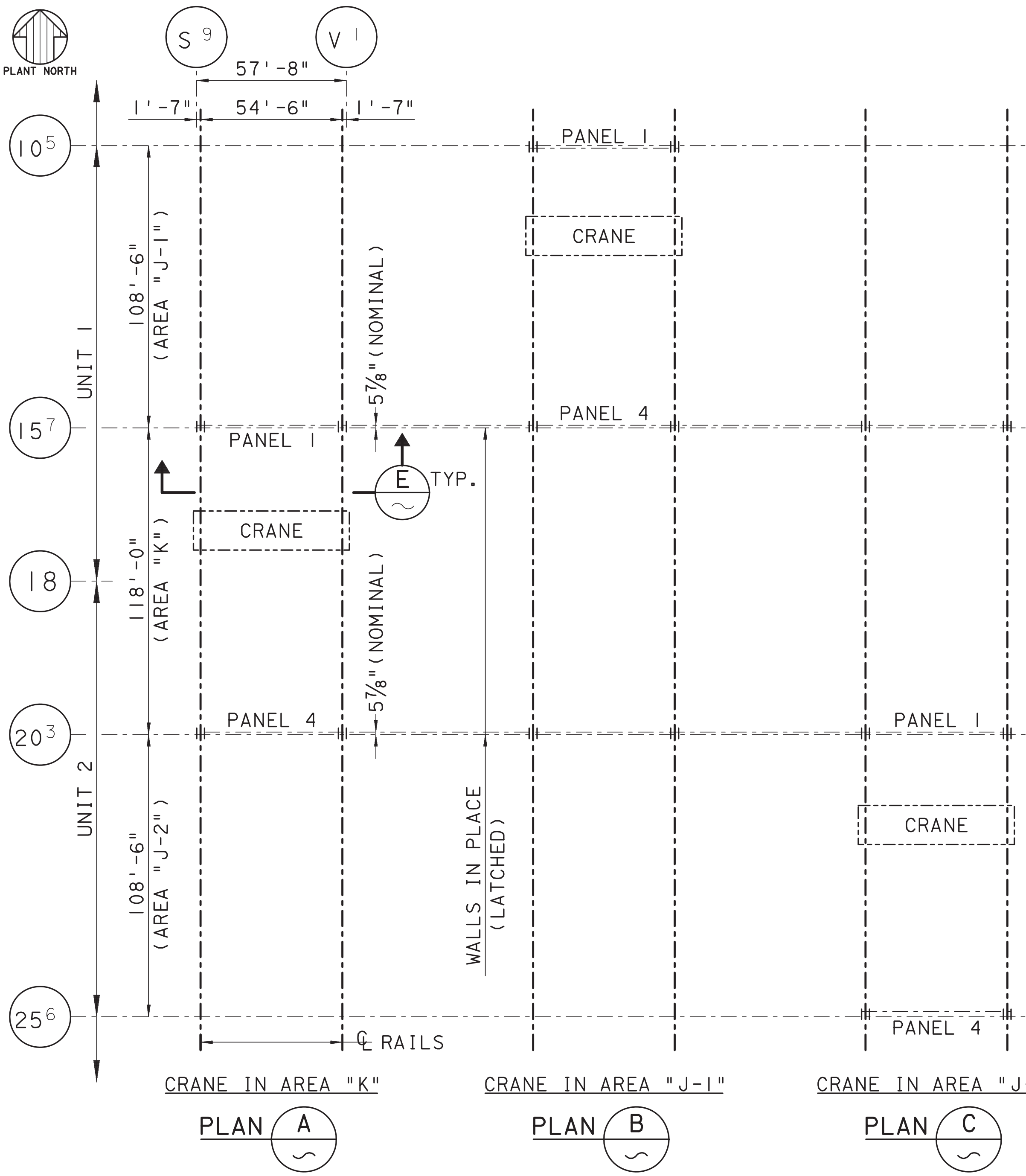
KEY PLAN



TOOL LOCATION PLAN - CONTAINMENT STRUCTURE
UNIT No. 1 SHOWN / UNIT No. 2 OPPOSITE HAND EXCEPT AS NOTED

FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 9.1-18 FUEL HANDLING TOOL LOCATIONS

Revision 11 November 1996

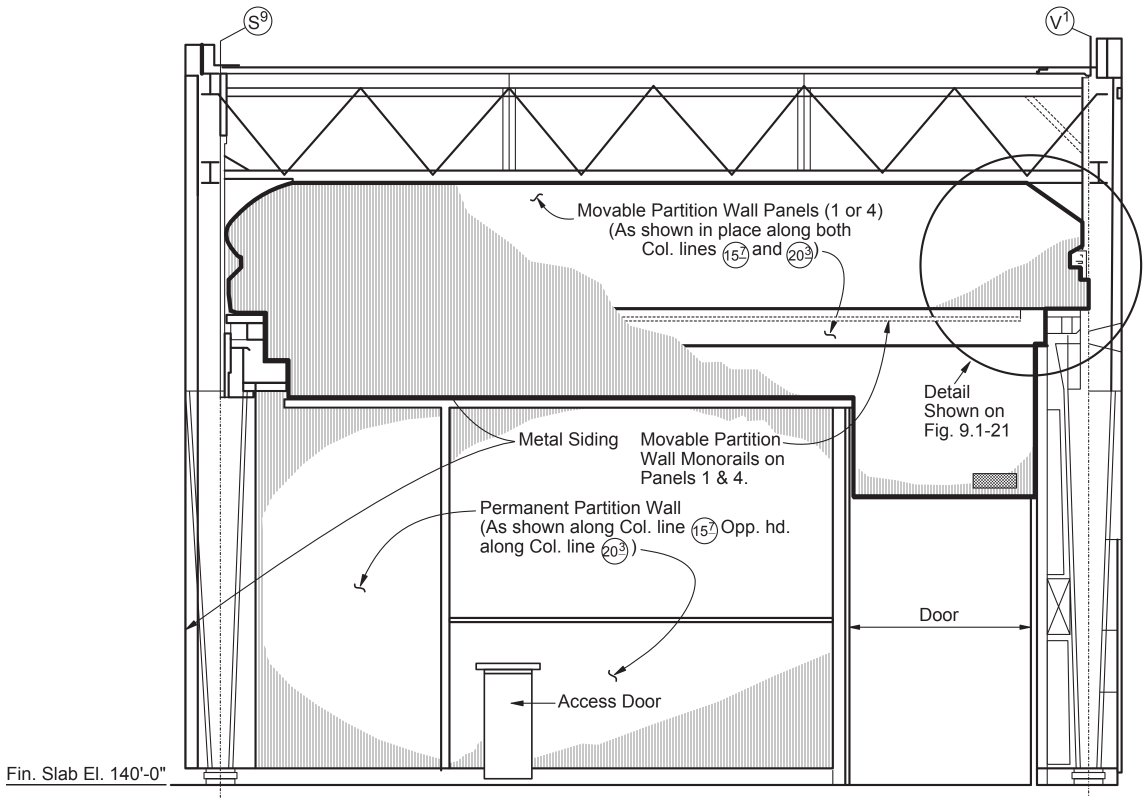


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.1-19
MOVEABLE PARTITION WALLS
LOCATION PLANS

Revision 19 May 2010

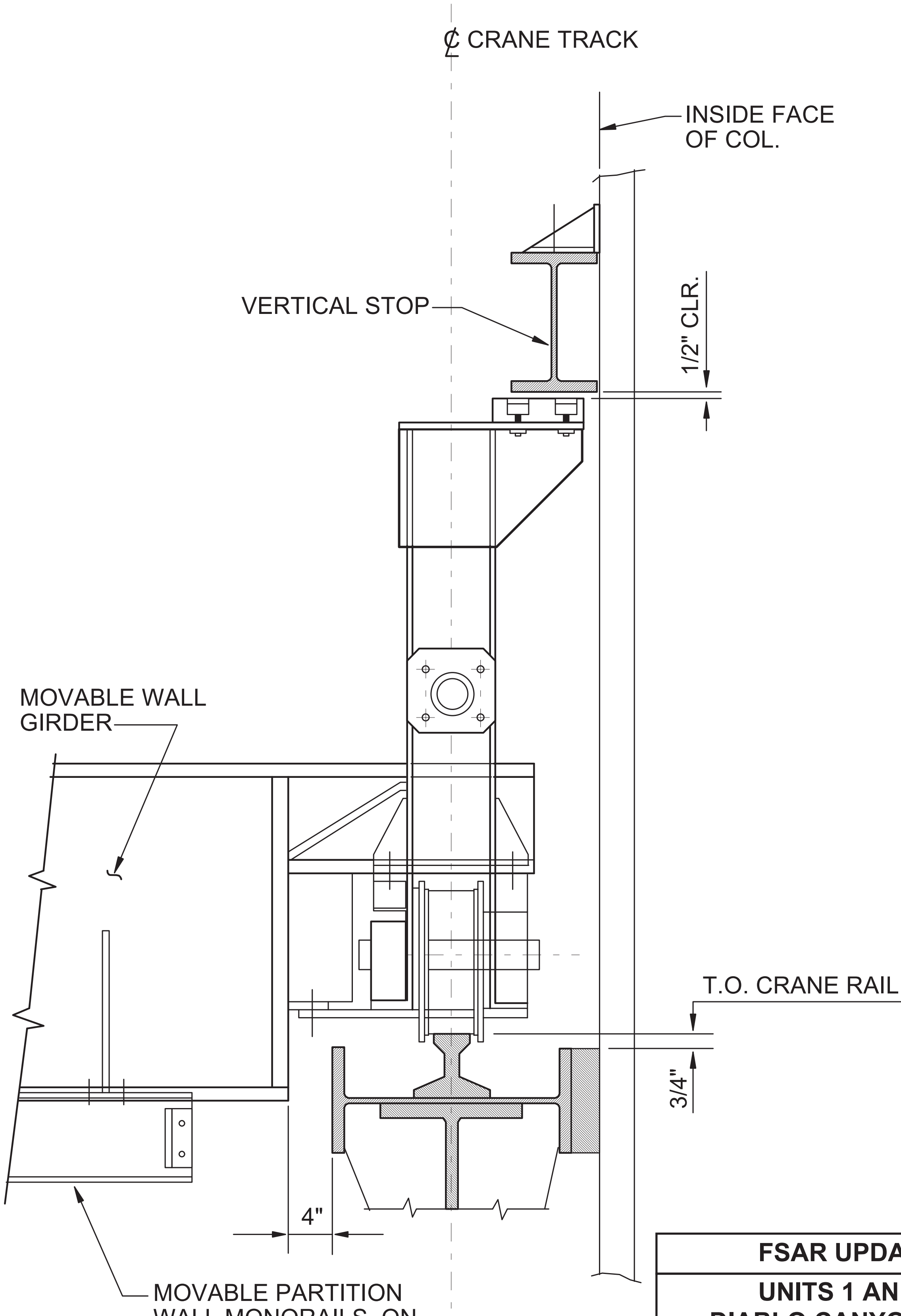


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.1-20
MOVEABLE PARTITION WALLS
ELEVATION AT COLUMN
LINE 15^{7} OR 20^{3}

Revision 19 May 2010

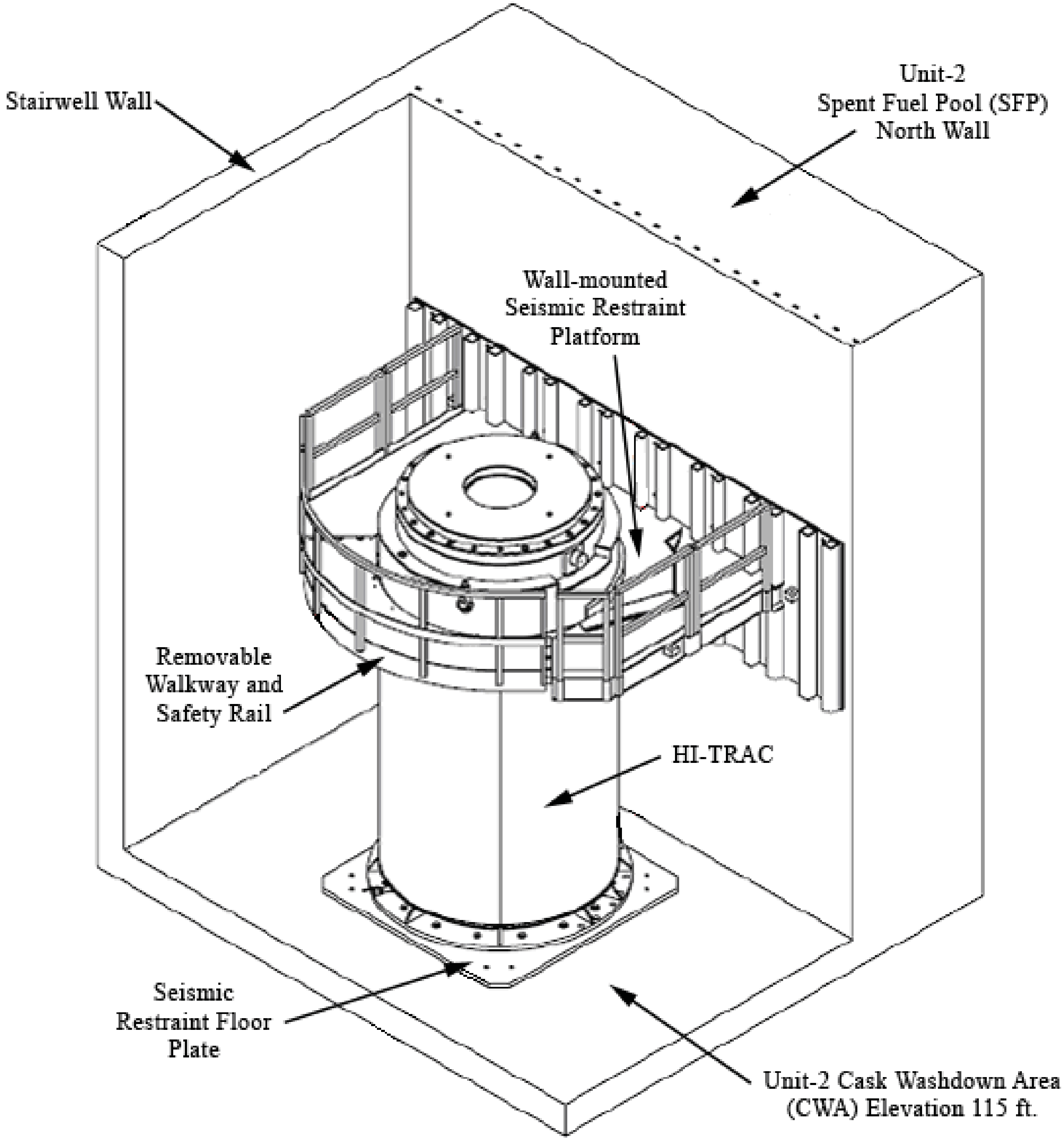


FSAR UPDATE

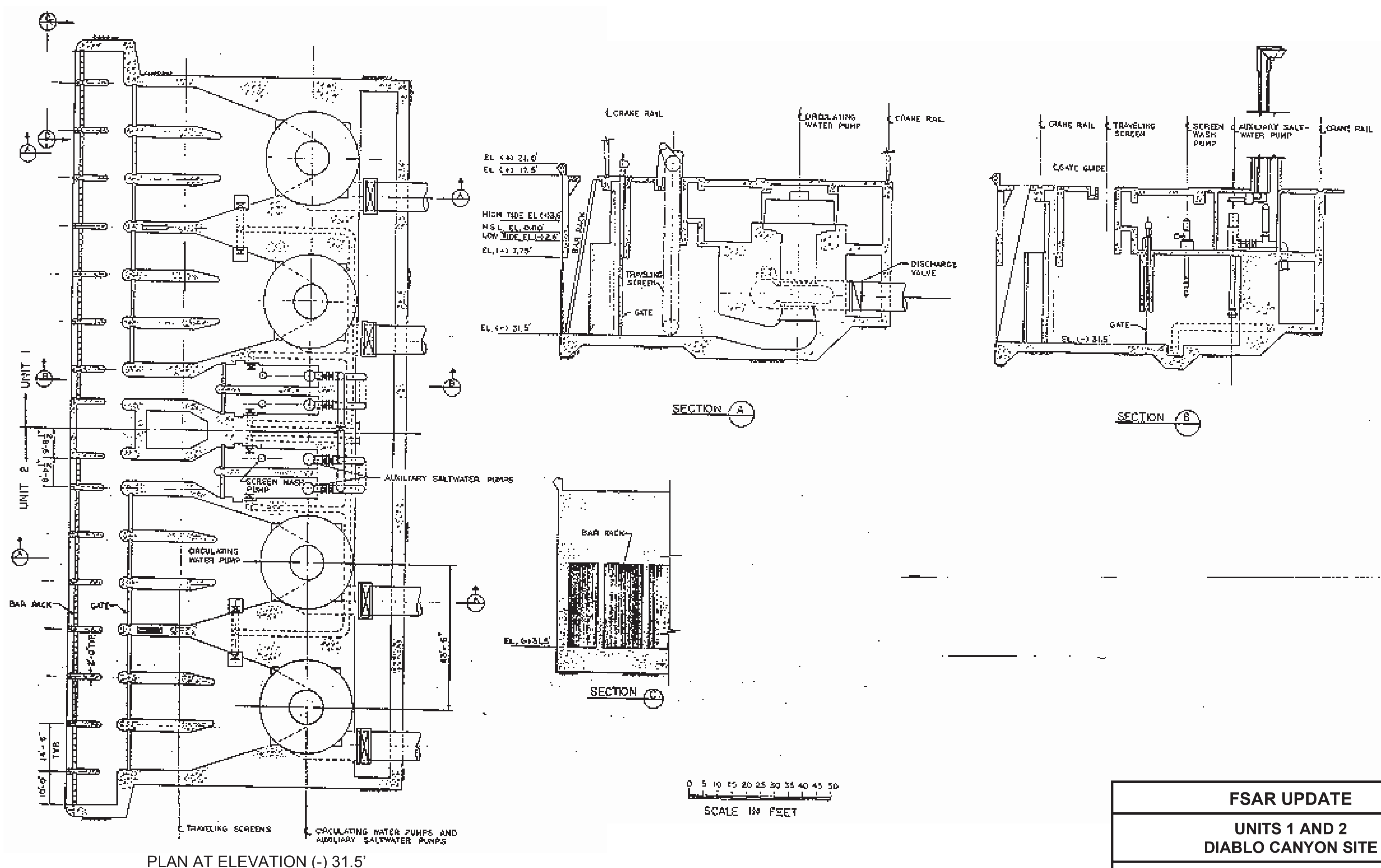
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.1-21
MOVEABLE PARTITION WALLS
DETAILS SHOWING WALL,
TRACK, AND VERTICAL STOP

Revision 19 May 2010



FSAR UPDATE
UNIT 2 DIABLO CANYON SITE
FIGURE 9.1-24 CASK WASHDOWN AREA RESTRAINT

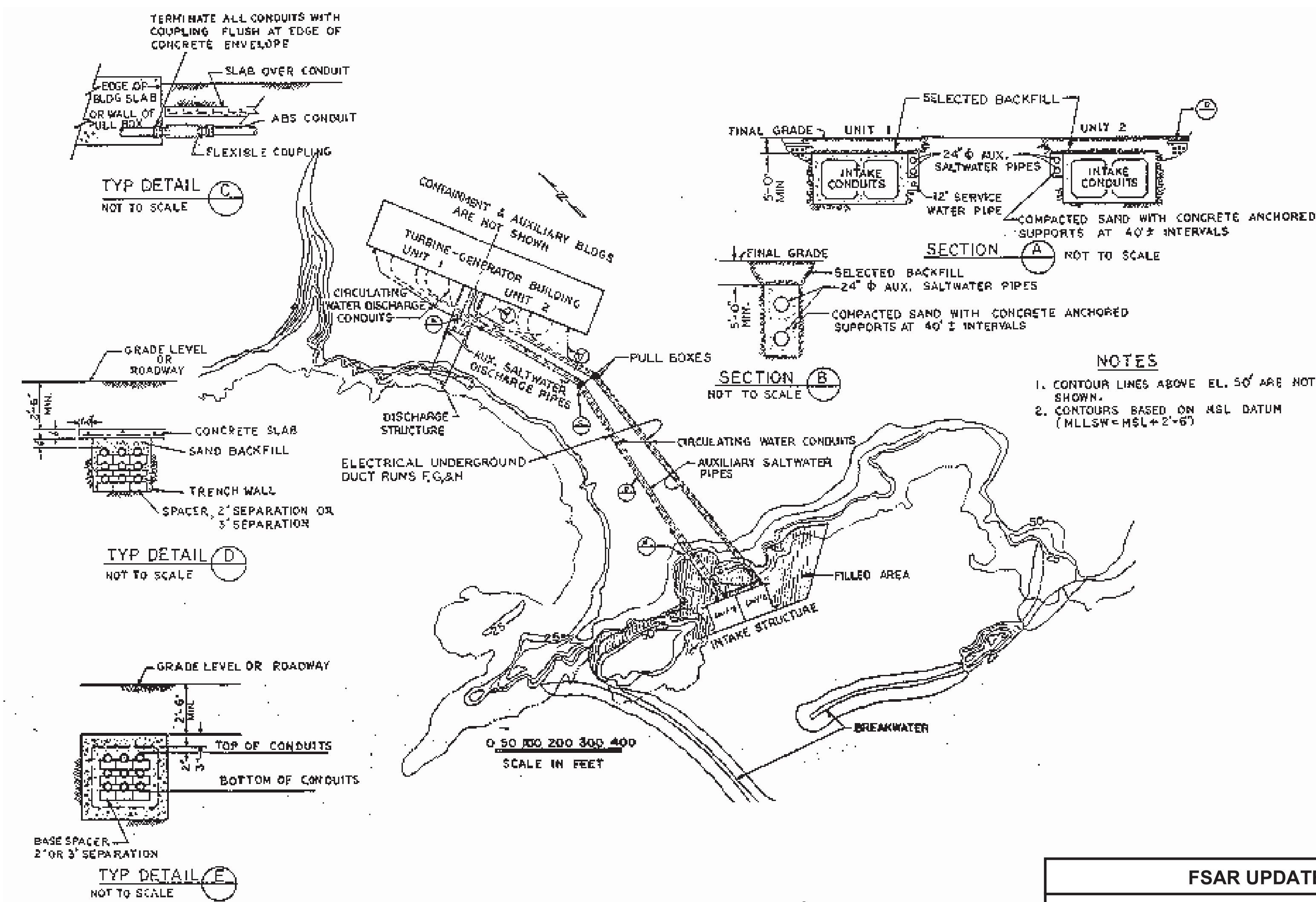


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.2-2
ARRANGEMENT OF INTAKE STRUCTURE

Revision 12 September 1998

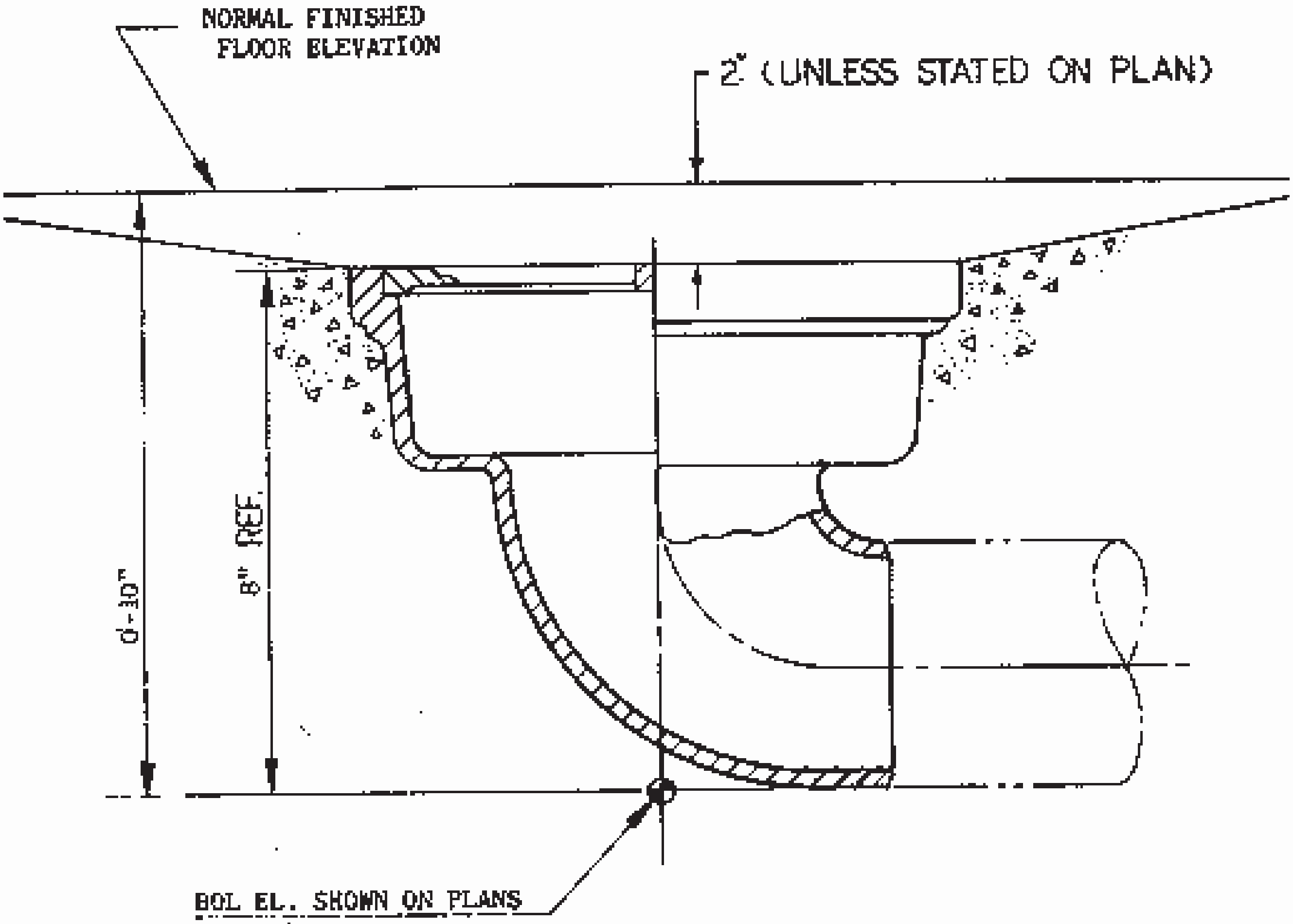


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.2-3
ARRANGEMENT OF AUXILIARY SALTWATER SYSTEM PIPING

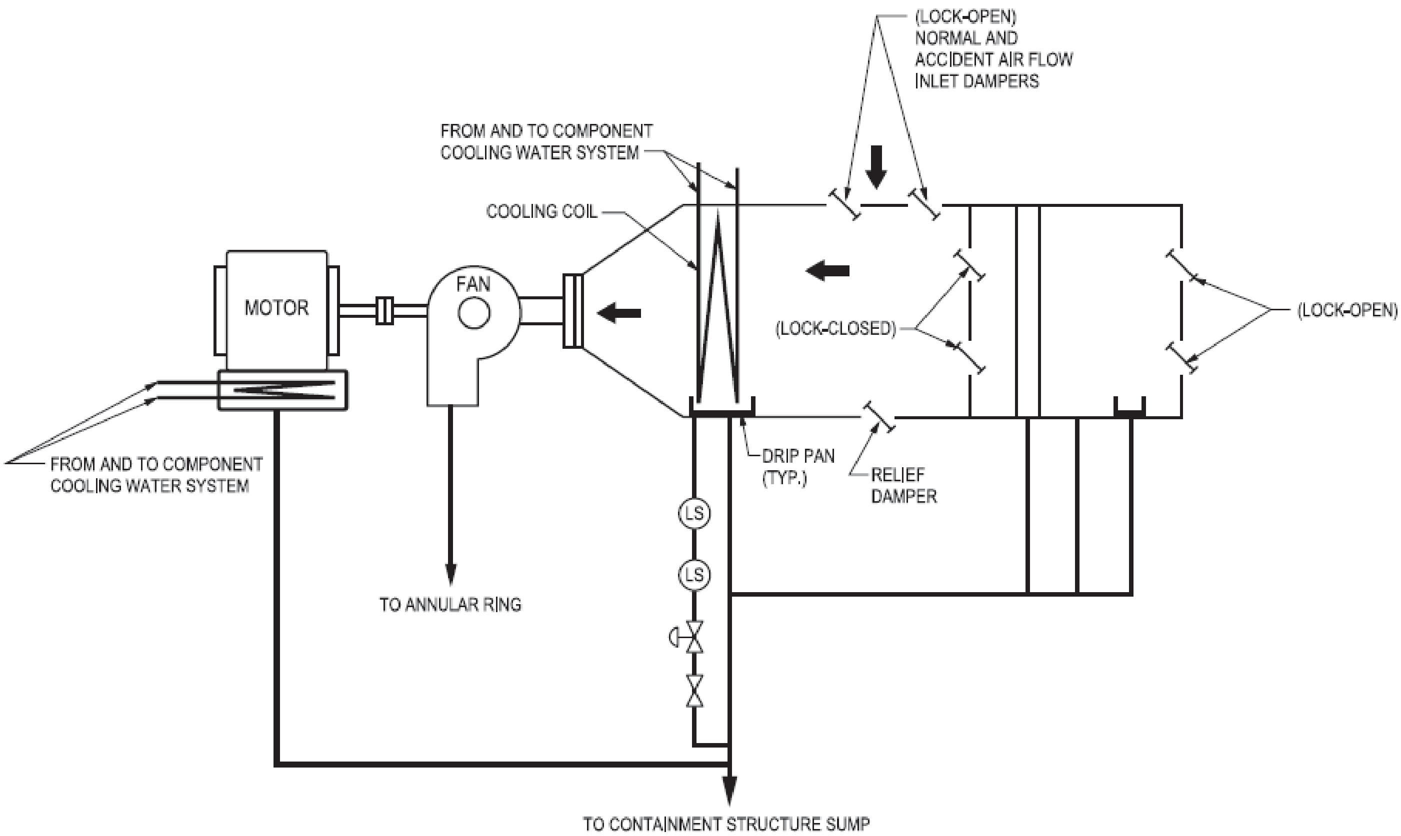
Revision 12 September 1998



NOTES:

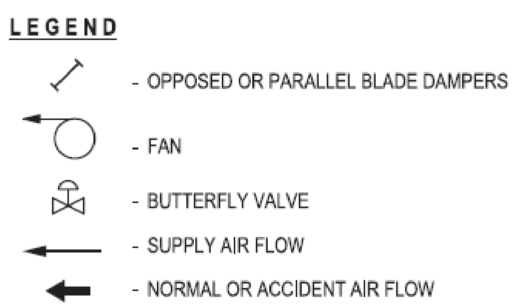
1. Represented by Symbol ⟨U⟩ on Drainage Piping Drawings.
2. It Shall be Furnished with Stainless Steel Grate.
3. Butt-Weld Connection to Match Schedule 10S
4. See Drainage Drawings for Pipe Size, Elevation, Etc.

FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 9.3-5 FLOOR DRAIN

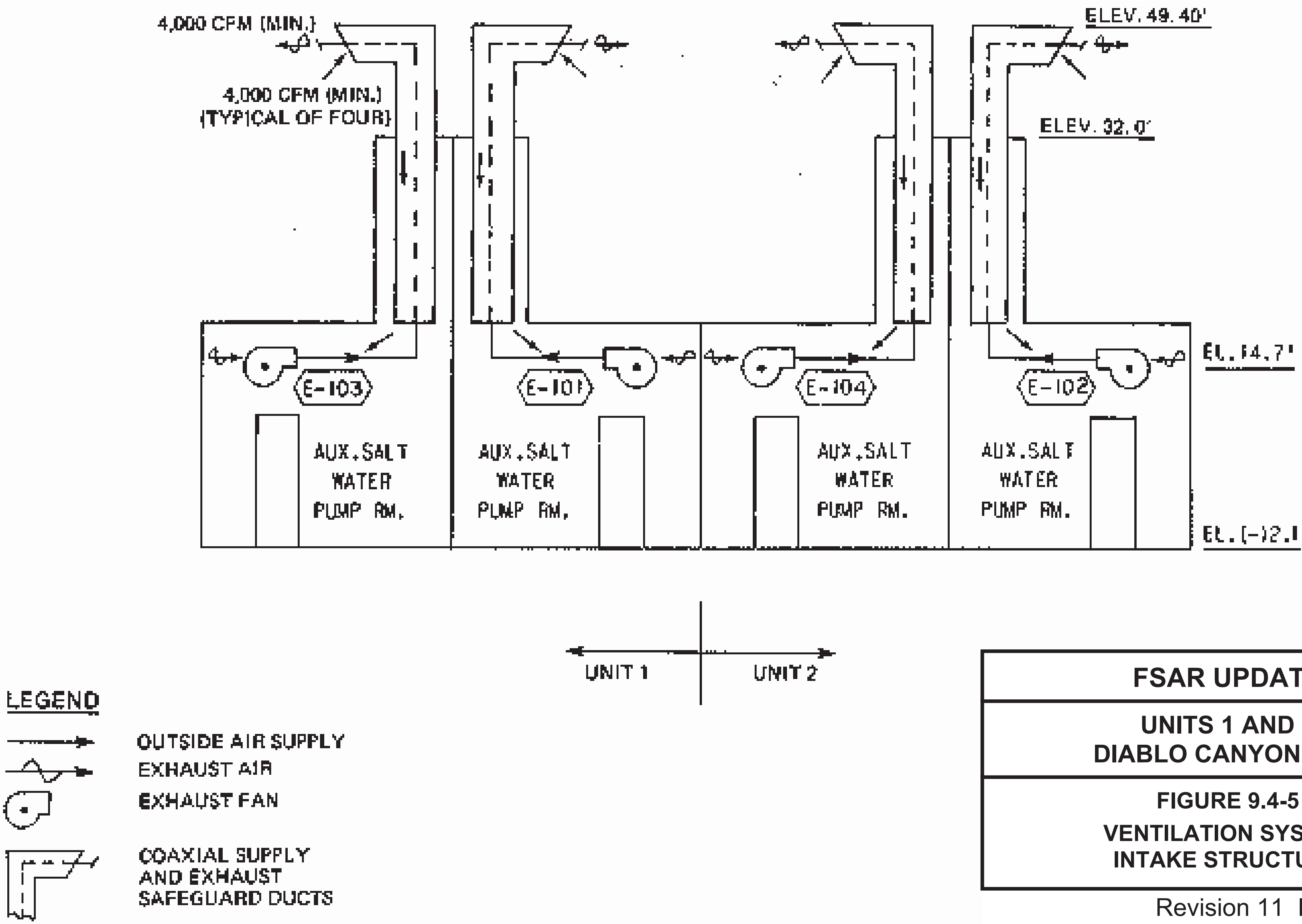


UNIT 1 CONTAINMENT FAN COOLER UNIT 1-1 (TYPICAL OF 5)

UNITS 1 AND 2 CONTAINMENT FAN COOLER UNITS ARE SIMILAR



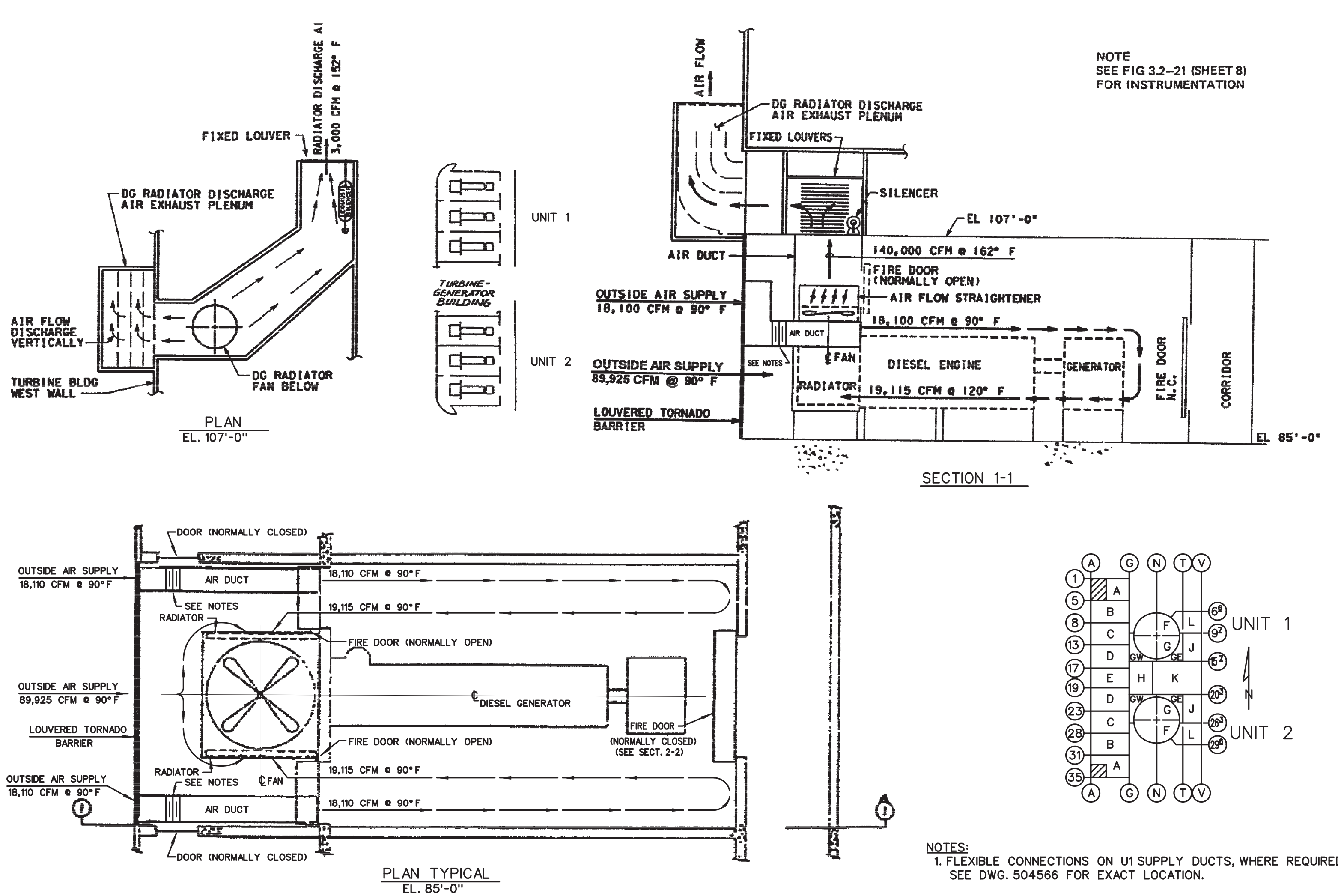
FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 9.4-4 CONTAINMENT FAN COOLER UNIT CONTAINMENT STRUCTURE



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.4-5
VENTILATION SYSTEM
INTAKE STRUCTURE



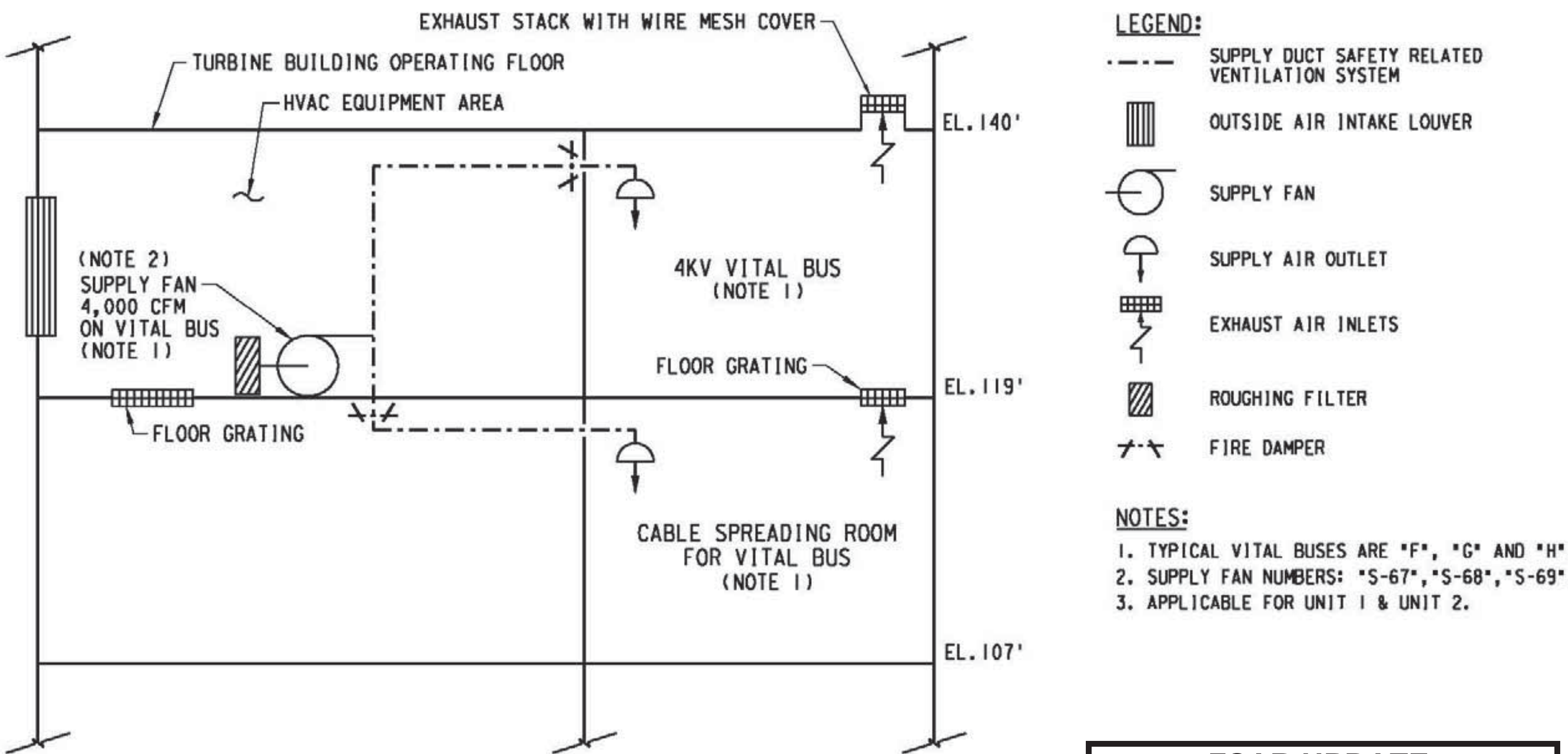
NOTES:

1. FLEXIBLE CONNECTIONS ON U1 SUPPLY DUCTS, WHERE REQUIRED. SEE DWG. 504566 FOR EXACT LOCATION.
2. FLEXIBLE CONNECTIONS ON U2 SUPPLY DUCT, WHERE REQUIRED. SEE DWG'S 521777 & 502591 FOR EXACT LOCATION.

FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.4-6
VENTILATION SYSTEM
DIESEL GENERATION

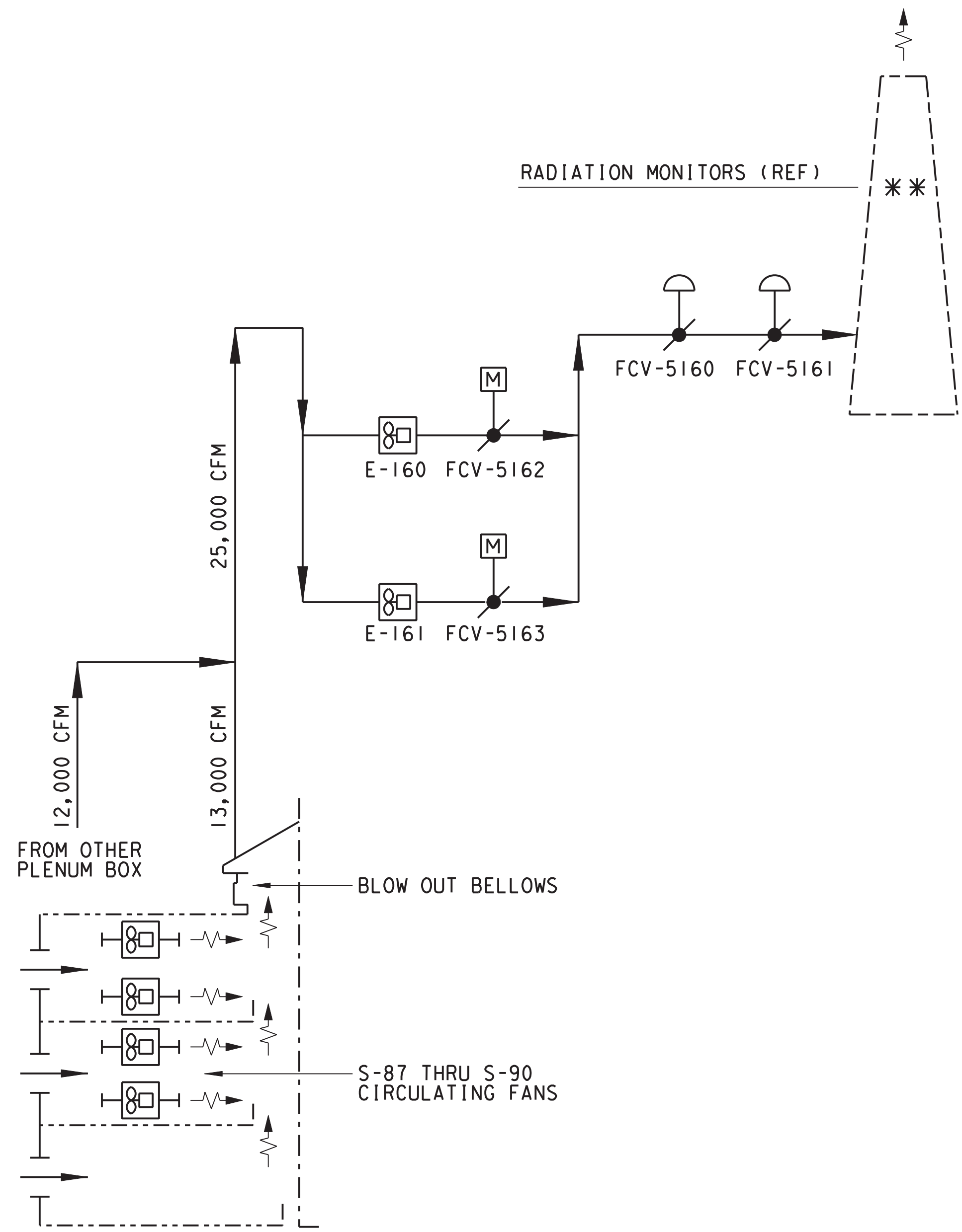


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.4-7

VENTILATION SYSTEM
4KV SWITCHGEAR ROOMS

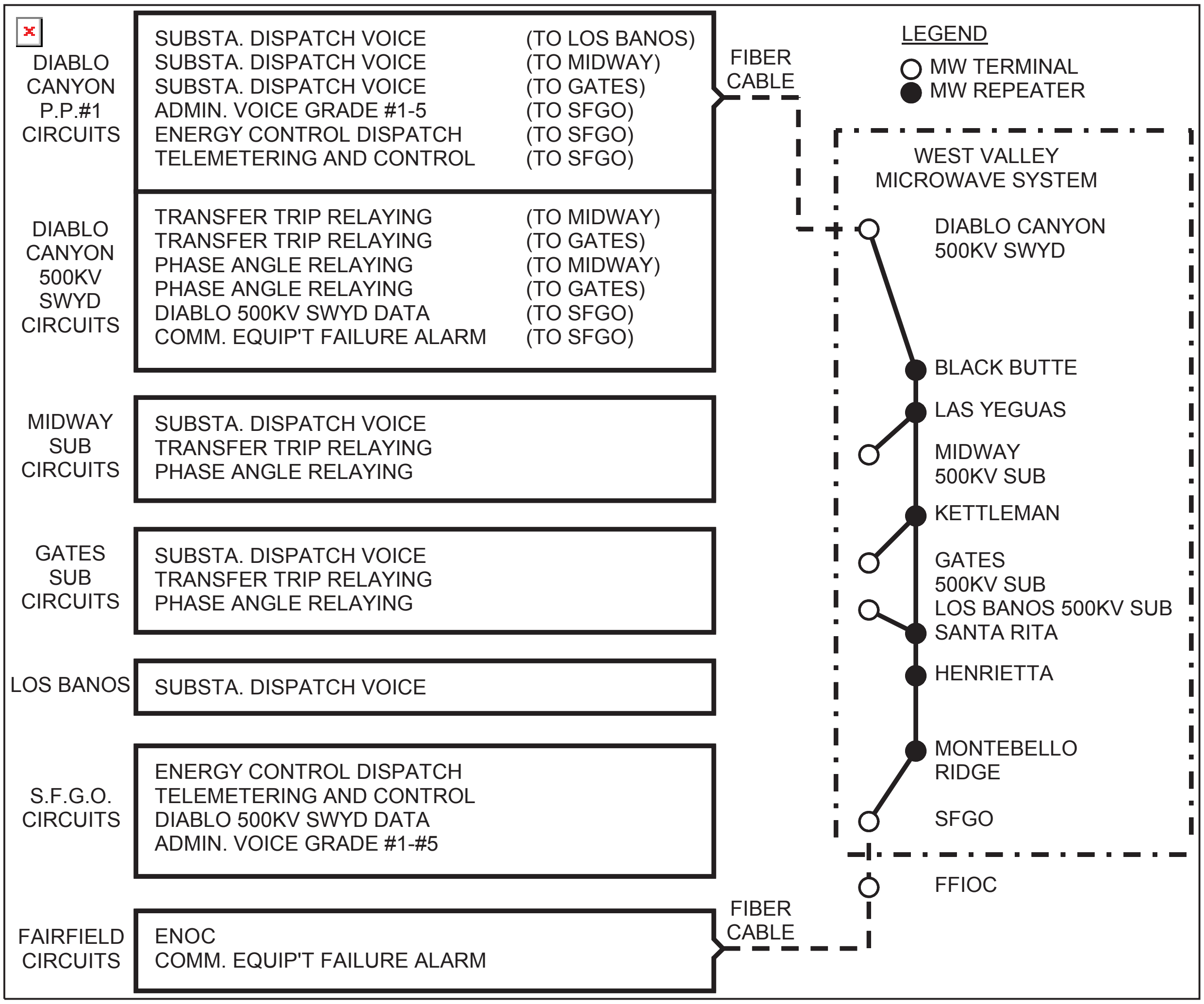


FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.4-11

VENTILATION SYSTEMS
CONTAINMENT PENETRATION
AREA GE/GW



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 9.5-5
PRIMARY COMMUNICATIONS SYSTEM