Chapter 12

RADIATION PROTECTION

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RADIATION PROTECTION

The purpose of this chapter is to demonstrate that both external and internal radiation dose resulting from operation of the Diablo Canyon Power Plant (DCPP) will be kept as low as is reasonably achievable (ALARA) and within applicable limits.

The principles and guidelines used in the design, construction, and operation of the radiation protection (RP) systems and programs described in Chapter 12 are specified in the individual sections of Chapter 12 and in Table 12.0-1.

12.1 RADIATION SHIELDING

This section describes the radiation shielding objectives and design configuration, identifies and characterizes source terms, summarizes important features of the area radiation monitoring system, describes those operating procedures that ensure external dose is kept ALARA, and gives estimates of dose to operating personnel and persons proximate to the DCPP site boundary.

12.1.1 DESIGN BASES

12.1.1.1 General Design Criterion 11, 1967 – Control Room

Adequate radiation shielding is provided to permit access to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel exceeding 10 CFR Part 20 limits.

12.1.1.2 General Design Criterion 19, 1999 – Control Room

Adequate radiation shielding is provided 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 accident.

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

Radiation shielding is provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR Part 20.

12.1.1.4 Radiation Shielding Safety Function Requirements

(1) Neutron Radiation Attenuation

The radiation shielding designs reduce potential neutron activation of equipment and mitigate the possibility of radiation-induced material damage.

(2) Non-Accident Unit Operation

The radiation shielding designs permit the continued operation of the other unit on the site in the unlikely event that a design basis accident (DBA) occurs at one unit.

12.1.1.5 10 CFR Part 20 – Standards for Protection Against Radiation

The radiation shielding designs support the protection of personnel from radiation sources such that doses are maintained below the limits prescribed in 10 CFR Part 20 and are ALARA. Note: Although personnel exposure limits must comply with the current regulation, the original shielding designs were to the pre-1994 regulation.

12.1.1.6 10 CFR 100.11 – Determination of Exclusion Area, Low Population Zone, and Population Center Distance

The radiation shielding designs provide adequate RP under tank rupture accident conditions to assure that direct radiation from plant structures is sufficiently low so that the total dose at the site boundary from both direct radiation and effluents is within the limits specified in 10 CFR 100.11.

12.1.1.7 10 CFR 50.67 – Accident Source Term

The radiation shielding designs provide adequate radiation protection under accidents analyzed using AST methodology, including a loss of primary coolant, to assure that direct radiation from plant structures is sufficiently low so that the total dose at the site boundary from both direct radiation and effluents is within the limits specified in 10 CFR 50.67.

12.1.1.8 Regulatory Guide 8.8, July 1973 – Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable (Nuclear Reactors)

The radiation shielding designs support the maintenance of occupational doses as low as practicable (i.e., ALARA).

12.1.1.9 NUREG-0737 (Items II.B.2, II.F.1, III.A.1.2, and III.D.3.4), 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 Used in Post-accident Operations: The plant radiation shielding configuration provides adequate access to vital areas and protection of PG&E Design Class I equipment during a postulated degraded core accident.

Item II.F.1 – Additional Accident-Monitoring Instrumentation

Position (2) – The radiation shielding designs support provisions for continuous sampling of plant effluents for post-accident releases of radioactive iodines and particulates and onsite laboratory capabilities.

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(f) - The technical support center (TSC) is provided with radiation shielding necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Item III.D.3.4 – Control Room Habitability Requirements: The radiation shielding for the control room is designed to ensure that the plant can be safely operated or shut down under DBA conditions.

It is noted that DCPP has incorporated a full implementation of Alternative Source Terms (AST) as defined in Regulatory Guide 1.183, July 2000, Section 1.2.1. The adequacy of the shielding associated with the control room and the technical support center have been re-evaluated using AST (refer to Section 15.5).

12.1.2 DESIGN DESCRIPTION

This section discusses the specific design criteria for individual radiation shielding systems required to achieve the overall objectives and describes the actual shielding design.

12.1.2.1 Radiation Shielding Locations and Basic Configurations

Figure 1.2-1 shows a plot plan of the site and indicates the location of roads, major plant buildings, and switchyards. It should be noted that the plant site is not served by railroad facilities. Figure 1.2-2 presents a detail of the plant layout and shows the location of outside tanks that could house potentially radioactive materials.

Figures 1.2-4 through 1.2-9 provide scaled plan views of Unit 1 buildings that contain process equipment for treatment of radioactive fluids, and indicate locations and basic configurations of the shielding provided. Figures 1.2-10 through 1.2-12 show similar views of Unit 2 structures. Corresponding sectional views of Unit 1 structures including shielding are shown in Figures 1.2-21 through 1.2-26. Comparable sectional views of Unit 2 structures are shown in Figures 1.2-28 through 1.2-30. Unit 1 and Unit 2 are similar with respect to radiation shielding design.

12.1.2.2 General Radiation Shielding Design Criteria and Features

One of the principal design objectives for plant radiation shielding is to reduce the expected radiation levels within plant structures to values that will allow plant personnel to gain access to normal work areas and remain there for sufficient time to perform required routine work without exceeding normal occupational dose limits. To implement this objective, plant areas capable of personnel occupancy are classified into one of five zones on the basis of expected frequency and duration of occupancy during routine operation, refueling, and maintenance. Note that zone maps show general zones and are not utilized within the RP program. A maximum design dose rate criterion is defined for each zone. Plant radiation shielding is designed to ensure that radiation dose rates in all plant areas are below the classified zone limits.

The radiation zone criteria are summarized in Table 12.1-1. The specific zoning for all plant areas during normal operation in Unit 1 is shown in Figures 12.1-1 through 12.1-12. Radiation zones for Unit 2 are similar to those for Unit 1.

Typical Zone 0 areas are the turbine building and turbine plant service areas, the control room, and the TSC. Typical Zone I areas are the auxiliary building work stations and corridors and the outer surfaces of the containment and auxiliary building, excluding penetration areas. Typical Zone II areas include some auxiliary building work stations and corridors, most areas of the FHB, and most areas on the 140 foot level of containment during reactor shutdown. Typical Zone III areas include most areas in the auxiliary building and containment outside the bioshield (during refueling) not previously identified as Zones I or II, excluding areas in close proximity to system components associated with dose rates more typical of Zone IV areas. These typical Zone IV areas are containment inside the bioshield during refueling and all of containment during power operations, areas that contain concentrating media such as demineralizers and filters, and all areas in close proximity o the following system components: CVCS, RCS, LRW, RHR, SI, and Charging. The post-accident radiation levels within the plant structures are discussed in Reference 1.

The radiologically controlled areas (RCAs) within plant structures (Zones I, II, III, and IV) are separated by barriers from the uncontrolled areas (Zone 0) to avoid inadvertent entry of unauthorized personnel. Entrance into the RCAs is normally made from a single access control station at the 85 foot elevation of the auxiliary building and is under procedural control. An auxiliary access control, located on the 140 foot elevation, may be utilized to provide more efficient access into the RCA, including containment buildings. Other access control stations may be temporarily established to support plant operations on an ad hoc basis. Within the RCAs, all areas are appropriately marked and/or barricaded in accordance with 10 CFR Part 20 and other applicable regulations. Areas designated Zone IV, such as the room containing the equipment and floor drain receiver tanks and the waste concentrator tanks, are accessible to plant personnel only at infrequent intervals, for limited periods of time, and then under strict radiological control. The dry active waste and resin liner storage areas in the radwaste storage building are also designated as Zone IV.

Care has been taken to ensure that RCA zones that are normally relatively low dose rate areas (i.e., Zones I and II) are not likely to be subjected to unexpected increases in dose rate due to the rapid introduction of radioactive materials into nearby process piping or other means. The routing of all plant piping is strictly controlled. Pipes that carry radioactive materials are routed in RCAs properly zoned for that level of activity.

Radiation shielding is arranged to protect personnel from direct gamma radiation that could otherwise stream through piping penetrations. Reach rods are provided where necessary to permit the operator to remain behind shielding while operating valves. For the radwaste storage building, exposure of site workers is minimized through the use of concrete shielding around the stored material, remote handling of high activity liners, and controlled access to the storage building.

12.1.2.3 Containment Radiation Shielding Design

Containment shielding is divided into four categories according to functions: primary shield, secondary shield, fuel handling shield, and accident shield. Each of these is discussed below.

12.1.2.3.1 Primary Radiation Shield

The primary shield consists of the core baffle, water annuli, barrel-thermal shield (all of which are within the reactor vessel), the reactor vessel wall, and a concrete structure surrounding the reactor vessel.

The primary shield (or parts thereof) performs the following functions:

(1) Reduces the energy-dependent neutron flux incident on the reactor vessel to prevent material property changes that might unduly restrict operation of the plant

(2) Attenuates reactor core neutron flux to prevent excessive activation of plant components and structures outside the primary shield

(3) Limits the gamma flux in both the reactor vessel and primary shield concrete to avoid large temperature gradients and/or dehydration of the concrete

(4) Reduces the radiation levels from reactor sources so that limited access is possible to certain areas within the reactor containment building during full power operation

(5) Reduces the residual radiation from the core to levels that will permit access to the region between the primary and secondary shields at a reasonable time after shutdown

The concrete structure immediately surrounding the reactor vessel extends up from the base of the containment and is an integral part of the main structural concrete support for the reactor vessel. It extends upward to join the reactor cavity. The reactor cavity, which is approximately rectangular in shape, extends upward to the operating floor.

The primary concrete shield is air-cooled to prevent overheating and dehydration from the heat generated by radiation absorption in the concrete. Eight "windows" are provided in the primary shield for insertion of the out-of-core nuclear instrumentation. Cooling for this instrumentation is also provided by air.

12.1.2.3.2 Secondary Radiation Shield

The secondary shield surrounds the primary shield and the reactor coolant loops and consists of the annular polar crane support wall, the concrete operating floor over the primary coolant loops, and the shell of the containment structure. The shell of the containment structure also serves as the accident shield.

The main function of the secondary shielding is to attenuate the radiation originating in the reactor and reactor coolant. Although the interior of the containment is a Zone IV area during full power operation, the secondary shielding is designed to reduce radiation levels to a point where limited access to certain areas within the containment is possible. The areas where limited accessibility is intended include the operating floor at elevation 140 feet and the annular areas between the crane wall and the containment shell on elevations 91 and 115 feet. The radiation levels in these areas are generally less than 15 mrem/hr. The secondary shield will also limit the full power dose rate outside the containment building to less than 1 mrem/hr.

12.1.2.3.3 Fuel Handling Radiation Shield

The reactor cavity, flooded during refueling operations, provides a temporary water shield above the components being withdrawn from the reactor vessel. The water height during movement of fuel assemblies is at least 23 feet above the reactor vessel flange. This height ensures that a minimum of 8 feet of water will be above the top of a withdrawn fuel assembly (about 9 feet of water above the active fuel). With upper internals in place, the water height during the unlatching of control rods is at least 23 feet above the fuel assemblies (at least 12 feet above the reactor vessel flange).

The fuel handling shield is designed to facilitate the removal and transfer of spent fuel assemblies and rod cluster control assemblies (RCCAs) from the reactor vessel to the spent fuel pool. It is designed to attenuate direct radiation from spent fuel and RCCAs to less than 2.5 mrem/hr at the refueling cavity water surface except during movement of a fuel assembly and as noted below. (Note that surface dose rates may typically be as high as 15 mrem/hr due to radioactive corrosion products in the water which is typically a much greater dose contributor than the fuel and RCCAs.

The fuel handling shield also provides attenuation of radiation from the reactor vessel internals. During removal of the upper internals package, the control rod drive shafts and guide tube assemblies must be raised above the water surface producing temporary radiation levels in excess of 1 R/hr. In the stored position, the very top of the control rod drive shafts extend from the surface producing localized dose rates to operators in the immediate area of less than 100 mrem/hr. The general area dose rate at the side of the pool is less than 5 mrem/hr near the upper internals. However, as noted above, surface dose rates may typically be as high as 15 mrem/hr due to radioactive corrosion products in the water.

The refueling canal is a passageway connected to the refueling cavity and extending to the inside surface of the reactor containment. The canal is formed by two concrete walls that extend upward to the same height as the refueling cavity. During refueling, the canal is flooded with borated water to the same height as the refueling cavity.

The spent fuel assemblies and RCCAs are remotely removed from the reactor containment through the horizontal spent fuel transfer tube and placed in the spent fuel pool. Concrete shielding and barriers protect personnel from radiation during the time a spent fuel assembly is being transferred from the containment to the spent fuel pool.

12.1.2.3.4 Accident Radiation Shield

The accident shield consists of the reinforced concrete cylindrical containment shell that is capped by a hemispherical reinforced concrete dome. This includes supplemental shielding for equipment and personnel hatches and the fuel transfer tube.

The equipment access hatch is shielded by a solid concrete block shadow shield. The main function of the accident shield is to reduce radiation levels outside the containment building to an acceptable level following a DBA.

12.1.2.4 Fuel Handling Area Radiation Shielding Design

Spent fuel is stored in the spent fuel pool located in the fuel handling area which is adjacent to the containment. The basic shield configuration for the Unit 1 spent fuel pool is shown in plan views in Figure 1.2-5 and in sectional views in Figures 1.2-23 and 1.2-24.

Water is used to provide shielding over the spent fuel assemblies so visual observation of fuel handling operations can be realized. The depth of the pool provides a submergence for the top of a fuel assembly of at least 8 feet during normal fuel handling operations and 23 feet submergence while fuel is stored in the fuel racks. Pool water level is indicated, and any water removed from the pool must be pumped out since there are no gravity drains.

The shielding for the fuel handling area restricts the dose rate to less than or equal to 5 mrem/hr in normally occupied areas.

Dose rates at the surface of the spent fuel pool will normally be less than or equal to10 mrem/hr.

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During transfer of a spent fuel assembly, the minimum water level above the active fuel is about 9 feet. With a peak fuel assembly (1.55 times full power level) being transferred, the maximum calculated dose rate at the surface of the pool is 50 mrem/hr. However, dose rates to the operator on the refueling platform are less than 20 mrem/hr. The calculated doses exclude any contribution to dose from radioactivity contained in the spent fuel pool water.

For additional information on the spent fuel pool water, refer to Section 9.1.3.2.

12.1.2.5 Auxiliary Building Radiation Shielding Design

The purpose of the radiation shielding in the auxiliary building is to protect personnel working near various system components in the chemical and volume control system (CVCS), the residual heat removal system, the waste disposal system, and the sampling system. The general layout of the shielding in the auxiliary building is shown on plan views of Figures 1.2-4 through 1.2-9. Sectional views are included in Figures 1.2-21 through 1.2-23, 1.2-25, and 1.2-26.

The shielding provided for the auxiliary building is designed to limit the dose rate during normal operation to less than 1 mrem/hr in normally occupied areas, and at or below 2.5 mrem/hr in areas requiring periodic occupancy. In addition, the auxiliary building shielding is designed to provide limited access to areas within the building during the long-term recirculation phase following a loss-of-coolant accident (LOCA).

The auxiliary building radiation shielding consists of concrete walls around equipment and piping that contain significant quantities of activity. Each equipment compartment is individually shielded so that compartments may be entered without having to shut down and/or decontaminate the adjacent system. In some cases, such as the tube withdrawal spaces for the abandoned boric acid and waste evaporators (refer to Figure 1.2-7), removable concrete block walls are provided to allow personnel access to equipment during maintenance periods. The shield material provided throughout the auxiliary building is regular concrete except for some of the shielding around the reactor coolant letdown filter, which is high-density concrete. The principal auxiliary building shielding provided is tabulated in Table 12.1-2.

12.1.2.6 Control Room Radiation Shielding Design

The control room radiation shielding consists of the concrete walls and roof of the control room. A plan view of the control room is shown in Figure 1.2-4, and sectional views are shown in Figures 1.2-25 and 1.2-26.

Normal radiation levels in the control room are less than 0.5 mrem/hr.

The limiting case for radiation shielding design is post-LOCA conditions. The control room radiation shielding limits the integrated doses under post-LOCA conditions to less than or equal to 5 rem Total Effective Dose Equivalent (TEDE) as specified in 10 CFR 50.67, GDC 19, 1999 and NUREG-0737, November 1980, Item III.D.3.4. For a discussion on post-accident shielding adequacy refer to Sections 6.4.1 and 15.5.17.2.4.

12.1.2.7 Technical Support Center Radiation Shielding Design

The TSC is designed to be habitable throughout the course of a DBA. Concrete shielding in the walls, roof, and floor limits the integrated doses under post-accident conditions to less than or equal to 5 rem TEDE consistent with the criterion for the TSC in Section 8.2.1(f) of NUREG-0737, Supplement 1, as amended by Regulatory Guide 1.183, Section 1.2.1, and 10 CFR 50.67. For a discussion on post-accident shielding adequacy refer to Section 15.5. Special labyrinth shields are furnished to cover each TSC doorway entrance to preclude significant dose contributions from radiation streaming.

12.1.2.8 Post-accident Sampling Compartment Radiation Shielding Design

The sampling compartment is shielded from external sources by concrete walls and concrete support columns.

12.1.2.9 Old Steam Generator Storage Facility

The old steam generators (OSGs) and old reactor vessel head assemblies (ORVHAs) were removed from DCPP Unit 1 and Unit 2 during the steam generator and reactor vessel head replacement projects. These ten large components are temporarily stored in the OSG storage facility (OSGSF) specifically constructed for this purpose. The OSGSF meets the radwaste storage requirements for temporary storage of the OSGs and ORVHAs until site decommissioning. The radiological design of the OSGSF meets the radiation shielding requirements of 40 CFR Part 190 and 10 CFR Part 20. The building is designed to have a maximum contact dose rate of 0.2 mrem/hr on the exterior wall surface. This value is less than and is bounded by the 0.5 mrem/hr radiation dose rate limitation requirement stated in Table 12.1-1 for the plant occupancy zone in which the OSGSF is located (Zone 0 – Unlimited Access). The building design also provides entrance doors with concrete labyrinths designed to provide shielding.

12.1.3 SOURCE TERMS

The normal full power sources utilized for shielding and dose calculations discussed in this this chapter are based on operation for 1 year at a core thermal power of 3568 MWt with an 85 percent capacity factor. The source terms were calculated using the EMERALD-NORMAL (Reference 2) computer code, and the source terms are assumed to be the maximum that would occur under either the design basis case or the normal operation case (including anticipated operational occurrences); both of these conditions are defined in Chapter 11. The isotopic source terms applicable to dose calculations are listed in the tables in Section 11.1 and Tables 12.1-3 through 12.1-13.

Maximum actual operating configuration is up to 21 months of operation, with a mixture of fuel with enrichments up to 5 percent, with maximum analyzed burnup of 50,000 MWD/MTU. Refer to the end of this section for an assessment of the impact of the actual maximum operating configuration on the adequacy of plant shielding.

The adequacy of shielding thickness for the shielded compartments was established during the original license application using computer code ISOSHLD (Reference 3). ISOSHLD performs gamma ray shielding calculations for isotopic sources in a wide variety of source and shield configurations. Attenuation calculations were performed by point kernel integration, with attenuation and buildup factors provided for shields with an effective atomic number of from 4 to 82. Section 15.5.8.4 provides a more detailed description of the code. For these shielding calculations, source and shield configurations were approximated by cylindrical or slab geometry, and the radiation exposure rates were calculated, using ISOSHLD, at all locations outside the shielded compartments where exposure to plant personnel is possible.

In addition, radiation dose rates were calculated for the storage tanks outside the auxiliary building; i.e., the primary water storage tank and the refueling water storage tank. Exposure rates were calculated, using ISOSHLD, immediately outside the tanks and at the site boundary (800 meters).

The results of these calculations are shown in Table 12.1-14. The calculations are for direct gamma exposure only; at distances such as 800 meters, the contribution from air-scattered gamma rays can increase the total dose rate by as much as a factor of 2 (Reference 4). The calculated exposure rates at the site boundary are small enough that any contribution from air-scattered gamma rays will still produce a negligible result.

A review was performed of the effect of the updated core inventory reported in Table 15.5-77, and the associated reactor coolant activity inventory with 1 percent fuel defects reported in Table 11.1-11A, on shielding adequacy. The assessment took into consideration the limit imposed on the reactor coolant concentrations by the plant technical specifications. The review demonstrated that the existing plant shielding remains adequate for safe plant operation, and that the current normal operation radiation zones are unaffected.

Refer to Section 12.2.3 for airborne radioactive source terms.

12.1.4 AREA MONITORING

The plant's area radiation monitoring system is described in detail in Section 11.4.

The area radiation monitoring system consists of fixed detectors mounted at the locations listed in Table 11.4-1.

The area radiation monitoring system is not required for safe shutdown of the plant. The principal purpose of the system is to alert personnel of increasing radiation levels in the monitored areas. Upon receipt of an alarm, the normal procedure is for operations personnel to investigate the cause and then take any action that is warranted. In general, the area radiation monitors have no automatic functions other than their alarm function. The exceptions to this are the instruments in the spent fuel and new fuel storage areas that automatically transfer the fuel handling building ventilation system (FHBVS) to the Iodine Removal Mode (refer to Section 9.4.4.2) and sound an alarm.

12.1.5 OPERATING PROCEDURES

The operating procedures that ensure external exposures will be kept ALARA can be grouped into three broad categories:

(1) Routine surveillance of the dose rate at various plant locations

(2) Preplanning and procedural control of radiation work

(3) Analysis of dose actually received

Each of these is discussed below:

(1) During the initial startup test program, a series of neutron and gamma dose rate measurements were performed to verify that there are no defects or inadequacies in the shielding that might hinder normal operation and/or maintenance activities. In addition, a comprehensive program of routine gamma dose rate measurements is an integral part of the plant RP program. This information is used to identify areas where special measures may be required to avoid unnecessary radiation exposure, to assist in the preplanning of work, and to help identify equipment malfunctions that lead to increased dose rates. Radiation areas are appropriately posted and/or barricaded in accordance with the requirements of 10 CFR Part 20 and the plant Technical Specifications (Reference 6).

(2) Under the provisions of the plant RP program, all radiation work is carried out under a radiation work permit. These work permits are instruction

sheets intended to ensure that appropriate precautions will be taken during the performance of all radiation work. As such, they specify protective clothing requirements, monitoring requirements, dosimetry requirements, expected radiation conditions, and any special measures required to control the dose received by personnel. Such special measures might include limiting the stay time in an area, erection of temporary shielding, use of remote handling tools, or other techniques appropriate to the specific situations. Personnel are instructed in RP in accordance with specific procedures established in Volume I of the Plant Manual.

(3) Self-reading dosimeters, coupled with the results of the thermoluminescent dosimeters (TLDs), are routinely checked by RP personnel to verify that each individual's exposure, as shown on the individual's permanent record, is within expected values. If a person's exposure appears to be higher than estimated, RP personnel investigate and initiate corrective action. Radiation workers are responsible for remaining cognizant of their current exposure status.

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12.1.6 ESTIMATES OF EXPOSURE

An assessment of the expected radiation dose to individuals as a result of DCPP operations was performed as part of the original license application. Provided below is a summary of the results and general conclusions:

(1) The annual man-rem external exposure in offsite locations resulting from direct shine from plant structures containing radioactive materials was extremely small. For example, the annual continuous occupancy dose at a distance of 800 meters contributed by direct shine from the containment was calculated to be approximately 1.5 mrem using the conservative assumption that the dose rate on its exterior surface was the maximum design value of 1 mrem/hr.

(2) The man-rem exposure to the general public was, for all practical purposes, the result of airborne and liquid releases from the radioactive waste disposal system. Although this exposure was very low, numerical estimates were made and are presented in Sections 11.2.2.7 and 11.3.2.5 for liquid and gaseous releases, respectively.

(3) Estimates of personnel exposures were obtained from surveys of exposure at other operating plants and from calculations based on anticipated occupancy times for various job classifications in various areas within the plant. The calculated exposures compared reasonably well with those experienced at other plants.

12.1.6.1 Calculated Exposure Estimates

The annual exposure to plant personnel for normal operation of the two units was calculated to be about 50 man-rem. This value was derived from anticipated occupancy times for various job classifications in various areas within the plant. The dose rates assigned to the various areas were based on normal plant operation assuming approximately 0.2 percent fuel defects. Table 12.1-15 presents a summary of the calculated values of man-rem exposure on the basis of occupancy factors listed in Table 12.2-17 and dose rates in various areas.

Experience at other pressurized water reactors has shown that normal operational activities generally account for only part of a plant's total exposure. Hence, the total estimated annual dose with both units operating, and including special maintenance and refueling activities, was about 400 man-rem.

12.1.6.2 Exposure Estimates Based on Operating Plant Experience

Reference 5 reports that for 1981 the annual average collective dose from a pressurized water reactor was 652 man-rem.

12.1.6.3 Exposure Estimates for Diablo Canyon Power Plant

Based on the above described exposure estimates from both analytical predictions and records of exposures at actual operating plants, it was believed that 200 man-rem per year per unit represented a reasonable estimate of the maximum total exposure to be expected for performance of all normal operations, testing, and maintenance at DCPP. The exposure for two-unit operation should be somewhat less than double the value for operation of one unit, since certain facilities, such as the radwaste treatment system, are common to both units.

12.1.7 SAFETY EVALUATION

12.1.7.1 General Design Criterion 11, 1967 – Control Room

The radiation dose in the control room under normal conditions is well below the limits specified in 10 CFR Part 20 (refer to Sections 12.1.2 and 12.1.3).

Areas outside the control room that are necessary to shut down and maintain safe control of the facility under normal operating conditions are provided with adequate radiation shielding such that operator dose is well below the limits specified in 10 CFR Part 20 (refer to Sections 12.1.2 and 12.1.3).

12.1.7.2 General Design Criterion 19, 1999 – Control Room

The control room shielding, in conjunction with the control room ventilation system (CRVS) 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 DBA (refer to Sections 12.1.2 and 12.1.3). An evaluation of post-accident control room radiological exposures is presented in Section 15.5.

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

Radiation shielding from spent fuel storage is provided by the concrete walls of the spent fuel pool and the depth of the water in the pool (refer to Sections 12.1.2, 12.1.3, and 9.1.2).

The purpose of the radiation shielding in the auxiliary building is to protect personnel working near various systems containing radioactivity from doses in excess of 10 CFR Part 20 limits (refer to Section 12.1.2.5). The purpose of radiation shielding in the radwaste building is to protect personnel working around the stored radwaste material (refer to Section 12.1.2.2).

12.1.7.4 Radiation Shielding Safety Function Requirements

(1) <u>Neutron Radiation Attenuation</u>

Refer to Section 4.2.2.5.4 for an evaluation of neutron radiation attenuation inside the concrete primary shield. Outside the concrete primary shield, neutron radiation is reduced to levels where neutron activation of equipment is not a concern (refer to Section 12.1.2.3.1).

(2) <u>Non-Accident Unit Operation</u>

During accident conditions, continued operation of the non-accident unit is made possible by the habitability of the shared control room (refer to Sections 6.4.1, 12.1.2, and 15.5).

12.1.7.5 10 CFR Part 20 – Standards for Protection Against Radiation

The regulations of 10 CFR Part 20 limit the Total Effect Dose Equivalent (TEDE) to 5 rem per year. Pacific Gas and Electric Company limits TEDE to 5 rem per year with guidelines for maintaining doses at levels below this value (refer to Sections 12.1.2 through 12.1.4).

If operating experience reveals areas where exposure problems exist, appropriate changes will be made in plant shielding, source strengths, locations, or operating practices as required to maintain personnel doses ALARA (refer to Section 12.1.5).

Refer to Section 12.1.3 for a discussion of the determination of radiation levels and radioactive material concentrations within structures, systems and components of the plant that could affect direct radiation exposures to members of the public.

12.1.7.6 10 CFR 100.11 – Determination of Exclusion Area, Low Population Zone, and Population Center Distance

Radiation shielding designs ensure that direct radiation from plant structures is sufficiently low so that the total dose at the site boundary from both direct radiation and effluents is within the limits specified in 10 CFR 100.11 for tank rupture accident conditions (refer to Sections 12.1.2 and 15.5).

12.1.7.7 10 CFR 50.67 – Accident Source Term

Radiation shielding designs ensure that direct radiation from plant structures is sufficiently low so that the total dose at the site boundary from both direct radiation and effluents is within the limits specified in 10 CFR 50.67 for accidents analyzed using AST methodology (refer to Sections 12.1.2 and 15.5).

12.1.7.8 Regulatory Guide 8.8, July 1973 – Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable (Nuclear Reactors)

The radiation shielding designs, where practicable, separate radiation sources from areas where personnel have normal or routine access. Movable shielding is provided where permanent shielding is impractical. Shielding is provided in areas containing radioactive wastes. Refer to Section 12.1.2 for detailed discussions of the plant shielding designs.

12.1.7.9 NUREG-0737 (Items II.B.2, II.F.1, III.A.1.2, and III.D.3.4), 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: A post-accident radiation shielding design review for DCPP, as required by NUREG-0737, November 1980 (Reference 7), was performed and is reported in Reference 1.

Adequate radiation shielding is provided to prevent the degradation of PG&E Design Class I equipment. Also, the control room, TSC, and switchgear rooms are the vital areas requiring access and occupancy during post-accident conditions. All three of these rooms, as well as access pathways, 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 (refer to Sections 6.4.1.3.16, 12.1.2, and 12.1.3).

Item II.F.1 – Additional Accident Monitoring Instrumentation

Position (2) – Plant vent high range iodine and particulate sampling may be performed by transferring the radiation monitor filter cartridges to the TSC laboratory. A lead transfer carriage is utilized to minimize personnel dose during the transfer of the cartridges.

Item III.A.1.2 – Upgrade Emergency Support Facilities:

Section 8.2.1(f) – Radiation shielding for the TSC, in conjunction with the TSC ventilation system, maintains TSC radiation exposures within 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, consistent with the criteria for habitability provided in NUREG-0737, Supplement 1, January 1983, Item 8.2.1(f) (refer to Sections 6.4.2.3.4, 12.1.2, and 12.1.3).

Item III.D.3.4 – Control Room Habitability Requirements: Calculations indicate that shielding thicknesses are adequate to limit post-LOCA dose rates inside the control room from all potential direct shine radiation sources in the auxiliary building, containment building, and containment penetration area to less than 1 mrem/hr. In addition, although radiation streaming from a possible radiation cloud could result in local hot spots near the control room doorway entrance adjoining the turbine building, the radiation shielding provided by the design of the control room is sufficient to permit unlimited personnel occupancy of the control room during post-LOCA operations (refer to Sections 12.1.2, 12.1.3, and 15.5.17).

It is noted that DCPP has incorporated a full implementation of Alternative Source Terms (AST) as defined in Regulatory Guide 1.183, July 2000, Section 1.2.1. The adequacy of the shielding associated with the control room and technical support center have been re-evaluated using AST (refer to Section 15.5). However, the estimated short-term operator mission doses while performing vital functions post-LOCA, continue to be based on TID-14844 assumptions as documented in Reference 1. This approach is acceptable based on the AST benchmarking study reported in SECY-98-154 (Reference 8) which concluded that results of analyses based on TID-14844 would be more limiting earlier on in the event, after which time the AST results would become more limiting. Post-LOCA access to vital areas usually occurs within the first one or two weeks when the original TID-14844 source term remains limiting.
Comparison of the missions documented in Reference 1, to the missions required per the current Emergency Operating Procedures (EOPs) indicates that a) there are no new destinations or access paths, b) many of the previously evaluated actions are deemed either nonessential or no longer required, and c) of the access requirements that are still valid, the required time for access and/or the access duration has changed in some instances. Review of the impact of the updated access requirements as well as the impact of changes in fuel design, enrichment and burnup implemented since issuance

of Reference 1 indicate that the operator mission doses associated with actions required by the current EOPs continue to remain below the regulatory limits provided in NUREG 0737, Item II.B.2.

12.1.8 REFERENCES

1. Diablo Canyon Units 1 and 2 Radiation Shielding Review, Revision 3, June 1984.

2. S. G. Gillespie and W. K. Brunot, EMERALD NORMAL - A program for the Calculation of Activity Releases and Doses from Normal Operation of a Pressurized Water Plant, Program Description and User's Manual, Pacific Gas and Electric Company, March 1973.

3. R. L. Engel, et al, ISOSHLD - A Computer Code for the General Purpose Isotope Shielding Analysis, BNWL-236, UC-34, Physics, Pacific Northwest Laboratory, Richland, Washington, June 1966.

4. Reactor Handbook, Second Edition, Volume III, Part B, Oak Ridge National Laboratory, 1962.

5. Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1981, NUREG-0713, Vol. 3, Nov. 1982.

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

7. NUREG 0737, Clarification of TMI Plan Requirements, USNRC, November 1980.

8. SECY-99-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," June 30, 1998.

12.2 VENTILATION

The ventilation systems at DCPP are designed to provide a suitable environment for personnel and equipment and also remove radioactive materials from the ventilation flows prior to release to the atmosphere during normal plant operation, including anticipated operational occurrences. These ventilation systems are described in this section, including the associated airborne radioactivity monitoring functions. Also included are the assumptions that were made as part of the original license to calculate normal operation airborne activity concentrations as well as estimates of inhalation exposure. The cooling function of the ventilation systems, including post-accident fission product removal functions, if any, are described in detail in Section 9.4.

In performing these atmospheric cleanup functions, the plant ventilation systems support the RP Program (refer to Section 12.3) by keeping radiation doses ALARA.

Parts of the ventilation systems also perform PG&E Design Class I functions such as cooling of engineered safety feature (ESF) motors, post-accident containment heat removal, and ensuring post-accident control room and TSC habitability. These are described in detail in Sections 6.4 and 9.4.

12.2.1 DESIGN BASES

12.2.1.1 General Design Criterion 17, 1967 – Monitoring Radioactivity Releases

The ventilation systems are 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.

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

The ventilation systems are provided with monitoring and alarm instrumentation for fuel and waste storage and handling areas for conditions that might contribute to radiation exposures.

12.2.1.3 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The ventilation systems include those means necessary to maintain control over the plant radioactive effluents during normal operation, including anticipated operational occurrences and during accidents.

12.2.1.4 10 CFR Part 20 – Standards for Protection Against Radiation

The ventilation systems maintain airborne radioactive material concentrations in normal work areas in the auxiliary building, fuel handling area, and turbine building within the maximum permissible concentration (MPC) values given in 10 CFR 20.1-20.601, Appendix B, Table I. Note: Although personnel exposure limits must comply with the current regulation, the original ventilation designs were to the pre-1994 regulation.

In addition, the ventilation systems provide the ability to maintain and/or reduce the airborne radioactive material concentrations in normally unoccupied areas within the plant structure to levels that will allow periodic access as required for nonroutine work.

12.2.1.5 10 CFR Part 50 Appendix I – Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents

The ventilation systems operate in conjunction with other gaseous waste disposal equipment to ensure that the dose from concentrations of airborne radioactive materials in unrestricted areas beyond the site boundary are within the limits specified in 10 CFR Part 50, Appendix I.

12.2.1.6 Regulatory Guide 8.8, July 1973 – Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable (Nuclear Reactors)

The ventilation systems are designed to support the maintenance of occupational doses as low as practicable (i.e., ALARA).

12.2.2 DESIGN DESCRIPTION

The following paragraphs present brief descriptions of the ventilation systems for each of the major plant structures. The descriptions include building volumes, flowrates, and filter characteristics that were used when estimating airborne activity concentrations in the various plant areas in support of the original licensing. As noted, more complete design descriptions of the ventilation systems can be found in Section 9.4.

12.2.2.1 Containment Ventilation Systems

Detailed descriptions of the containment iodine removal and ventilation systems, including design criteria, are provided in Section 9.4.5. In terms of RP during normal operation, these systems include:

(1) Containment purge supply and exhaust system

(2) Iodine removal units

The containment purge system includes a single supply fan and a single exhaust fan. Supply air is drawn from the atmosphere through a roughing filter. The purge exhaust fan draws air from the main ventilation header in the containment and exhausts it to the plant vent, from which it is released to atmosphere at the top of the containment. The purge exhaust air is not filtered. This system is not in operation during power operation, but is provided for use as required prior to personnel entry during modes 5 and 6.

Each containment building is provided with two iodine removal units consisting of a recirculation fan complete with roughing filter, high-efficiency particulate air (HEPA) filter, and charcoal filter on the fan suction. These units are operated as required during normal operation to control airborne iodine and particulate concentrations in the containment atmosphere.

Parameters used for the normal operation containment airborne activity concentration analysis developed in support of the original licensing are presented in Table 12.2-1.

12.2.2.2 Control Room Ventilation System

A detailed description of the CRVS, including design criteria, is provided in Section 9.4.1. During normal operation, the quantity of potentially radioactively contaminated air entering the control room is controlled by CRVS MODE 1 in which 73 percent of the control room air is recirculated, 27 percent of the air is outside makeup, and 100 percent of the air is passed through roughing filters.

Parameters used for the normal operation airborne activity concentration analysis developed in support of the original licensing are presented in Table 12.2-2.

12.2.2.3 Auxiliary Building Ventilation System

A detailed description of the auxiliary building ventilation system (ABVS), including design criteria, is provided in Section 9.4.2. Briefly, the system for each unit contains two full-capacity supply fans that draw air from the atmosphere just above the auxiliary building and then discharge it to the occupied areas of the building and to the ESF pump compartments whenever they are in operation. Two full-capacity exhaust fans draw air from various locations throughout the building and discharge it to the plant vent, where it is released at the top of the containment.

Under normal circumstances (i.e., Building Only Mode), the exhaust air is passed through a roughing filter and HEPA filter prior to entering the vent.

In all modes of operation, the ventilation flow patterns are designed so that the air flows from areas of lower potential contamination to areas of higher potential contamination. The system is balanced so that the building is normally under a slight negative pressure.

Parameters used for the normal operation airborne activity concentration analysis developed in support of the original licensing are presented in Table 12.2-3.

12.2.2.4 Fuel Handling Building Ventilation System

A detailed description of the FHBVS, including design criteria, is provided in Section 9.4.4. Two full-capacity supply fans discharge into duct work in the corridors and equipment compartments below the spent fuel pool floor. Three full-capacity exhaust fans are provided. They collect air from along one side of the pool, just above the surface. In this manner, the air provides a sweeping action over the surface of the pool. During the normal mode operation, one non-Class 1E exhaust fan is in operation and the air is passed through a roughing and HEPA filter before being discharged to the plant vent.

Parameters used for the normal operation airborne activity concentration analysis developed in support of the original licensing are given in Table 12.2-4.

12.2.2.5 Turbine Building Ventilation

A detailed description of the turbine building ventilation system, including design criteria, is provided in Section 9.4.3. Ventilation in the turbine building is provided by a number of cabinet fans mounted on the exterior wall of the building. These fans draw air from the surrounding atmosphere into the building through roughing filters. The air is discharged from the roof of the building without treatment. This system is intended primarily to provide personnel comfort since the potential for introduction of airborne radioactivity into the turbine building, as a result of water or steam leakage from the steam system, is very low.

The volume of the turbine building served by the cabinet fans is 5.125×10^6 cubic feet (one unit). The ventilation flowrate is 420,000 cfm.

12.2.2.6 Technical Support Center Ventilation

A detailed description of the TSC ventilation system, including design criteria, is provided in Section 9.4.11. The TSC is provided with its own ventilation system. Self-contained air conditioning units are also provided for the operations center and laboratory area.

12.2.2.7 Post-Accident Sampling Compartment Ventilation

A detailed description of the post-accident sampling compartment ventilation system, including design criteria, is provided in Section 9.4.10. During normal operation, a ventilation fan delivers 300 cfm of outside air to the post-accident sampling compartment. This 300 cfm then exits the compartment through exfiltration.

12.2.3 SOURCE TERMS

12.2.3.1 Auxiliary Building Source Terms

The ABVS has been designed to prevent the transport of airborne radioactive materials into normal work areas. For example, equipment representing potential sources is located in compartments off the main corridors, with the ventilation flow directed from the corridors to the compartments and then to the plant vent. As a result, the occurrence of a situation wherein an equipment leak would introduce radioactive materials into the air of a normally occupied area is minimized. However, in support of the original plant design, for purposes of estimating the maximum air activity concentrations that could occur in normally occupied operating spaces of the auxiliary building, the following source terms were assumed:

(1) Two-unit leakage of 20 gpd per unit of primary coolant at 0.2 percent fuel defects uniformly distributed in the auxiliary building main corridors (volume = 370,000 cubic feet) with a ventilation exhaust flow of 75,000 cfm

(2) Partition factors of 0.005 for iodines, 1 for noble gases, and 0.26 for tritium as tritiated water.

No credit was taken for condensation of tritiated water or plateout of iodines.

The results of this analysis are presented in Table 12.2-5.

The maximum expected airborne activity concentrations during normal operation occur within the CVCS letdown heat exchanger room, the volume control tank room, the charging pump rooms, and the gas decay tank rooms. Occasional entry may be required into these areas during the course of normal operations. Access to these areas will be under procedural control at all times. Thorough radiation surveys will be conducted prior to access to these spaces so that necessary controls can be prescribed to limit personnel exposure. It should be emphasized that the airborne activity concentrations calculated as part of the original plant design for these rooms are the maximum that could occur in spaces where access is controlled, and do not reflect the anticipated concentrations in areas of normal occupancy.

The source term for the CVCS letdown heat exchanger room was based on the following assumptions:

(1) CVCS leakage of 1 gpd of hot primary coolant at 0.2 percent fuel defects occurs upstream of the letdown heat exchanger

(2) The volume of the compartment is taken to be 6500 cubic feet with a ventilation flowrate of 1200 cfm

(3) Partition factors of 0.10 for iodines, 1 for noble gases, and 0.35 for tritium as tritiated water are assumed

The source term for the volume control tank room was based on the following assumptions:

(1) CVCS leakage of 10 gpd of cold primary coolant at 0.2 percent fuel defects occurs upstream of the tank

(2) The room volume is taken to be 2140 cubic feet with a ventilation flowrate of 600 cfm

(3) Partition factors are assumed to be 0.001 for iodines, 1 for noble gases, and 0.01 for tritium as tritiated water

The source term for the charging pump compartment was based on the following assumptions:

(1) CVCS leakage of 10 gpd of cold primary coolant at 0.2 percent fuel defects occurs upstream of the pump

(2) The compartment volume is taken to be 3900 cubic feet with a ventilation flowrate of 400 cfm

(3) Partition factors of 0.001 for iodines, 1 for noble gases, and 0.01 for tritium as tritiated water are assumed

The source term for the gas decay tank compartment is based on the following assumptions:

(1) Gas decay tank leakage of 0.01 scfm is assumed with tank activity inventory as shown in Table 11.3-5

(2) The compartment volume is taken to be 3490 cubic feet with a ventilation flowrate of 40 cfm

(3) A partition factor of 1 is assumed for noble gases at the leakage point

The resulting maximum airborne activity concentrations in these spaces during normal operation are summarized in Tables 12.2-6 through 12.2-9. (Note that the actual ventilation flowrates for the above rooms are higher than the assumed values used for the source term analysis. The higher flowrates would result in lower airborne activity concentrations in these spaces and would be enveloped by the values shown in Tables 12.2-6 through 12.2-9.)

12.2.3.2 Fuel Handling Area Source Term

Airborne activity in the fuel handling area is produced primarily from tritium evaporation and iodine and noble gas partitioning from the spent fuel pool. The evaporation of tritium is discussed in Section 11.2.2.5.2, and the calculated airborne tritium concentrations above the spent fuel pool as a function of plant operating time, developed as part of original plant design, are shown in Figure 11.2-7. The iodine and noble gas releases from the spent fuel pool were based on the following assumptions:

(1) Fuel handling area volume of 4700 cubic feet with a ventilation flowrate of 35,750 cfm

(2) Partition factors of 0.001 for iodines and 1 for noble gases

(3) Spent fuel pool activity concentrations and production rates are listed in Table 12.2-10

The resulting airborne activity concentrations during normal operation in the fuel handling areas are summarized in Table 12.2-11.

12.2.3.3 Containment Source Term

The source term developed as a part of original plant design for containment airborne activity during normal operation was based on the following assumptions:

(1) Leakage of 240 lb/day of primary coolant at 0.2 percent fuel defects

(2) Partition factors of 0.10 for iodines, 1 for noble gases, and 0.35 for tritium as tritiated water at the leakage point

(3) Ninety days of activity accumulation. No credit taken for plateout, containment leakage, cleanup recirculation unit operation, or other activity removal except natural decay

The resulting airborne activity concentrations developed as part of original plant design are listed in Table 12.2-12.

12.2.3.4 Turbine Building Source Term

The source term developed as part of original plant design for the turbine building was based on the following assumptions:

(1) Two-unit main steam leakage of 1700 lb/hr per unit and condenser water leakage of 5 gpm per unit into the turbine building based on 20 gpd per unit of primary-to-secondary system leakage of primary coolant with 0.2 percent fuel defects

(2) Partition factors of 1 for noble gases, iodines, and tritium for steam leakage at the point of leakage

(3) Partition factors of 0.001 for iodines and 0.01 for tritium as tritiated water for condenser water leakage

(4) Turbine building volume of 10.25 x 10^6 cubic feet with a ventilation flowrate of 840,000 cfm (two units)

The resulting airborne activity concentrations during normal operation developed as part of original plant design are listed in
Table 12.2-13.

12.2.3.5 Control Room Source Term

The source terms developed as part of original plant design for the control room are assumed to result from the total plant gaseous waste releases as indicated in Table 11.3-3. The airborne activity concentration at the control room intake was calculated using an assumed annual average χ/Q of 1.78 x 10^{-4} sec/m^3, and the total gaseous release from both units.

The source term for the control room itself was calculated using the following assumptions:

(1) Intake airborne activity concentrations developed as part of original plant design are provided in Table 12.2-14

(2) Control room MODE 1 operation with intake and exhaust flowrates assumed to be 4200 cfm. The control room volume is taken as 125,000 cubic feet

(3) No credit is taken for filtration or other removal of activity from the incoming air

The resulting control room airborne activity concentrations for normal operation developed in support of original plant design are presented in Table 12.2-15.

12.2.3.6 Technical Support Center Source Term

The TSC airborne activity concentrations for normal operation are expected to be similar to those in the control room.

12.2.4 AIRBORNE RADIOACTIVITY MONITORING

The instruments and methods used for airborne radioactivity monitoring include certain channels in the process monitoring system, the plant area monitoring system, continuous air monitors (CAMs), and portable low volume air samplers.

12.2.4.1 Process and Area Monitoring Systems

The process and area monitoring systems (including particulate collection) are described in detail in Section 11.4. The monitors, with their readout locations, are listed in Table 11.4-1.

Based on operational data, permanently installed air particulate and gas monitors (APGMs) may be correlated against air samples collected in close proximity to the sample collection point. Grab samples are gross counted and analyzed for isotopic and quantification as appropriate. The response of the APGMs during the period of grab sampling may be correlated to the total μCi/cc measured in the grab sample and this correlation may be used to develop the instrument response in counts per minute versus concentration in μCi/cc. The effect of ambient background is taken into account. Experience has shown that the vast majority of such samples are statistically indistinguishable from background.

Correlation frequencies may be established that are appropriate for the specific instrument involved based on considerations such as likely variation in isotopic mixture, history of the instrument in terms of calibration shift, use of the instrument for quantitative work, and the potential for a statistically significant measured value above background resulting from licensed material.

12.2.4.2 Grab Sampling Program

The grab sampling program consists of collection of air moisture for tritium analysis and air for noble gas particulate and halogen analysis. The location and frequency of the samples are determined based on the potential for a statistically significant measured value above background resulting from licensed material. Some samples may be scheduled on a periodic basis.

12.2.4.2.1 Tritium and Noble Gas Analyses

Collection of air moisture for tritium analysis and air for noble gas analysis may be performed during certain activities such as flood up of the reactor cavity and subsequent fuel movement. DCPP radiation control procedures define the scope, procedure, and frequency of these analyses.

12.2.4.3 Continuous Air Monitors

Portable CAMs may be used at selected locations as part of the airborne radioactivity surveillance program. Use of the CAMs is based on the potential for airborne radioactivity as a result of plant conditions or work activities.

12.2.5 OPERATING PROCEDURES

The grab air sampling program and the use of portable CAMs are described in DCPP procedures.

12.2.6 ESTIMATES OF INHALATION DOSES

The calculations of in-plant inhalation and immersion doses to plant operating and maintenance personnel are based on the estimated airborne concentrations for plant areas presented in Tables 12.2-5 through 12.2-15 and on the estimated occupancy factors for these areas presented in Table 12.2-17. The dose to plant personnel also depends on engineering controls to minimize airborne concentrations, on the type of respiratory protection equipment, if any, being worn, and on other administrative procedures such as purging of contaminated areas, limiting occupancy, etc. Note: These calculations are historical in nature and were completed prior to the 1994 new 10 CFR Part 20. At that time the concept of MPC based on a presumed chronic uptake and resultant body burdens over the years was dropped and replaced by the concept of the derived air concentration (DAC) based on annual dose limits and the assumption of acute rather than chronic exposures. Although prior to 1994 compliance was demonstrated by the number of MPC hours accumulated in a week, Table 12.2-18 reflects doses that are very conservatively calculated and far higher than what has historically been encountered during more than 2 decades of operation. These doses are still bounding and the MPC values will not be replaced with DACs.

The newer values and definitions are currently contained in 10 CFR Part 20 and included in plant procedures as appropriate.

Respiratory protective equipment may be used to limit dose from iodine, and particulates in accordance with 10 CFR Part 20 requirements. Tritium dose may be limited by either respiratory protection and protective suits to reduce the effective concentration below the 10 CFR Part 20 level, or by limiting personnel occupancy in areas of high concentration.

The estimated inhalation and immersion doses to plant personnel for normal full power operation are presented in Table 12.2-18 in units of person-rem/year.

It should be noted that the calculated doses to plant personnel in Table 12.2-18 are conservative estimates and, in view of the administrative controls over personnel dose due to the conservative assumptions used in the calculation of the source terms listed in Section 12.2.3, are much higher than would be expected under normal operating conditions. In particular, the assumptions for primary coolant leakage to the auxiliary building are extremely conservative, since continuous leakage of 20 gpd into the corridors and into three compartments simultaneously is assumed, giving a total leakage rate twice that of the anticipated operational occurrences case.

It is expected that personnel inhalation dose will be low and essentially negligible in comparison to external dose.

12.2.7 SAFETY EVALUATION

12.2.7.1 General Design Criterion 17, 1967 – Monitoring Radioactivity Releases

The containment ventilation systems, CRVS, ABVS, and FHBVS are provided with means for monitoring containment atmosphere, the facility effluent discharge paths, and the facility environs for the release of radioactivity as described in Sections 12.2.2.1 through 12.2.2.4 and 12.2.4.

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

The fuel and waste storage and handling areas are provided with monitoring and alarm systems for radioactivity, and the plant vents are monitored for radioactivity as described in Sections 12.2.2.2, 12.2.2.4, 12.2.4, and 11.5.2.6.

12.2.7.3 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The ABVS, FHBVS, and post-accident sampling compartment ventilation system control the release of airborne radioactive materials during normal operation and anticipated operational occurrences as described in Sections 12.2.2.3, 12.2.2.4, and 12.2.2.7.

12.2.7.4 10 CFR Part 20 – Standards for Protection Against Radiation

The containment ventilation systems, CRVS, ABVS, FHBVS, and turbine building ventilation systems control airborne radioactive materials during normal operation and anticipated operational occurrences such that doses to plant personnel are maintained ALARA and below the limits of 10 CFR Part 20 refer to Section 12.2.2.1 through 12.2.2.5.

In addition to the ventilation systems described above, the plant radiation shielding (refer to Section 12.1) supports ALARA principles as described in the RP Program (refer to Section 12.3).

12.2.7.5 10 CFR Part 50 Appendix I – Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents

The ABVS, FHBVS, and post-accident sampling compartment ventilation system control the release of airborne radioactive materials during normal operation and anticipated operational occurrences as described in Sections 12.2.2.3, 12.2.2.4, and 12.2.2.7.

The inhalation doses during normal operation at offsite locations are the result of releases of gaseous radioactive waste. These doses meet the criteria of 10 CFR Part 50, Appendix I as described in Section 11.3.

12.2.7.6 Regulatory Guide 8.8, July 1973 – Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable (Nuclear Reactors)

The ventilation systems control airborne contaminants to protect personnel during normal operations and maintenance activities and are designed for easy access and service in order to maintain doses ALARA (refer to Sections 12.2.1.4, 12.2.6, and 12.2.7).

12.3 RADIATION PROTECTION PROGRAM

This section describes the objectives, facilities and equipment, and dosimetry methods and procedures related to radiation protection of personnel at DCPP.

12.3.1 DESIGN BASES

12.3.1.1 10 CFR Part 19 – Notices, Instructions and Reports to Workers; Inspection and Investigations

DCPP has established requirements for notices, instructions and reports to individuals participating in U.S. Nuclear Regulatory Commission (NRC) licensed and regulated activities in accordance with 10 CFR Part 19.

12.3.1.2 10 CFR Part 20 – Standards for Protection Against Radiation

The RP Program supports the protection of personnel from radiation sources such that doses are maintained below the limits prescribed in 10 CFR Part 20 with noted exemptions.

Noted exemptions from the requirements of 10 CFR Part 20, as approved by the NRC, are:

- Exemption from Appendix A, Footnote d-2(c) allows the use of a radioiodine protection factor of 50 for Mine Safety Appliances GMR-I canisters.

- Authorization to: (1) use French-designed respiratory protection equipment that has not been tested and certified by the National Institute for Occupational Safety and Health; (2) not provide standby rescue persons whenever this equipment is used; and, (3) take credit for an assigned protection factor of 5,000 for this equipment.

12.3.1.3 Regulatory Guide 1.8, Revision 2, April 1987 – Qualification and Training of Personnel for Nuclear Power Plants

The Radiation Protection Manager (RPM) meets or exceeds the qualifications of Regulatory Guide 1.8, Revision 2 for RPM.

12.3.1.4 Regulatory Guide 8.8, July 1973 – Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable (Nuclear Reactors)

The RP Program supports the maintenance of occupational doses as low as practicable (i.e., ALARA).

12.3.2 FACILITIES AND EQUIPMENT

The principal RP facilities for the plant are discussed below.

(1) Access Control

Entrance and exit from the main RCAs of the plant are normally made through a central access control point on the 85 foot elevation. This area is used for administratively processing personnel in and out of the RCA, as well as providing a final contamination control point between the RCA and the rest of the plant. An auxiliary access control, located on the 140 foot elevation, may be utilized to provide more efficient access into the RCA, including containment buildings. Other access control stations may be temporarily established to support plant operations on an ad hoc basis.

The access controls on the 85 foot and 140 foot elevations include provisions for logging personnel in and out of the RCAs on radiation work permits. There is a portal monitor located at the exit of these access controls to serve as a final contamination monitor for personnel exiting the RCA. The 85 foot access control area has a decontamination facility that drains into the liquid radwaste system.

(2) Radiochemical Laboratory and Counting Room

These facilities are used for plant chemistry and radiochemistry programs as well as for processing samples for RP analyses. These facilities include detectors tied into a gamma spectroscopy system. Other counters and detectors are available and are used for gross alpha and beta counting and for tritium analyses.

(3) Calibration Facility

A calibration facility is provided for onsite calibration of most of the portable radiation monitoring instrumentation and some of the process monitors. The calibration facility is equipped with an irradiator for routine calibration of gamma-sensitive dose rate instruments. The irradiator is designed so that instruments can be accurately positioned for reproducible dose rates. The irradiator is traceable to the National Institute for Standards and Technology (NIST). Another irradiator is used for calibration of self-reading dosimeters. The irradiator is traceable to the NIST. Other irradiators, traceable to the NIST are also be used for calibration activities at DCPP. Calibration of instruments is performed using controlled vendor manuals or approved procedures. The RP Program also provides for instruments to be returned to the manufacturer or other appropriate contractors for calibration.

In addition to the sources located in the calibration facility, additional sources for calibration of some process radiation monitoring instruments are stored in the calibration facility or shielded safes near the chemistry laboratory.

(4) First Aid and Medical Facilities

The medical facility is staffed with trained emergency medical personnel. The medical facility serves as a general first aid area for minor injuries and an interim treatment area for seriously injured personnel until they can be transported to an offsite hospital or care facility. The medical facility has the capability of responding to injured persons who are also radiologically contaminated.

(5) Laboratory

A laboratory adjacent to the TSC may be used for counting in-plant samples if the normal counting room facilities become unusable following a postulated accident. The laboratory is equipped with a gamma spectroscopy system.

(6) Laundry Facility

An onsite laundry facility is provided for on-site cleaning and monitoring of protective clothing and respirators. The laundry facility is located above the solid radwaste storage facility.

The major categories of RP equipment are described below.

(1) Portable radiation survey instruments for alpha, beta, and gamma radiation detection and dose rate instruments for measuring beta, gamma, and neutron dose rates are described in Table 12.3-1. Some of the dose rate instruments are extended-range instruments to provide emergency monitoring capability.

(1) Air sampling equipment and CAMs are described in Table 12.3-2. This equipment is described further in Section 12.2.4.

(2) Respiratory protection equipment available for routine and emergency use is described in Table 12.3-3.

(3) Protective clothing is available for routine and emergency use.

(4) Several types of emergency, evacuation, and decontamination kits are available at the plant site and at key offsite locations. The contents of the

kits vary according to their intended use and include some or all of the following:

(a) Portable radiation monitoring instruments

(b) Air sampling equipment - some with batteries

(c) Environmental sampling and labeling equipment

(d) Protective clothing and respiratory protection equipment

(e) Portable radio communication equipment

(f) Decontamination supplies

(g) Procedures, maps, area drawings, etc.

12.3.3 PERSONNEL DOSIMETRY

The official and permanent record of accumulated external radiation dose received by individuals is obtained from interpretation of the TLDs. All individuals who are required to be monitored by 10 CFR Part 20 are issued beta-gamma TLDs and are required to wear them in the RCAs. TLDs are typically supplied and processed by a contractor. Dosimetry badges are changed on a routine basis, although the TLD of any individual may be processed at any time to determine the individual's dose status. Extremity or neutron dosimetry, as well as additional TLDs, are available and are issued as required.

Personnel working in the RCAs are provided with a means of estimating their accumulated external dose. Ordinarily, this is accomplished with the use of self-reading dosimeters. Dose estimates are updated daily, or more frequently when conditions warrant. These estimates are replaced by official dose records when the TLDs are analyzed. Information regarding an individual's dose is available so that personnel may keep themselves informed of their current dose status. Reports giving official personnel dose information are available to supervisors. These reports serve as a tool for the supervisor in making future job assignments. Individuals are closely monitored and may be restricted from further radiation work if their dose estimate reaches the administrative guideline, which is set below the dose limits established by 10 CFR Part 20.

The control of internal exposure to radioactive material is supplemented by a routine bioassay program consisting of whole body counting and passive monitoring using personnel contamination and portal monitors. Whole body counting is normally performed onsite. Urinalysis performed by an outside contractor may be used on a non-routine confirmatory basis as required. The frequency of sampling depends on the person's potential dose to airborne hazards.

Although engineering controls are normally used to control airborne radioactivity, use of respiratory protection equipment, control of access, limitation of exposure times, or other controls may be required to help maintain personnel exposure ALARA.

12.3.4 STORAGE OF RADIOACTIVE MATERIALS

In addition to the areas described in Section 12.3.2, which describes the principal radiation protection facilities for the plant, DCPP utilizes the following facilities and areas for the storage of radioactive materials:

12.3.4.1 Warehouse A Radioactive Material Storage Area

Warehouse A is located at the north end of the protected area outside the contiguous RCA. Inside the building, at the east end, is a caged area controlled by Radiation Protection and used to store radioactive material. Typical items stored in this area include, but are not limited to, outage related materials, materials associated with used fuel storage, and radioactive warehouse spare parts.

Materials stored in Warehouse A are contaminated with low leve3ls of radioactive material typically found in low level radioactive waste streams at DCPP. The radioactive content of the Warehouse A radioactive material storage area is managed such that the 10 CFR 20.1301, member of the public dose limits, are not challenged.

12.3.4.2 Area 10 Turbine Rotor Storage Building Radioactive Material Area

The turbine rotor storage building is located in Area 10, an area outside the protected area in the southern area of the owner controlled area. Inside the turbine rotor storage building, in the west end of the building, is a caged area controlled by Radiation Protection for the storage of radioactive material. Material stored in this area includes, but is not limited to, the spare reactor coolant pump rotor, and controlled warehouse items.

Materials in this area are contaminated with low levels of radioactive material typically found in low level radioactive waste streams at DCPP. The radioactive content of the Area 10 turbine rotor storage building radioactive material area is managed such that the 10 CFR 20.1301, member of the public dose limits, are not challenged.

12.3.4.3 Radioactive Material Storage Building

The radioactive material storage building is located along the southwest contiguous RCA inside the fence line. Normally, items stored in this area include, but are not limited to, contaminated and non-contaminated equipment used for online and outage maintenance.

Materials in this area are contaminated with low levels of radioactive material typically found in low level radioactive waste streams at DCPP. The radioactive content of the radioactive material storage building is managed such that the 10 CFR 20.1301, member of the public dose limits, are not challenged.

12.3.4.4 85’ Elevation Unit 2 Turbine Building Well Source Room

The 85’ elevation Unit 2 turbine building well source room is normally used for, but is not limited to, overflow storage of radiation protection instruments.

Items stored in the well source room may be contaminated with low levels of radioactive material typically found in low level radioactive waste streams at DCPP. The radioactive content of the 85’ elevation Unit 2 turbine building well source room is managed such that the 10 CFR 20.1301, member of the public dose limits, are not challenged.

12.3.4.5 Main Warehouse JD Room

The JD room is located on the southwest end of the first floor of the main warehouse. Typical items stored in this area include, but are not limited to, warehouse manufactured items containing small amounts of radioactivity. The JD room may also be used for the staging and holding area for incoming or outgoing radioactive material packages.

The radioactive content of the main warehouse JD room is managed such that the 10 CFR 20.1301, member of the public dose limits, are not challenged.

12.3.4.6 Main Warehouse Environmental Storage Area

The main warehouse environmental storage area is located on the northwest end of the first floor of the main warehouse. The main warehouse environmental storage area is typically used to store, but is not limited to, warehouse items containing small amounts of radioactivity which require controlled conditions for storage.

The radioactive content of the main warehouse environmental storage area is managed such that the 10 CFR 20.1301, member of the public dose limits, are not challenged.

12.3.4.7 RCA East Yard

Various containers are housed in the east yard which typically contain, but are not limited to, storage of radioactive material used for outages and online maintenance. A modular building in the north end of the RCA east yard is used by the Instrument and Control Group as a contaminated calibration facility.

Materials stored in the RCA east yard are contaminated with low levels of radioactive material typically found in low level waste streams at DCPP. The radioactive content of the RCA east yard is managed such that the 10 CFR 20.1301, member of the public dose limits, are not challenged.

12.3.5 SAFETY EVALUATION

12.3.5.1 10 CFR Part 19 – Notices, Instructions and Reports to Workers; Inspection and Investigations

The RP Program ensures the instructions provided to workers are commensurate with the potential radiological health problems present in the work place in accordance with the requirements of 10 CFR 19.12.

The RP Program maintains procedures that ensure routine reports to workers are provided in accordance with the requirements 10 CFR 19.13.

12.3.5.2 10 CFR Part 20 – Standards for Protection Against Radiation

The RP Program ensures that the radiation dose to personnel is ALARA in accordance with 10 CFR Part 20.

Program elements include:

- Instructions (refer to Sections 12.1.5 and 12.3.5.1)
- Dosimetry (refer to Section 12.3.3)
- Access control and protective equipment (refer to Sections 12.3.2 and 12.3.4)

The plant radiation shielding and ventilation systems, as described in Sections 12.1 and 12.2 respectively, support the ALARA principles.

In addition, the RP Program supports compliance with 40 CFR Part 190 as specified in 10 CFR 20.1301 (refer to Section 12.3.4).

12.3.5.3 Regulatory Guide 1.8, Revision 2, April 1987 – Qualification and Training of Personnel for Nuclear Power Plants

As a minimum, qualification requirements, including education, experience, and previous training for the RPM meet or exceed the qualifications of Regulatory Guide 1.8, Revision 2 in accordance with Technical Specification 5.3.1(a). Qualification requirements for other positions are described in Chapter 13.

12.3.5.4 Regulatory Guide 8.8, July 1973 – Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable (Nuclear Reactors)

The RP Program for the plant is carried out in accordance with PG&E's program directives. The program directives are statements of the policy covering each aspect of the RP Program and are based on appropriate NRC regulations. The program directives are implemented by various interdepartmental and department level administrative procedures and working level procedures contained in the Plant Manual.

The plant operating organization is described in Section 17.1.2 and illustrated in Figure 17.1-2. The RPM is responsible for administering, coordinating, planning, and scheduling all RP activities at the plant. The Chemistry and Environmental Operations Manager is responsible for administrating, coordinating, planning and scheduling all chemistry, radiochemistry, and environmental activities at the plant.

TABLE 12.1-1

PLANT ZONE CLASSIFICATIONS

Zone	Condition of Occupancy	Design Maximum Dose Rate, mrem/hr[a]
O	Unlimited access - areas that do not require controlled access for radiological reasons and can be occupied by plant personnel or visitors on an unlimited time basis	$\leq$ 0.5
I	Normal access - areas to which access is controlled for radiological reasons, but which require, or would permit, continuous occupancy by radiation workers during normal working hours	$\leq$ 1.0
II	Controlled access requiring periodic occupancy	$\leq$ 2.5
III	Controlled access requiring short-term occupancy	$\leq$ 15
IV	Controlled access requiring infrequent occupancy	> 15

(a) Basis: These are typical dose rates expected in the various zones based on full power operation of both Units with 1 percent failed fuel.

Figures 12.1-1 through 12.1-12
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