D.12 System and Transportation Safety

This section describes the environmental and regulatory setting, significance criteria, and an evaluation of system safety/risk of upset (i.e., accident) impacts associated with the proposed DCPP Steam Generator Replacement Project. The CEQA Guidelines recommend identifying hazards and risks to the public or environment caused by the project. However, the CEQA Guidelines do not provide any recommended significance criteria for radioactive hazards or risk of upset, and federal government control limits the ability of the CPUC to mitigate impacts in this area. It should be noted that the power plant is required to comply with NRC regulations for the possession, handling, storage, and transportation of radioactive materials in use at a nuclear power plant. These regulations address radioactive hazards, safety issues, and spent fuel handling and storage. The State of California, including the California Public Utilities Commission, and local jurisdictions such as San Luis Obispo County, are preempted from imposing any regulatory requirements concerning radiation hazards and nuclear safety on nuclear power plant operators. The operation of nuclear generating facilities and the possession, handling, storage, and transportation of radioactive materials are therefore precluded from State regulation. The Applicant's operating licenses require them to comply with all NRC regulations that apply to the operations and activities, including the replacement of key facility components and disposal of contaminated material. These regulations are described in Section D.12.2 of this EIR.

The CPUC and San Luis Obispo County have jurisdiction over limited activities that are not preempted by federal government control. This includes the permitting of the construction (e.g. preparation of the ground surface or grading, and minimum building standards) and transport activities for the Proposed Project (jurisdiction over possible activities offsite, such as for the Original Steam Generator Offsite Disposal Alternative, is discussed separately below). These activities are governed by the County's Local Coastal Program and other local ordinances. However, San Luis Obispo County does not have any jurisdiction over safety issues related to the proposed DCPP Steam Generator Replacement Project; these are solely under NRC jurisdiction. This System and Transportation Safety section is included in this EIR to provide the reader with an understanding of the safety issues associated with the Proposed Project. The CPUC and San Luis Obispo County cannot use the information contained in this section of the EIR to regulate or condition the safety issues of the Proposed Project given the NRC's sole jurisdiction over safety issues associated with the permitting, construction, and operation of the DCPP, including the replacement of steam generators. However, this information is included to provide full disclosure of potential environmental safety impacts associated with the Proposed Project.

D.12.1 Environmental Setting for the Proposed Project

Baseline safety and risk of upset conditions at the DCPP have been evaluated in several documents, including the Safety Analysis Report (SAR and updates) (PG&E, 1991, 1997, 2001) for the DCPP, the SAR for the Independent Spent Fuel Storage Installation (ISFSI) (PG&E, 2002a) and the Final EIR prepared for the ISFSI (SLO County, 2004). There are three major systems that contribute to baseline safety conditions at the DCPP, including:

- Units 1 and 2 reactors and systems,
- Spent Fuel Pools (SFP), and
- Independent Spent Fuel Storage Installation (ISFSI).

There are a wide variety of potential releases that could occur from the DCPP facilities that are summarized in the references listed above. In response to potential radiation releases, the DCPP has a comprehensive Emergency Response Plan (ERP) in place. The ERP identifies Emergency Planning Zones (EPZ) and Protective Action Zones (PAZ) for the region. The DCPP EPZ map is provided in Figure D.12-1, while the EPZ and PAZ descriptions are contained in Table D.12-1.

Table D.12-1. DCPP Protective Action Zones (PAZ)

Zone #	Zone Area	PROTECTIVE ACTION ZONES (PAZ)
PAZ 1	Within 2-Mile Radius from Plant	Identified residences, isolated hill areas within a two-mile radius from the plant.
PAZ 2	Within 6-Mile Radius from Plant	Identified residences, plant access road, upper segments of See Canyon and Prefumo Canyon Road, Montaña de Oro State Park, isolated hill areas extending out to a six-mile radius from the plant.
PAZ 3	Avila San Luis Bay See Canyon Squire Canyon	Avila Beach, Port San Luis, Pirate's Cove, San Luis Bay Estates, Avila Road, San Luis Bay Drive, See Canyon Road outside the six-mile limit, Squire and Gragg Canyons, Sunset Palisades extending to the east and south to about 9 or 10 miles from the plant
PAZ 4	Prefumo Canyon Los Osos Valley	Prefumo Canyon Road outside the six-mile limit, Los Osos Valley Road between Turri Road and Foothill Boulevard extending out to approximately 10 miles from the plant.
PAZ 5	Baywood Los Osos	Baywood Park, Los Osos, Turri Road, Los Osos Valley Road west of Turri Road, Clark Valley extending to the north approximately 10 miles from the plant.
PAZ 6	City of Pismo Beach	City of Pismo Beach, Shell Beach south of Spyglass Drive (including adjacent beaches). This area is more than 10 miles from the plant.
PAZ 7	Indian Knob Price Canyon	Price Canyon Road and isolated hill areas north of Pismo Beach. This area is more than 10 miles from the plant.
PAZ 8	San Luis Obispo Area	City of San Luis Obispo, Cal Poly, California Men's Colony, Camp San Luis Obispo, Cuesta College, O'Connor Way, Orcutt Road north of East Corral de Piedra Creek, Edna, Country Club, Crestmont Drive & Davenport Creek area. This area is more than 10 miles from the plant.
PAZ 9	Morro Bay Cayucos	Highway 1 west of Cuesta College, Morro Bay, Cayucos, Whale Rock Reservoir area. This area is more than 10 miles from the plant.
PAZ 10	Five Cities (Southern Portion)	City of Arroyo Grande, City of Grover Beach, Oceano, Halcyon and Pismo State Beach. This area is more than 10 miles from the plant.
PAZ 11	Orcutt Road Lopez Drive/Route 227	Canyon area north of Five Cities (bounded by Price Canyon, Orcutt Road, Huasna Creek and northern limits of Arroyo Grande and Pismo Beach). This area is more than 10 miles from the plant.
PAZ 12	Nipomo North of Willow Road	Nipomo Mesa north of Willow Road, Cienega Valley, Oceano Dunes State Vehicle Recreational Area. This area is more than 10 miles from the plant.
Zone #	Zone Area	PUBLIC EDUCATION ZONES (PEZ)
PEZ 13	Nipomo	Nipomo Mesa south of Willow Road, Nipomo Valley, Santa Maria Valley north of Santa Maria and Cuyama Rivers.
PEZ 14	Cuesta Pass Santa Margarita	US 101 north of San Luis Obispo, Santa Margarita, isolated hill areas north and east of San Luis Obispo within 20 miles of the plant.
PEZ 15	Route 41 Old Creek Road	Highway 1 north of Cayucos, Old Creek Road, Highway 41, isolated hill areas north and east of Cayucos and Morro Bay within 20 miles of the plant.

Source: San Luis Obispo County Office of Emergency Services, 2004.

Figure D.12-1. DCPP Emergency Planning Zone (EPZ) Map
CLICK HERE TO VIEW

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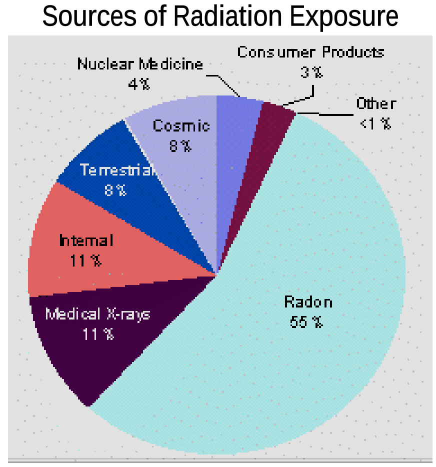
Radiation Terminology and Background

This section of the EIR contains many terms associated with standard measures of nuclear radiation that the average reader may find unfamiliar. In defining exposure levels in subsequent sections of the EIR, most exposures are references by a dose or equivalent dose. Therefore, the following definitions have been provided.

Dose	The absorbed dose, given in rads, that represents the energy in ergs or Joules absorbed from the radiation per unit mass of tissue. Furthermore, the biologically effective dose or dose equivalent, given in rem or Sieverts, is a measure of the biological damage to living tissue from radiation exposure.
Nuclide	Any species of atom that exists for a measurable length of time. A nuclide can be distinguished by its atomic mass, atomic number, and energy state.
Curie	The original unit used to describe the intensity of radioactivity in a sample of material. One curie equals thirty-seven billion disintegrations per second, or approximately the radioactivity of one gram of radium. This unit is no longer recognized as part of the International System of units. It has been replaced by the Becquerel.
Becquerel	An SI unit of radioactivity, defined as one disintegration per second. Replaces the Curie.
Sievert	A measure of dose (technically, dose equivalent) deposited in body tissue, averaged over the body. Such a dose would be caused by an exposure imparted by ionizing x or gamma radiation undergoing an energy loss of 1 joule per kilogram of body tissue (l gray). One Sievert is equivalent to 100 mrem or 0.1 rem.
Rem	An acronym for Roentgen Equivalent Man. A unit which measures radiation in terms of the energy involved (the same as RAD), weighted by a factor related to the type of radiation. For the types of radiation used in radiologic procedures this factor is equal to one, so the REM is equivalent to the RAD.
mrem	One millirem, or one thousandth of a rem.
RAD	An acronym for Radiation Absorbed Dose. A unit which measures radiation in terms of the absorbed dose. For radiological procedures it is equivalent to the rem, and the two units are used interchangeably. It represents the absorption of 100 ergs of nuclear (or ionizing) radiation per gram of absorbing material, such as body tissue.
erg	A metric unit of energy equal to work done by a force of 1 dyne (or 1 g-cm/s^2) acting over a distance of 1 cm. 10^7 (ten million) erg-s^{-1} (ergs per second) = 1 watt. Also, 1 Calorie = 4.2×10^{10} (42 billion) ergs.

In the United States, human exposure to potentially harmful radiation is commonly measured in units called millirem (one one-thousandth of a rem). On average, each individual receives about 360 millirem of radiation each year. About 300 millirem, or 82 percent of the total, is natural background radiation (from radon and other natural sources). The remaining 18 percent of our radiation exposure is from manmade sources as shown in Figure D.12-2 (National Safety Council, 2002).

Figure D.12-2.
Sources of Radiation Exposure



Source: National Safety Council, 2002.

Reactor Risk Baseline

Potential accidents associated with the operation of pressurized water reactor (PWR) nuclear power plants have been well documented. As noted above, PG&E maintains a SAR to address all potential off-normal (i.e., events that are not considered normal operating conditions but do not result in an accident) and accident (conditions that could lead to or result in a release of radioactive material) scenarios for the DCPP. Most off-normal and accident conditions would not result in any release of radioactive material, nor any impact to the surrounding environment.

The Reactor Safety Study (referred to as WASH-1400) was published in by the NRC in 1975. It was intended to estimate the probabilities of occurrences of accidents involving radioactivity release and to assess the risk of such accidents relative to other risks. The study involves (1) a list of potential accidents in nuclear reactors, (2) estimation of the likelihood of accidents resulting in radioactivity release, (3) estimation of health effects associated with each accident, and (4) comparison of nuclear accident risk with other accident risks. The findings of WASH-1400, which was published prior to the Three Mile Island accident, found that the risk of a nuclear accident was small and almost negligible compared with more common risks (e.g., automobile accident). Based on the number of U.S. reactors and the Three Mile Island accident, the calculated core damage frequency (CDF) is 1 in 2679 reactor-years or 3.7×10^{-4}/reactor-year (MIT, 2003), which translates to 1 in 1340 years or 7.5×10^{-4}/year for DCPP. Probabilistic Risk Assessment (PRA) experts estimate the CDF to be on the order of 1.0×10^{-4}/reactor-year (MIT, 2003) or an equivalent of 2.0×10^{-4}/year at the DCPP. In NUREG-0933[1], the NRC estimates the CDF for pressurized water reactors such as the type at the DCPP to be much lower at 1.236×10^{-5}/reactor-year (NRC, 2004a), which translates to 1 in 40,450 years or 2.472×10^{-5}/year for DCPP. This information consistently demonstrates that the probability of a core-damaging accident for a facility like DCPP is greater than ten in one million per year (1×10^{-5}/year), which is a widely accepted probability for defining significant risk.

In a study published by the U.S. House of Representatives in 1982, "Calculation of Reactor Accident Consequences (CRAC2) for U.S. Nuclear Power Plants (Health Effects and Costs) Conditional on an 'SST1' Release," potential offsite consequences of a large meltdown release from either of the DCPP reactors were evaluated. The study, based on 1982 population and dollars, found the potential for approximately 10,000 peak early fatalities, 11,000-12,000 peak early injuries, 12,000 peak cancer deaths and property damage on the order of $155-158 billion. While the CRAC2 modeling was considered conservative and for a highly unlikely worst-case release, it is clear that the consequences associated worst-case nuclear power plant accidents would be substantial.

At the present time, the combined frequency and potentially severe consequences of a core-damaging accident at the DCPP represent a substantial public safety risk as defined by generally accepted standards (as noted above, greater than ten in one million per year [1×10^{-5}/year]; with consequences including 10-12,000 people severely affected and $155-158 billion in property damage). While there are numerous events that can lead to an off-normal or accident condition at the DCPP reactors, the worst-case events would likely be associated with a Loss-of-Coolant Accident (LOCA) or reactor vessel failure (recent experience at the Davis-Besse reactor found a large cavity in the reactor vessel head that seriously jeopardized reactor vessel safety [MIT, 2003]). It should be noted that steam generator tube failures are a substantial contributor to overall facility risk and radioactive leak risk.

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1 NRC document "A Prioritization of Generic Safety Issues" (NRC, 2004a).

Spent Fuel Risk Baseline

In addition to reactor accidents, a high level of attention has been given to the issue of spent fuel handling and storage. Hazards associated with spent fuel pool fires and dry storage facilities have been evaluated by Sandia National Laboratory (1979), the NRC (2001), the Electric Power Research Institute (EPRI, 2002) and others (Alvarez et al., 2003). These studies were summarized in the Diablo Canyon Independent Spent Fuel Storage Installation Environmental Impact Report (SLO County, 2004).

The Proposed Project would occur at the same time as a regular refueling cycle for each unit, and all fuel would be removed from the reactors. These refueling events happen approximately every 18 to 21 months and last for 30 to 40 days (potentially as long as 75 to 80 days), during which the power plant temporarily ceases operations in order to replenish the enriched uranium needed as fuel to produce electricity. The DCPP nuclear reactors use enriched uranium oxide fuel pellets stacked end-to-end in 12-foot Zirconium alloy-based sealed rods to generate heat for conversion to electric power. These rods are arranged in groups known as fuel assemblies containing 264 fuel rods. Each reactor core has 193 fuel assemblies.

During normal refueling operations, spent fuel is transferred into water-filled pools (wet racks) to allow for further cooling and radiation shielding. Each reactor has a dedicated fuel handling system and spent fuel storage pool. During the Proposed Project, all fuel from the reactor would be placed in the spent fuel pools. Temporarily relocating the fuel is considered part of the baseline because this normally occurs during refueling outages.

The spent storage pools are located in the fuel handling building/auxiliary building, which is adjacent to the containment buildings. Spent fuel assemblies are normally removed from the reactor and transferred underwater to metal racks submerged in a pool of borated water. Diablo Canyon has two spent fuel pools. Each pool has a cooling system to maintain the pool temperature below 140 degrees F. The pool water cools the fuel rods and also serves as a shield against radiation. The pools are equipped with a redundant set of cooling pumps to serve as a backup in the event of loss of the main pumps.

When DCPP was originally built, each of the two spent fuel pools were designed to hold 270 fuel assemblies. The 270 fuel assemblies of each pool were sufficient for one normal refueling of 76 assemblies and a reserve capacity to receive one full core of 193 assemblies. At the time of the original design, it was expected that the spent fuel would be removed from the site for reprocessing for long-term storage at a federal site. Facilities for the commercial reprocessing of spent fuel or a long-term storage facility are currently not available.

In 1987, the Applicant received approval from the NRC to re-rack the spent fuel storage pools, increasing the density of spent fuel storage from 270 to 1,324 fuel assemblies per pool. As part of this application a number of abnormal and accidental conditions were addressed. These included:

- A dropped fuel assembly accident,
- Abnormal location of a spent fuel assembly, and
- Lateral rack movement.

The analysis contained in the application and the NRC licensing review for re-racking showed that none of these postulated events would result in exceeding the design reactivity factor for the pools. The reader is referred to the Reracking Application and associated NRC licensing record for additional information on these accidental events.

In its ISFSI SAR (PG&E, 2002a), PG&E has identified and evaluated several normal and operational deviations and off-normal handling conditions associated with storage cask handling and operations. Potential radiation exposures would be managed in a manner that maintains personnel radiation doses "as low as reasonably achievable" (ALARA). DCPP's ALARA program complies with the requirements of 10 CFR 20[2] and 10 CFR 50.[3] The ALARA program is implemented through NPG program directives, administrative procedures, and working level procedures. These documents will be revised by PG&E as needed to address ISFSI operations prior to operation of the ISFSI.

The Health Physics Program used for operating the Diablo Canyon ISFSI would implement the requirements of 10 CFR 20, 10 CFR 72,[4] and the NPG policy for implementation of the ALARA philosophy for all site activities involving potential radiation exposure. The Radiation Protection Manager is responsible for administering, coordinating, planning, and scheduling all radiation protection activities involving the ISFSI. The primary objective of the Health Physics Program is to maintain radiation exposures to workers, visitors, and the general public below regulatory limits and otherwise ALARA.

According to the ISFSI SAR (PG&E, 2002a) the Holtec HI-STORM 100 System, chosen for use at the Diablo Canyon ISFSI, has been designed with the principles of ALARA considered for the operation, inspection, maintenance, and repair of the cask system. PG&E provides the facilities, equipment, and the trained and qualified staff to ensure that any radiation exposures due to ISFSI operations are ALARA. The ISFSI storage pad will be monitored and evaluated on a routine basis to ensure that radiation exposures from the ISFSI storage pad to unrestricted areas are ALARA.

The ISFSI SAR also identifies radiation exposure levels for workers and the public for normal operations, with maximum worker exposure levels not to exceed 2 rem/year per worker, which is well below the allowable threshold of 5 rem. Under normal operating conditions, offsite exposure would also be well below applicable regulatory exposure limits.

"Off-normal" operations are events that deviate from normal operations, but are not considered accidents. Many of the off-normal events will likely occur over the life of the project, but with minimal consequences. Off-normal operations identified in the ISFSI SAR include:

- Off-normal pressures
- Off-normal environmental temperatures
- Confinement boundary leakage
- Partial blockage of air inlets
- Cask drop less than allowable height
- Loss of electric power
- Cask transporter off-normal operation

None of these events were considered to represent a threat to public safety. As with off-normal operations, PG&E has identified and thoroughly evaluated several accidental conditions associated with storage cask handling and operations. Many of these scenarios include "external events" that could occur which are beyond the control of PG&E, such as natural disasters. The accident scenarios postulated by PG&E include:

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2 Standards for Protection Against Radiation

3 Domestic Licensing of Production and Utilization Facilities [for nuclear facilities]

4 Licensing Requirements For The Independent Storage Of Spent Nuclear Fuel, High-Level Radioactive Waste, And Reactor-Related Greater Than Class C Waste.

- Earthquake
- Tornado
- Flood
- Drops and tip-over
- Fire
- Explosion
- Leakage through confinement boundary
- Electrical accident
- Loading of an unauthorized fuel assembly
- Extreme environmental temperature
- Hi-Trac transfer cask loss-of-neutron shielding
- Adiabatic heat-up
- Partial blockage of MPC vent holes
- 100% fuel rod rupture
- 100% blockage of air inlet ducts
- Transmission tower collapse
- Nonstructural failure of a CTF lift jack

The ISFSI SAR found that these ISFSI accident scenarios would not cause substantial public safety impacts (PG&E 2002a).

Low Level Radioactive Waste Baseline

Low level radioactive waste (LLW or LLRW) from nuclear power plants ranges from trash suspected of being slightly contaminated to highly radioactive material such as activated structural components found within or in close proximity to the reactor. LLW includes reactor components, tools, spent demineralizer resins, evaporator concentrates, used filters, and miscellaneous contaminated wastes such as rags, mops, paper, and protective clothing. Normal DCPP operations generate such LLW and are considered part of the baseline conditions.

The DCPP utilizes the Electric Power Research Institute (EPRI) Multi-Bed Test Unit and nuclide species chemical characterization to minimize LLW. In addition, all solid LLW at the DCPP is handled according to PG&E Procedure RP2 – Solid Low-Level Radioactive Waste Management. The LLW that DCPP generates is disposed at the Envirocare of Utah facility in Clive, Utah.

The following discussion from the NRC (NRC, 1996) summarizes the conditions for LLW onsite storage, which is proposed by PG&E as part of the Proposed Project (see Section 12.3.4, Original Steam Generator Removal, Transport, and Storage):

> *LLW is normally stored on site on an interim basis before being shipped off site for permanent disposal. Onsite storage facilities are designed to minimize personnel exposures. High-dose-rate LLW is isolated in a shielded storage area and is easily retrievable. The lower-dose-rate LLW is stacked or stored to maximize packing efficiencies. NRC requirements and guidelines ensure that LLW is stored in facilities that are designed and operated properly and that public health and safety and the environment are adequately protected (EPRI NP-7386). NRC requirements and guidelines include the following:*

- *The amount of material allowed in a storage facility and the shielding used should be controlled by dose rate criteria for both the site boundary and any adjacent offsite areas. Direct radiation and effluent limits are restricted by 10 CFR Part 20 and 40 CFR Part 190. The exposure limits given in 10 CFR 20.1301 apply to unrestricted areas.*
- *Containers and their waste forms should be compatible to prevent significant corrosion within the container. After a period of storage, the subsequent transportation and disposal should not cause a container breach.*
- *Gases generated from organic materials in waste packages should be evaluated periodically with respect to container breech. After a period of storage, the subsequent transportation and disposal should not cause a container breach.*
- *Gases generated from organic materials in waste packages should be evaluated periodically with respect to container breech. High-activity resins should not be stored more than 1 year unless they are in containers with special vents.*
- *A program of at least quarterly visual inspection should be established.*
- *A liquid drainage collection and monitoring system should be in place. Routing of the drain should be to a radwaste processing system (EPRI NP-7386).*

NRC has historically discouraged the use of onsite storage as a substitute for permanent disposal. NRC Generic Letter 81-38 (NRC 1981) states that no facility should be built to store waste for longer than 5 years under a licensee's 10 CFR 50.59 evaluation. Specific NRC approval should be obtained. This limitation was based in part on safety considerations but was aimed at encouraging the development of permanent LLW disposal facilities. However, recognizing that the 5-year limit has not influenced the development of new waste disposal facilities and that the states continue to make slow progress, NRC has eliminated in its guidance any language that the 5-year term is a limit beyond which storage would not be allowed.

Regarding nuclear power reactors, the 5-year limit is associated with the need to obtain a separate Part 30 license to store LLW. Generic Letter 81-38 states that under certain conditions, Part 50 licensees should obtain a Part 30 materials license to store LLW. These conditions are that (1) there exists an unreviewed safety question with the proposed storage facility, (2) the existing license conditions or technical specifications prohibit increased storage, or (3) the planned storage time exceeds 5 years. Other than for the conditions noted, NRC regulations and procedures do not call for a separate Part 30 license for power reactors for LLW storage, because power reactor licensees are already authorized under Part 30 to possess by-product materials produced by the operation of the facility within the limits of their operating license.

Generic Letter 81-38 states that the application for a Part 30 license is for the administrative convenience of the Commission and is not intended to be substantively different from an application for amendment of the facility operating license (i.e., the Part 50 license). Because Part 50 licensees are already authorized under Part 30 to possess their LLW, NRC staff revised the guidance to state that these licensees should amend their Part 50 licenses when the storage of LLW is not within the limits of their current operating license. On February 1, 1994, the Commission, in responding to SECY-93-323, which recommended withdrawal of the onsite storage rulemaking, directed the staff to eliminate the requirement for power reactor licensees to obtain a separate Part 30 license (SECY-94-198). Agreement states are currently reviewing proposed changes to existing guidance.

> *Several events have increased the trend towards longer onsite storage. These events include the closure of the Beatty, Nevada, site in 1992; the restriction of the Richland, Washington, facility to Northwest Compact and Rocky Mountain Compact states and the restriction of the Barnwell, South Carolina, site to waste generated by Southeast Compact states. As of July 1994, 33 states were without access to licensed full-service disposal facilities. The status of state efforts to form compacts and identify new disposal sites is discussed in Section 6.4.3.3 [NUREG-1437 Vol. 1]. However, as of July 1, 1995, all states except North Carolina have access to the Barnwell site. The Envirocare site in Utah takes limited types of waste from certain generators.*

The NRC (1996) also notes that there will likely be additional quantities of LLW generated by the nuclear industry due to "extended power plant operations under renewed licenses" and the need for steam generator replacements in PWRs.

The NRC has entered into agreements with 33 states (including California), called Agreement States, to allow these states to regulate the management, storage and disposal of certain nuclear wastes outside of nuclear power plants. The NRC relinquishes to the Agreement States portions of its regulatory authority to license and regulate byproduct materials (radioisotopes); source materials (uranium and thorium); and certain quantities of special nuclear materials. Federal law, however, does not permit the NRC to delegate its responsibility for regulating nuclear power plants to Agreement States (Ohio State University, 2001). Nuclear power reactors in the United States must be licensed by the NRC and must comply with NRC regulations and conditions specified in the license in order to operate.

The Low-level Radioactive Waste Policy Amendments Act of 1985 gave the states responsibility for the disposal of their LLW. The Act encouraged the states to enter into compacts that would allow them to dispose of waste at a common disposal facility. Most states have entered into compacts; however, due to numerous political and technical issues, no new disposal facilities have been built since the Act was passed (NRC, 2005).

California became an Agreement State in 1962 and is a member of the Southwest Compact (including Arizona, North Dakota and South Dakota). The California Agreement State program is located in the Department of Health Services, Radiologic Health Branch. The Radiologic Health Branch administers the radioactive materials program. The California program regulates approximately 2,182 specific licenses authorizing radioactive materials (NRC, 2004b). The State of California does not have authority to regulate the management, storage, or disposal of LLW at nuclear power plants (California Department of Health Services, Radiologic Health Branch, 2005).

The Southwest Compact does not have a licensed disposal site, resulting in the need for facilities to utilize one of three nationwide disposal sites or store LLW onsite. California was originally slated to host the Southwest Compact disposal facility at the Ward Valley Low Level Radioactive Waste facility. However, potential contamination groundwater migration at the Ward Valley LLW site were never satisfactorily resolved and the facility was never permitted or constructed.

The three U.S. commercial land disposal facilities accept waste only from certain states or accept only limited types of LLW as discussed below. The remainder of the LLW is stored primarily at the site where it was produced, such as at nuclear power plants, hospitals, research facilities, and clinics (NRC, 2002b). Each of the three operating LLW disposal sites is located in and regulated by an Agreement State (i.e., South Carolina, Washington, and Utah). Each state has an oversight program that consists of periodic inspections of the facilities. The inspectors examine whether incoming shipments are properly documented and analyzed for their radioactive material content. They also ensure that licensees properly implement the radiation safety and waste disposal requirements.

The Barnwell, South Carolina disposal site is licensed by the State of South Carolina to receive wastes in Classes A through C.[5] The site accepts waste from all U.S. generators except those in Colorado, Nevada, and New Mexico (Rocky Mountain Compact states) and Idaho, Montana, Oregon, Utah, Washington, and Wyoming (Northwest Compact states). Beginning in 2008, Barnwell will only accept waste from Connecticut, New Jersey, and South Carolina (Atlantic Compact states) (NRC, 2005).

The Hanford, Washington disposal site is licensed by the State of Washington to receive wastes in Classes A through C. The site accepts waste only from generators located in Colorado, Nevada, and New Mexico (Rocky Mountain Compact) and Idaho, Montana, Oregon, Utah, Washington, and Wyoming (Northwest Compact) (NRC, 2005).

The Envirocare disposal site located in Clive, Utah is licensed by the State of Utah for Class A waste only. This site accepts waste from all regions of the U.S. (NRC, 2005). If no new LLW disposal sites are licensed, after 2008, California-generated LLW will only be accepted by the Envirocare site. California's inability to construct and operate a LLW disposal site will result in there being no place to ship commercial Class B and C waste by 2008, and Envirocare of Utah will have a monopoly on disposal of Class A waste.

The steam generators replaced under the Proposed Project would be classified as a Class A waste. Many other nuclear power plant facilities undergoing similar steam generator replacements have opted to store steam generators onsite, including: Palo Verde in Arizona, Oconee in South Carolina, Calvert Cliffs in Maryland, and Sequoyah in Tennessee. Facilities will continue to store OSGs onsite until alternative disposal sites become available that allow for the cost-effective disposal of LLRW.

Facility Security and Terrorism Issues

Subsequent to the events of September 11th, 2001 the NRC and commercial nuclear industry have implemented a variety of measures aimed at reducing the likelihood of a successful terrorist attack on nuclear facilities. Within hours of the September 11th attacks, the NRC issued a series of classified, security-related advisories to power reactor licensees which were above and beyond current regulatory requirements. These security enhancements were later formalized in an order issued on February 25, 2002. The order imposed Interim Compensatory Measures (ICMs), which PG&E implemented by August 31, 2002. Later, in April 2003, NRC issued a new design basis threat (DBT), which establishes the maximum terrorist threat that a facility must defend against, and required plants to develop and implement new security plans to address the new threat by October 2004. The NRC DBT and DCPP security plans are not publicly available and are not discussed in this EIR.

The DCPP has been designed and constructed to withstand potential hazards associated with natural external events, such as earthquakes, tornadoes, floods, and hurricanes. Terrorist attacks by fire or explosion would be analogous to external natural events and their implications for damage and release of radioactivity. The reactors are protected by a robust containment structure, which is typically protected by about four feet of reinforced concrete with a thick steel liner, and the reactor vessel, which is made of steel that is about six inches thick. Based on a peer-reviewed study prepared by EPRI, areas of the plant that house the reactor would withstand the impact of a widebody commercial aircraft (EPRI, 2002) and containment would not be breached. Therefore, it is unlikely that a terrorist attack on a nuclear reactor would result in a large-scale radioactivity release. However, nuclear power plant spent

[5] Based on the requirements of 10 CFR 61, LLW is classified as A, B, C or GTCC (greater than Class C) according to the half-lives and concentrations of key radionuclides. In general, requirements detailing waste form, stability, and disposal methods become more stringent when going from class A to GTCC.

fuel facilities are not afforded the same level of protection as the reactors and could pose a potential risk of radioactivity releases in the event of a terrorist attack.

The Final EIR for the DCPP ISFSI identified numerous accidental release scenarios that could occur in the baseline conditions, including those resulting from terrorism damage to the ISFSI storage casks. Potential terrorist-related impacts that were evaluated included a willful aircraft strike on the ISFSI and an attack utilizing an anti-armor missile. These impacts were considered by San Luis Obispo County to be less than significant following the implementation of the proposed mitigation measures. However, the proposed mitigation measures were only recommendations, and it is unclear if any of the measures were adopted by the NRC or implemented by PG&E. Therefore, it is likely that potential terrorist-related impacts on the ISFSI identified by San Luis Obispo County remain as a substantial baseline risk to public safety.

Potential terrorist threats to the DCPP spent fuel pools were also identified in the ISFSI FEIR. It has been known for some time that loss of water in densely-packed spent fuel pools could result in spent fuel heating up relatively quickly to temperatures where the fuel cladding could catch fire resulting in the release of nuclear material to the environment (Sandia National Laboratories 1979). The NRC has estimated the probability of this type of event to be less than one chance in 100,000 per pool-year (NRC, 2001), which is below the level at which the NRC would require mitigation. However, a terrorist act, such as a crash of a large jet into the pool, could occur. Draining the pools via an existing pipe or valve at DCPP is an extremely remote chance because the lowest pipe elevation is 6 feet above the spent fuel assemblies. In addition, the pipes are equipped with anti-siphon holes that would prevent siphoning of the pool inventory due to a leak at a lower elevation elsewhere in the system (SLO County, 2004). Although it is unlikely that a jet could be crashed into the spent fuel pools given their location, directly behind the main containment structures, in the event of such an incident, the pools and or spent fuel assemblies could be damaged, and a fire could occur from any spilled aviation fuel. Such a scenario could lead to the loss of cooling water flow. It could also make it difficult for any emergency response, particularly if any of the spent fuel assemblies were breached in the attack (SLO County, 2004).

With loss of cooling water flow, the temperature of the water in the spent fuel pools would increase to the point where it could begin to boil off. The time it would take for the water in the spent fuels to boil down enough to expose the spent fuel road assemblies would depend on how much recently discharged spent fuel was in the pool. Boil down could take up to 10 days if the most recent fuel discharge was more than a year old (Alvarez et al., 2003). Given this amount of time and the fact DCPP has redundant cooling water pumps this scenario is highly unlikely to occur. Even if both sets of cooling pumps were lost, the time available would be great enough to repair or replace the required equipment to reestablish the cooling supply.

A recent Electric Power Research Institute (EPRI) study found that a Boeing 767-400 jet traveling at 350 miles per hour would not penetrate the wall of a spent fuel pool. The study found that the concrete around the pool would crack, but that the stainless steel walls would not be breached (EPRI, 2002). The detailed results of this study have not been released to the public for security reasons. As such, no independent verification of the analysis was possible.

Another concern with an aircraft impact would be fire. While the impact of the aircraft may not breach the spent fuel pool, it could result in the collapse of the building and a resultant fire from the jet fuel. The heat from the fire could be enough to evaporate some of the water in the pool. The crash of an aircraft similar to those used on September 11th could provide enough heat from the burning fuel to vaporize

500 tons of water (Alvarez et al., 2003). For DCPP, the location of the spent fuel pools and the fact that the pools are underground make it very unlikely that a large commercial aircraft could be flown into the pools.

A number of studies have been done to estimate the impacts associated with a spent fuel pool fire. Depending on the amount of nuclear material released, the areas impacted could be as high as 17 million acres (69,000 square kilometers) (Alvarez et al., 2003).

An NRC study estimated that upwards of 1.8 million acres (7,000 square kilometers) could be impacted as a result of a pool fire that released nuclear material (NRC, 1997).

It should be noted that the NRC and Nuclear Energy Institute (NEI) have serious concerns related to the Alvarez article cited in the ISFSI EIR (see SLO County, 2004 for the NEI comments and EIR responses). The Alvarez article contains many assumptions, some of which would be considered conservative, but also provides a relevant overview of the benefits of dry storage of spent fuel. The EIR authors also recognize that one of the contributing authors to the Alvarez article was employed by a transport and dry storage equipment vendor while others are actively associated with groups that have viewpoints that are at odds with the nuclear industry. The NEI listed four areas of concern based on the NRC review of the article including:

1. Provides no justification for the postulated probabilities of worst-case spent fuel pool damage;
2. Overestimates radiation release;
3. Overestimates consequences and societal costs for the published severe event; and
4. Underestimates the costs of the author's main remediation.

While the NEI made many valid points in its comments and the overall risk associated with a spent fuel pool fire, the overall baseline risk is still considered substantial. To address this risk, the ISFSI EIR recommended accelerating the transfer of spent fuel to the dry storage casks from the spent fuel pools.

D.12.2 Applicable Regulations, Plans, and Standards

Federal, State, and local agencies have established standards and regulations that affect the Proposed Project. A summary of the regulatory setting for system and transportation safety is provided below.

Federal and State Standards

The NRC and federal Department of Transportation (DOT) regulate the use and transport of nuclear materials and protection of public safety. The roles of these agencies for the storage and transport of low level radioactive waste are:

> NRC – Regulates users of radioactive material and the design, construction, use, and maintenance of onsite storage facilities and shipping containers used for larger quantities of radioactive material and fissile material (such as uranium). NRC regulations for transport are located in the Code of Federal Regulations, Title 10, "Energy," Part 71, "Packaging and Transportation of Radioactive Material." NUREV-0810 regulates the onsite storage of LLW. The NRC would provide oversight of all activities associated with the onsite storage of LLW, such as the OSGs.

DOT – Regulates shippers and carriers of radioactive material and the conditions of transport (including routing, tiedowns, radiological controls, vehicle requirements, hazard communication, handling, storage, emergency response information, and employee training). DOT regulations are located in the Code of Federal Regulations, Title 49, "Transportation." The DOT would regulate all activities associated with the offsite transportation of the OSGs, at such time that the OSGs are removed from the DCPP site for disposal.

The following federal regulations apply to other operations at DCPP:

10 CFR – Energy	
Part 19	Notices, Instructions and Reports to Workers
Part 20	Standards for Protection Against Radiation
Part 61	Licensing Requirements for Land Disposal of Radioactive Waste
Part 71	Packaging and Transportation of Radioactive Material

49 CFR – Transportation (Subchapter C – Hazardous Materials Regulation)	
Part 171	General Information, Regulations and Definitions
Part 172	Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information and Training Requirements
Part 173	Shippers – General Requirements for Shipments and Packaging
Part 174	Carriage by Rail
Part 176	Carriage by Vessel
Part 177	Carriage by Public Highway
Part 178	Specifications for Packaging
Part 180	Continuing Qualifications and Maintenance of Packaging
Part 390	Federal Motor Carrier Safety Regulations
Part 391	Qualifications for Drivers
Part 392	Driving of Commercial Motor Vehicles
Part 393	Parts and Accessories Necessary for Safe Operations
Part 395	Hours of Service of Drivers
Part 396	Inspection, Repair and Maintenance
Part 397	Transportation of Hazardous Materials; Driving and Parking Rules

The NRC NUREG-series publications establish requirements that are also applicable to the Proposed Project, including:

- NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. Potential safety issues associated with the project, such as OSG removal and storage, RSG installation and all staging activities would be subject to NUREG-0800 review.
- NUREG-0810, Design Guidance for Temporary Onsite Appendix 11.4-A Storage of Low Level Radioactive Waste. The design of the OSG Storage Facility would be subject to NUREG-0810 review.
- NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants. Although PG&E has not applied for a license renewal for the DCPP units, the Environmental Impact Statement in NUREG-1437 provides information on the NRC process of environmental evaluation for steam generator replacement activities related to license renewal.

The Proposed Project would involve storage of the OSGs onsite until the time of decommissioning. At decommissioning, OSG disposal must meet the waste disposal requirements of 10 CFR 61 as follows. NRC regulations in 10 CFR 61.55 divide low level radioactive waste into classes A, B and C. These classifications are based on the concentration of radionuclides, particularly long-lived radionuclides. Class A waste has the lowest concentration, and Class C has the highest. Class C waste must meet rigorous requirements on waste form to ensure stability, and requires additional measures at the disposal facility to protect against inadvertent intrusion. Waste that exceeds the requirements for Class C is considered Greater Than Class C (GTCC) and is generally considered unsuitable for near surface land disposal. The OSGs would be classified as a Class A waste. The 10 CFR 61 performance objectives address long-term safety of radioactive waste storage. The performance objectives addressed by 10 CFR 61 for land disposal of LLW are:

1. Long-term protection of the public health and safety (and the environment);
2. Protection of an inadvertent intruder;
3. Protection of workers and the public during operation of a LLW disposal facility; and
4. Long-term stability of the disposal site after closure.

Offsite transport of the Class A OSGs, an alternative of the Proposed Project, would be required to meet the requirements 10 CFR 71, radiation level limits for radioactive waste transportation. Those limits are:

i. 200 mrem/hour or less at the waste container surface.

ii. 200 mrem/hour at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load, and on the lower external surface of the load.

iii. 10 mrem/hour or less at any point two meters from the vertical planes projected from the outer edges of the vehicle.

iv. 2 mrem/hour in normally occupied spaces.

Under 10 CFR 20, DCPP would be required to update the existing Radiation Protection Program to include issues associated with the Proposed Project. Specifically, under 10 CFR 20.1101, PG&E would be subject to the following requirements:

a) Each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of this part. (See Section 20.2102 for recordkeeping requirements relating to these programs.)

b) The licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

c) The licensee shall periodically (at least annually) review the radiation protection program content and implementation.

d) To implement the ALARA requirements of Section 20.1101 (b), and notwithstanding the requirements in Section 20.1301 of this part, a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established by licensees other than those subject to 10 CFR 50.34a, such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem

(0.1 mSv) per year from these emissions. If a licensee subject to this requirement exceeds this dose constraint, the licensee shall report the exceedance as provided in Section 20.2203 and promptly take appropriate corrective action to ensure against recurrence.

As noted above, the NRC has sole jurisdiction over the regulation of nuclear power plant operations, including radioactive hazards, safety issues, and radioactive waste handing and storage. The State of California (and its local jurisdictions) is preempted from imposing any regulatory requirements concerning radiation hazards and nuclear safety on nuclear power plant operators. The possession, handling, storage, and transportation of radioactive materials similarly are precluded from State regulation.

Local Ordinances and Policies

The CPUC and San Luis Obispo County have jurisdiction over limited activities that are not pre-empted by federal government control. This includes the permitting of the construction and transport activities for the Proposed Project (jurisdiction over possible activities offsite, such as for the Original Steam Generator Offsite Disposal Alternative, is discussed separately above). These activities are governed by the County's Local Coastal Program. However, San Luis Obispo County does not have any jurisdiction over safety issues related to the proposed DCPP Steam Generator Replacement Project.

D.12.3 Environmental Impacts and Mitigation Measures for the Proposed Project

D.12.3.1 Definition and Use of Significance Criteria

The scope of this analysis is to prepare a qualitative evaluation of Proposed Project system and transportation safety and risk of upset. However, some quantification, although well short of a quantitative risk analysis (QRA), is used to determine the significance of potential impacts.

The CEQA Guidelines recommend identifying hazards and risks to the public or environment caused by the project. However, the CEQA Guidelines do not provide any recommended significance criteria for radioactive hazards or risk of upset, and federal government control limits the ability of the CPUC to mitigate impacts in this area. Criteria from federal guidance are selected in order to facilitate full disclosure of potential impacts. NRC guidance offers many screening level approaches where either the probability of an event or exposure levels define the acceptability of an event.

Significance of Accident Probability. In order to evaluate the significance of potential impacts, it is necessary to consider the complete risk picture which includes the consequences of an accidental release, as well as the probability that an event can occur. In many cases, events with catastrophic consequences are considered insignificant because the probability that such an event can occur is so unlikely, or is preventable by many layers of protection, that it is highly speculative to assume that the event can ever occur.

Substantial guidance on acceptable accident probability is provided in the NUREG guidelines. Generally, when screening potential accident scenarios an event with a probability of less than one in one million per year (1×10^{-6}/year) is considered less than significant. Accident scenarios with a probability of less than ten in one million per year (1×10^{-5}/year) are considered acceptable and require no further mitigation.

Significance of Consequences. In evaluating potential impacts from this project, the radiation exposure that would be considered significant is set at a low level that would essentially represent a "no adverse effect" level. An adverse consequence under an accident scenario is defined in 10 CFR 72.106:

> *Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent may not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or any extremity may not exceed 0.5 Sv (50 rem).*

Acceptable exposure levels defined in 10 CFR 72.104 for normal and anticipated off-normal operations are substantially lower than those defined in 10 CFR 72.106 as follows:

> *During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid and 0.25 mSv (25 mrem) to any other critical organ as a result of exposure to:*
>
> 1. *Planned discharges of radioactive materials, radon and its decay products excepted, to the general environment,*
>
> 2. *Direct radiation from ISFSI or monitored retrievable storage (MRS) operations, and*
>
>   a. *Any other radiation from uranium fuel cycle operations within the region.*
>
>   b. *Operational restrictions must be established to meet as low as is reasonably achievable objectives for radioactive materials in effluents and direct radiation levels associated with ISFSI or MRS operations.*
>
>   c. *Operational limits must be established for radioactive materials in effluents and direct radiation levels associated with ISFSI or MRS operations to meet the limits given in paragraph (a) of this section.*

While the proposed OSG Storage Facility would store low level waste and would not be directly regulated by 10 CFR 72.106, the exposure limits identified in this regulation can be used to characterize the maximum acceptable exposure during an accident.

The acceptable exposure amounts associated with radioactive materials are principally established by the NRC and the USEPA. Therefore, the provisions of USEPA radiation exposure limits under 40 CFR 190 apply to all DCPP operations, including Proposed Project activities and the proposed OSG Storage Facility, since the NRC is responsible for the implementation of this standard for licensed power reactors. Under Section 190.10(a), the annual dose equivalent must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations.

In addition, the NRC also requires that doses be kept “as low as reasonably achievable” (ALARA). The limits are set as maxima which must not be exceeded, and the goal is to keep doses as far below these limits as practical. The NRC further defines acceptable exposures levels for various worker exposure scenarios and the public as shown in Table D.12-2.

Table D.12-2. Dose Limits per Year

Radiation Workers:	Dose
Total Effective Dose Equivalent (TEDE)	5 rem
Dose Equivalent to the Eye	15 rem
Shallow Dose Equivalent to skin, extremities	50 rem
TEDE to any other individual organ	50 rem
TEDE to embryo/fetus of declared pregnant woman	0.5 rem
Minors (aged under 18)	10 percent of worker limit

Source: 10 CFR 20.1201 and 20.1301.

The USEPA has also established an individual radiation protection limit from Department of Energy (DOE) facilities. The total radiation dose limit for individual members of the public as defined by 10 CFR 20.1301 is 1 mSv/year (100 mrem/year), not including the dose contribution from background radiation. Limits on emissions of radionuclides to the air from DOE facilities are set such that they will not result in a dose greater than 0.1 mSv/year (10 mrem/year) to any member of the public (40 CFR 61.92).

In the event of a major accident, 10 CFR 100.11 (determination of exclusion area, low population zone, and population center distance) establishes exposure guidelines as a whole body dose of 25 rem which corresponds numerically to a once in a lifetime accidental or emergency dose for radiation workers. However, it is not intended to imply that 25 rem constitutes acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value has been set forth as a reference value, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation. This exposure level is typically used to define zones of emergency response and exposure limits for emergency responders.

In evaluating the significance of potential exposures to the public, the more stringent applicable exposure levels as defined in 10 CFR 72.104, 40 CFR 190, 10 CFR 20.1301, 40 CFR 61.92, and 10 CFR 20.1101 (radiation protection programs), in combination with NUREG probability thresholds, were used to estimate significant exposures and potential impacts.

D.12.3.2 Replacement Steam Generator Transport

Transport of the RSGs would not pose any radiological hazard since they would be newly manufactured (i.e., not used and exposed to radiation) The proposed delivery route would generally avoid interaction with the public for most of the route, remaining in the shipping lanes far offshore until the approach to Port San Luis. As the RSG barge approaches and enters Port San Luis, potential conflicts with existing marine traffic could pose a safety hazard. In addition, transport of the RSGs between Port San Luis and the DCPP site could compromise emergency response capabilities. See Section D.10 for further details.

Impact S-1: RSG barges would create a navigational hazard in Port San Luis

Barges would deliver the RSGs through Port San Luis following the fairway markers to Harford Pier and a location just northeast of Harford Landing (see Figure D.12-3). Delivery would likely occur during high tide, and offloading would occur to a parking lot approximately 1,000 feet from the DCPP Access Gate. The size of the barges would require the temporary relocation of some moored vessels to safely accommodate the barge. It is also likely that measures taken to assure the safe transport and offloading of the RSGs would disrupt port operations. However, with implementation of Mitigation Measure S-1a, potential impacts can be mitigated to a level that is considered less than significant (Class II).

Mitigation Measures for Impact S-1, RSG barges would create a navigational hazard in Port San Luis

S-1a **Barge Navigational Safety Plan.** The Applicant shall develop a barge navigational safety plan to minimize the impact on existing Port operations. The plan shall be submitted to, and approved by the Port San Luis Harbor District and CPUC. At a minimum the plan shall include the following elements:

- Identify moored vessels that will need to be temporarily relocated, and provide necessary temporary mooring facilities or funding for the Port District to accommodate the temporary relocation of moored vessels.
- Identify activities such as home fleet and dry dock operations in the Port that may conflict with barge transport and/or offloading. Working with the Harbor District, identify procedures that will minimize conflicts with existing operations.
- Working with the Harbor District, identify additional navigational aids and security that will be necessary to safely move the barges through the Port. The Applicant can provide the additional navigational aids and/or security, or contract with the Harbor District to provide the necessary services.

Impact S-2: RSG transport between Port San Luis and the DCPP could impede emergency response vehicles

During RSG delivery, the road between Avila Beach Drive and the DCPP site would be temporarily blocked. This would limit the ability of emergency service personnel to respond to incidents at DCPP. The possible disruption of fire protection or other emergency services (Impact U-2, as described in Section D.10, Public Services and Utilities) could lead to adverse public safety impacts. However, potential impacts can be mitigated to a level that is considered less than significant with Mitigation Measure U-2a (Class II).

Mitigation Measure for Impact S-2, RSG transport between Port San Luis and the DCPP could impede emergency response vehicles

Implement Mitigation Measure U-2a.

D.12.3.3 Replacement Steam Generator Staging and Preparation

All Proposed Project staging and preparation activities would occur well away from areas with public access. No radiological hazard would occur because staging and preparation activities would not involve handling nuclear fuel or radioactive waste. Therefore, these activities would not pose any appreciable safety hazard to the public. No impacts would occur.

Figure D.12-3. Barge Transport in Port San Luis

CLICK HERE TO VIEW

D.12.3.4 Original Steam Generator Removal, Transport, and Storage

OSG removal, transport, and storage activities would result in worker and public exposure to residual OSG radiation,[6] and onsite storage of the OSG would introduce new hazards related to the proposed OSG Storage Facility. The impacts related to these hazards are discussed below.

Equipment used during OSG transport could obstruct access for emergency response vehicles, which could lead to adverse public safety impacts (see discussion under Impact S-2 and Impact U-2 in Section D.10). OSG transport from the DCPP reactors to the OSG Storage Facility would temporarily block roads within the DCPP site, thus limiting the ability of emergency personnel to respond to incidents. However, potential impacts could be mitigated to a level that is considered less than significant with Mitigation Measure U-2a (Class II).

Impact S-3: Residual contamination would be present on the OSGs with the potential for radiation exposure during removal and transport

Potential radiation exposures would be managed in a manner that maintains personnel radiation doses in accordance with the existing DCPP Radiation Protection Program. DCPP's ALARA program complies with the requirements of 10 CFR 20 (Radiation Protection Program, see section D.12.2) and 10 CFR 50 (Domestic Licensing of Production and Utilization Facilities [for nuclear facilities]). The ALARA program is implemented through PG&E Nuclear Power Generation (NPG) program directives, administrative procedures, and working level procedures. A detailed ALARA plan is maintained by PG&E and periodically reviewed by the NRC during the normal course of the NRC license and requirements. During the Proposed Project, the ALARA plan would be modified as necessary to address shielding and source removal. This plan modification would be included as part of the radiation work permit used to control radiation exposure to OSG removal workers. A radiation work permit is issued by the DCPP Radiation Protection Manager (a PG&E employee) prior to any activity that could result in radiation exposure. Compliance with these regulations and with the ALARA and radiation work permit are required by federal law.

The existing DCPP Health Physics Program[7] would be used during Proposed Project activities and would implement the requirements of 10 CFR 20 and the NPG policy for implementation of the ALARA philosophy for all site activities involving potential radiation exposure. The Radiation Protection Manager is responsible for administering, coordinating, planning, and scheduling all radiation protection activities involving the Proposed Project. The primary objective of the Health Physics Program is to maintain radiation exposures to workers, visitors, and the general public below regulatory limits and otherwise ALARA.

The estimated contact radiation dose rates on the exterior of each OSG would be less than or equal to 30 mrem per hour. Because dose rates decline when one moves further away from the radiation source, workers moving the OSGs to the OSG Storage Facility would receive radiation exposure at 30 mrem per hour or less (PG&E, 2004a and 2004b), which would be substantially lower than the DCPP worker exposure limit of 2 rem per year.

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6 Residual radiation refers to radiation that lingers after a facility or component (for example, a decommissioned nuclear plant or OSGs, respectively) is removed or otherwise is no longer operable. Often, allowable levels of residual radiation are so low as to fall below background radiation (i.e., what would occur had the facility never been built) (Wald, 2003).

7 The Health Physics Program covers all aspects of radiation exposure at the DCPP and is designed to comply with applicable radiation exposure limits and minimize radiation exposure to as low as reasonably achievable.

Contact dose rates on the exterior of the OSGs would be less than or equal to 30 mrem per hour with a mid-channel head (internal area) dose rate of 3 to 5 rem per hour. This assumes no system decontamination beyond shutdown chemistry control. Prior to transport to the OSG Storage Facility, the OSGs would be encapsulated in a protective coating to prevent the release of loose contamination. Steel covers would also be installed on piping openings to seal the internal portion of the OSG, effectively minimizing potential exposure to residual contamination (PG&E, 2004a and 2004b).

At the time of OSG removal, the radionuclide mixture would be representative of a typical reactor coolant system cleanup filter. The spectrum of the expected radionuclide source during OSG removal is shown in Table D.12-3. Based on the spectrum as shown in Table D.12-3, PG&E estimates the activity to be about 240 Curies (one Curie equals approximately the radioactivity of one gram of radium) for one steam generator.

Table D.12-3. OSG Radionuclide Inventory

Nuclide*	Curies/OSG	Nuclide	Curies/OSG
Fe-55	1.23E+02	Pu-241	3.60E-01
Zn-65	3.91E+01	Ni-59	1.54E-01
Co-60	3.57E+01	Cm-242	1.27E-02
Ni-63	1.50E+01	Sr-89	5.80E-03
Co-58	1.13E+01	Sr-90	3.72E-03
C-14	5.83E+00	Pu-238	2.27E-03
H-3	2.30E+00	Pu-239	1.93E-03
Nb-95	2.10E+00	Pu-240	1.93E-03
Mn-54	1.69E+00	Am-241	1.84E-03
Zr-95	1.05E+00	Cm-243	1.71E-03
Ru-106	7.94E-01	Cm-244	1.71E-03
Sb-125	6.69E-01	Tc-99	2.99E-05
Ce-144	6.69E-01	Pu-242	2.35E-05
Ag-110m	5.37E-01	Np-237	2.14E-06
Co-57	4.18E-01	I-129	6.99E-09
Sn-113	3.82E-01	Total:	2.41E+02

Source: PG&E, 2004a.
* A general term applicable to all atomic forms of an element.

Based on the radiological spectrum of each OSG and radioactivity of 240 Curies per steam generator, potential combined offsite project-related radiation exposure from the entire transport process can be estimated. In order to address potential impacts associated OSG transport and radiation emissions, a dispersion modeling analysis was conducted to identify areas that would be vulnerable to elevated radiation dosage levels.

Radiological source terms were based on the isotope inventory provided by PG&E (PG&E 2004a), as shown in Table D.12-3. The release fractions were adjusted according to the methodologies contained in DOE standards DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, and DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23*, *Nuclear Safety Analysis Reports*.

Dispersion parameters were based on a standard Gaussian plume model as described in the DOE's Handbook on Atmospheric Diffusion using daytime and nighttime stability and wind speed combinations (Stability Class D at 5 m/s and Stability Class F at 2 m/s, respectively). A modeling analysis was conducted utilizing the Hotspot model developed at University of California Lawrence Livermore National Laboratory (LLNL). The LLNL Hotspot model contains several dispersion models that are more appropriate than a standard Gaussian plume model, including models for explosions and fires.

Dose conversion factors were based on DOE/EH-0071, *Internal Dose Conversion Factors for Calculations of Dose to the Public*. Potential dosage was estimated using a standard breathing rate (3.5×10^{-4} m^3/sec) for the duration of the release.

Modeling results indicate that the potential for adverse offsite impact is minimal. Worst-case modeling indicated a Proposed Project–related total exposure of 0.003 mrem at the DCPP control area boundary and 1.4×10^{-5} mrem at Avila Beach. These worst-case exposure levels are well below the most stringent public exposure limit of 10 mrem/year. Therefore, potential impacts associated with the removal and transport of the OSGs to the proposed onsite storage facility are considered less than significant (Class III).

Impact S-4: An aircraft accident could result in damage to the OSG Storage Facility with a subsequent release of radioactive material

Storage of the OSGs onsite would occur in a new facility designed to minimize the release of radioactive material. The most potentially damaging accident scenario that could compromise the integrity of the OSG Storage Facility is presumed to be an unintentional impact by aircraft. The risk of terrorism is addressed in Impact S-6 below. Potential aircraft accidents were evaluated in the ISFSI SAR (PG&E 2002a). This analysis was modified for the proposed OSG Storage Facility (see Attachment to Section D.12.7, OSG Storage Facility Aircraft Hazards at the end of this section). Taking into account the proximity of surrounding airports, established flight paths, and the number of flights in nearby corridors, the probability of an inadvertent aircraft strike on the OSG Storage Facility would be less than the threshold of one in one million per year (1×10^{-6}/year). Although an inadvertent aircraft strike would be improbable, the potential consequences are summarized below.

The consequences of an inadvertent aircraft strike on the proposed OSG Storage Facility were evaluated using the dispersion modeling methodology discussed under Impact S-3, with the exception that more radiation would be released in any fire that would occur subsequent to aircraft impact, and the resulting plume would be thermally buoyant. Given the large thermal plume rise, worst-case concentrations would occur on the hillsides surrounding the DCPP, with a maximum expected event dose of 4 mrem, and a maximum offsite dose of 1.4 mrem. These dosages conservatively assume that individuals would remain in the area during the duration of a fire and would also be exposed to substantial amounts of smoke. Comparable normal exposure values would be 20 mrem for normal background radiation and 20 mrem for a chest X-ray. These worst-case exposure levels are lower than acceptable exposure limits of 10 mrem per year for normal operations, and substantially lower than acceptable levels for one-time exposure accidents of 5 rem (5,000 mrem). Therefore, impacts associated with an inadvertent aircraft strike on the OSG Storage Facility and subsequent catastrophic loss of containment are less than significant (Class III).

Impact S-5: Seismic activity could compromise the integrity of the OSG Storage Facility

The ISFSI SAR and a 1988 Long Term Seismic Program (LTSP) report completed by PG&E addressed potential ground shaking, fault rupture, and seismicity at the DCPP site. Numerous additional published documents, and data on earthquake activity in the area over the past 20 years, are available to supplement and compare the information provided in the PG&E reports. The anticipated ground motions at the DCPP site as determined in the LTSP form the present design basis for the proposed OSG Storage Facility. Severe ground shaking (Impact G-3, as described in Section D.5, Geology, Soils, and Paleontology) could compromise the integrity of the OSG Storage Facility, if the facility design does not incorporate recent earthquake data. Seismic activity would result in a less than significant impact with implementation of the 1988 LTSP update as recommended by Mitigation Measure G-3a (Class II).

Mitigation Measures for Impact S-5, Seismic activity could compromise the integrity of the OSG Storage Facility

Implement Mitigation Measure G-3a.

Impact S-6: A terrorist attack could result in damage to the OSG Storage Facility with a subsequent release of radioactive material

Under the Proposed Project, potential terrorist attacks on the proposed OSG Storage Facility would be considered inconsequential, as demonstrated in the analysis for Impact S-4. The analysis for an accidental aircraft strike on the OSG Storage Facility assumed worst case conditions with consequences similar to a willful terrorist attack on the OSG Storage Facility. Thus, although impacts associated with a potential terrorist attack on the OSG Storage Facility could affect PG&E and ratepayers, potential safety risks would be considered less than significant (Class III).

D.12.3.5 Replacement Steam Generator Installation

Replacement of the DCPP steam generators would, at a minimum, allow the facility to operate through the end of its current license periods for each unit. The NRC DCPP Unit 1 and 2 operating licenses expire in September 2021 and April 2025, respectively. Therefore, the Proposed Project would extend the operating life of the DCPP, and the environmental effects of current operations would continue as a result of CPUC approval of the project. However, the risk associated with the DCPP operating to the end of the current license periods has already been evaluated, and therefore is part of the baseline.

Comments received on the Notice of Preparation (NOP) touched on a wide variety of issues associated with the continued operation of the DCPP. The Applicant's stated position is that the continued operation of the DCPP through the existing license period is part of the CEQA baseline (i.e., the conditions that existed at the time the NOP was issued). Other parties contend that approving the Proposed Project would extend the operational life of the DCPP at least until the expiration date of the licenses, because the DCPP would be more likely to shut down as a result of OSG deterioration if the CPUC denies Proposed Project cost recovery.

Since the NRC evaluated potential impacts associated with the DCPP through the end of the current license periods and, more importantly, the NRC licenses for DCPP Units 1 and 2 were approved for operation through the end of these license periods, the Proposed Project does not extend the life of the DCPP beyond that period for which potential environmental impacts have already been evaluated. PG&E does not need CPUC approval to replace the steam generators, a process that has occurred at numerous other commercial reactors around the country, but only approval for project cost recovery. In addition, no amendment to the NRC license is required for the steam generator replacement project.[8] Therefore, given that both DCPP impacts have been evaluated through the end of the license periods and that DCPP received license approval to operate through the end of these license periods, the Proposed Project does not change the baseline risk of routine DCPP operations.

Should PG&E seek to extend the current DCPP license periods, additional environmental review would be required that is beyond the scope of this EIR. See Section G for additional information on the NRC license renewal process and PG&E's position on license renewal at DCPP.

[8] PG&E has stated that "the steam generator replacement outages will not create any non-normal refueling situations and will not require any updates or changes to the NRC license" (PG&E, 2004e), and "it will not be necessary to file for a license amendment with the NRC to install the replacement steam generators" (PG&E, 2004d).

D.12.4 Environmental Impacts and Mitigation Measures for the Alternatives

D.12.4.1 Replacement Steam Generator Offloading Alternative

Transport of the RSGs to the DCPP Intake Cove would avoid transportation and navigational safety hazards identified in Impact S-1, and Mitigation Measure S-1a would not be necessary. There would be no appreciable transportation hazards to the public associated with this alternative since the facility currently employs a one-mile exclusion zone seaward of the facility. In addition, the only offshore activities near DCPP would be associated with PG&E or contractor activities.

D.12.4.2 Temporary Staging Area Alternatives

All activities associated with each of the temporary staging options would be the same as the Proposed Project in terms of public safety. Therefore, no adverse impacts would occur to public safety.

D.12.4.3 Original Steam Generator Storage Facility Location Alternatives

From a public safety perspective, each of the OSG Storage Facility Location Alternatives is essentially the same as the proposed OSG Storage Facility location.

D.12.4.4 Original Steam Generator Offsite Disposal Alternative

OSG disposal offsite would avoid the minimal risks associated with the OSG Storage Facility, which are shown in Section D.12.3.4 to be less than significant. Barge traffic necessary to remove the OSG from the coast would cause essentially the same impacts as addressed in Impact S-1, which could be reduced to a less than significant level through development of a barge navigational safety plan (Mitigation Measure S-1a). Additionally, transport of the OSG offsite would result in potential offsite public exposure to residual radiation as the OSGs are transported from the DCPP to an approved offsite disposal facility.

As noted in Section D.12.1, there are only three facilities in the U.S. that can accept LLW like the OSGs that would be removed from DCPP. Only two of these facilities will accept LLW from facilities in California, and beginning in 2008, which is when the OSGs would be removed from service, only one site will be available to receive LLW from California facilities. Therefore, all LLW generated at the DCPP can only be stored onsite or disposed at the Envirocare disposal site located in Clive, Utah. The OSGs would likely be classified as Class A LLRW, which is the only waste class accepted at the Envirocare facility. CPUC does not have any authority to mitigate potential impacts at LLW disposal sites.

Many issues associated with offsite disposal could be faced by LLW compacts, compact host-site disposal states, and unaffiliated states during license renewal across the nuclear industry. Although routine waste stream volumes continue to decline, major refurbishments that occur during license renewal could produce additional short-term volumes that could tax available offsite disposal space in some compact and unaffiliated states, including California. Should the Envirocare of Utah disposal site experience a short-term increase in volume in 2008, it would be possible that the facility may not be able to accept the DCPP steam generators, leaving PG&E with no other alternative than to store the OSGs onsite, at least temporarily.

Impact S-7: Residual contamination would be present on the OSGs with the potential for radiation exposure during offsite transport

Assuming a worst-case scenario of transporting the OSGs through Port San Luis, those members of the public that would choose to remain in the immediate area of OSG barge loading would experience some exposure to residual OSG contamination. Several measures would be taken to minimize residual radiation exposure (PG&E, 2004), including:

- All openings and penetrations would be covered and welded closed,
- OSGs would be treated with a polymer coating to bind any external contamination, and
- External shielding would be utilized to reduce occupational and transportation dose as appropriate.

Exposure to individuals in the area during OSG barge loading would likely experience radiation exposure, but at levels that are far below applicable regulatory levels (e.g., 10 mrem/year). Modeling results indicate that the unmitigated exposure rate would be approximately 0.1 mrem/hour at a distance of 50 meters. This would provide for a total loading time of 100 hours before the acceptable public exposure limit of 10 mrem/year is reached. Radiological monitoring would be required by NRC to protect worker and public health, thus it is unlikely that public exposure would exceed 10 mrem/year. Therefore, potential impacts associated with OSG transport offsite would be considered less than significant (Class III). To further reduce the severity of this impact, the following measure is recommended for protecting public health to the maximum extent feasible.

Mitigation Measures for Impact S-7, Residual contamination would be present on the OSGs with the potential for radiation exposure during offsite transport

S-7a **Alternate OSG Barge Loading Site.** In order to avoid exposing the public to residual OSG contamination, the Applicant shall utilize the DCPP Intake Cove for OSG barge loading should offsite transport and storage be required.

D.12.5 Environmental Impacts of the No Project Alternative

The No Project Alternative describes a range of options for replacing the 2,200 MW of DCPP baseload system generation. The most likely option would involve natural gas-fired combined-cycle power plants, based on the 60-hertz version of GE's most advanced gas turbine technology, the H System.

Potential safety impacts associated with natural gas-fired combined-cycle power plants have been well documented as part of the California Energy Commission (CEC) facility siting process. Depending on the exact location of a facility in relation to the public, potential safety impacts can be significant. However, safety impacts associated with any proposed power generation facility would be subjected to site-specific environmental review.

Typical natural gas-fired combined-cycle power plants would likely be equipped with selective catalytic reduction (SCR) emission control equipment to reduce emissions of criteria air pollutants. SCR utilizes ammonia, either anhydrous or aqueous solutions, as part of the SCR process. Hazards associated with the transportation, storage and use of ammonia can result in significant public risk in the event of an accidental release near populated areas. Typical measures to avoid this impact would be identifying the route or method of delivery for ammonia (e.g., truck, rail, or pipeline) that would avoid populated areas or accident hazards or requiring the use of a urea-based system.

The siting of natural gas-fired combined-cycle power plants usually requires the construction of a large capacity, high pressure natural gas pipeline. Hazards associated with natural gas transmission pipelines can result in significant public risk in the event of an accidental release. However, power plants are generally sited in industrial areas which minimize potential public and sensitive receptor exposure to increased risk.

The description of the environmental setting above demonstrates that the probability of a core-damaging accident for a facility like DCPP is greater than ten in one million per year (1×10^{-5}/year). Should the Proposed Project not move forward, resulting in a cessation of DCPP operation before the end of the current licensing periods for both units, the baseline risk associated with routine DCPP operations would no longer be present. Under the No Project Alternative, PG&E estimates that the DCPP Unit 1 and 2 reactors would need to be shut down approximately 10 and 12 years before the end of their current license periods, respectively. Thus, the DCPP would shut down and the risk of a reactor-related accident with severe consequences, including early fatalities, injuries, cancer deaths, and related property damage, would cease. Based on the decreased probability of a core-damaging accident associated with the decreased plant life that would occur under the No Project Alternative, this reduction in DCPP risk would be considered a beneficial impact (Class IV).

Baseline safety and risk conditions are characterized by potential releases from three main systems at the DCPP, including:

- Units 1 and 2 reactors and systems,
- Spent Fuel Pools, and
- Independent Spent Fuel Storage Installation.

Potential hazards associated with spent fuel handling, both at the spent fuel pool and ISFSI were evaluated as part of the ISFSI EIR prepared by San Luis Obispo County (SLO County, 2004), and were found to represent a significant risk in the absence of mitigation. Since the NRC has sole jurisdiction over all aspects of nuclear safety, radiological health issues, and the storage and handling of spent nuclear fuel, many of the mitigation measures identified in the ISFSI EIR were not adopted. Therefore, the continued operation of these facilities was considered by San Luis Obispo County to cause a significant and unavoidable impact to public safety.

The probability of an accidental release associated with spent fuel also increases with time as more spent fuel is accumulated. Historically, the solution to increasing volumes of spent fuel has been the re-racking of the spent fuel pools to accommodate more waste, and increasing the probability of consequences of an accidental release. As the ISFSI is constructed in a phased approach, the likelihood of an accident increases as the number of operational casks increases. ISFSI-related safety issues were evaluated in the ISFSI EIR (SLO County, 2004). The No Project Alternative would result in over 1,000 less spent fuel assemblies being moved into storage during the 10 to 12 years leading up to NRC license expiration. Therefore the No Project Alternative would reduce the risk associated with spent fuel handling, resulting in a beneficial impact (Class IV).

The issue of equipment and infrastructure aging at the DCPP was raised in NOP comments citing the need to replace the DCPP steam generators as an example. All equipment at the DCPP has a limited useful service life, with reliability being a concern as equipment ages. All equipment failure rates are measured in units of time, such as failures per year or mean time between failures, thus implying that the probability of a failure increases as components age. The replacement of the DCPP steam generators is in direct response to the long-term wear of these components and the concern for future failures. Metal fatigue, cracking and corrosion has limited the efficiency and reliability of these components. Nevertheless, continued operation of the DCPP would result in an increased probability of component failure and an accidental release. However, it should be noted that replacement of the steam generators would actually reduce the probability of an accidental release for those accidents related to steam generator tube failures.

The NRC (2004) has noted that the risk associated with steam generator tube ruptures is relatively low due to the effectiveness of NRC regulatory guidance and requirements, and represents only a small fraction of the facility risk. This is mainly due to the effectiveness of isolating defective tubes in the steam generators, and minimizing the potential for a tube rupture. Therefore, replacement of the steam generators would not likely change long term risk, and the core damage frequency for the DCPP Pressurized Water Reactors would remain higher than 1.0×10^{-5}/reactor-year with or without the Proposed Project. Reducing the operational life of the DCPP under the No Project Alternative would limit the number of years that the DCPP would operate, along with the associated probability of an accident due to steam generator tube ruptures, and would be considered a impact (Class IV).

Similar to the aging infrastructure issue, at least one NOP comment addressed the issue of an aging DCPP workforce, with 39 percent of the PG&E workforce becoming eligible for retirement during the planned project. While worker knowledge and retention has an effect on safety, it would be speculative to assume that all 39 percent of the eligible workforce would retire during the Proposed Project. In addition, PG&E would need to replace workers as necessary and provide all required training. Thus, no change in safety or risk can be attributed to an aging workforce.

Terrorism is also an issue that has been raised in the NOP comments. As noted in the ISFSI EIR (SLO County, 2004), prior to the events of September 11, 2001, the prospect of an airborne terrorist attack on the DCPP would have been considered highly speculative under CEQA and dropped from further analysis. While it would be nearly impossible to estimate the probability of an aircraft-based terrorist attack on the DCPP, much less the likelihood of a successful attack and containment breach, the possibility of such an attack cannot be totally discounted. The consequences associated with such an attack would be substantial, as described in the environmental setting above. The No Project Alternative would lead to a cessation of DCPP operations, which would reduce the consequences of a terrorist attack, resulting in a beneficial impact (Class IV). With or without the Proposed Project, the vulnerability of the spent fuel pools and ISFSI would remain.

D.12.6 Mitigation Monitoring, Compliance, and Reporting Table

Table D.12-4 shows the mitigation monitoring, compliance, and reporting program for System and Transportation Safety.

Table D.12-4. Mitigation Monitoring Program – System and Transportation Safety

IMPACT S-1	**RSG barges would create a navigational hazard in Port San Luis (Class II)**
MITIGATION MEASURE	**S-1a: Barge Navigational Safety Plan.** The Applicant shall develop a barge navigational safety plan to minimize the impact on existing Port operations. The plan shall be submitted to, and approved by the Port San Luis Harbor District. At a minimum the plan shall include the following elements: • Identify moored vessels that will need to be temporarily relocated, and provide necessary temporary mooring facilities or funding for the Port District to accommodate the temporary relocation of moored vessels. • Identify activities such as home fleet and dry dock operations in the Port that may conflict with barge transport and/or offloading. Working with the Harbor District, identify procedures that will minimize conflicts with existing operations. • Working with the Harbor District, identify additional navigational aids and security that will be necessary to safely move the barges through the Port. The Applicant can provide the additional navigational aids and/or security, or contract with the Harbor District to provide the necessary services.
Location	Port San Luis
Monitoring / Reporting Action	Submit Navigational Safety Plan for approval
Effectiveness Criteria	Avoidance of vessel collisions and Port operations
Responsible Agency	CPUC, Port San Luis Harbor District
Timing	Prior to RSG delivery
IMPACT S-2	**RSG transport between Port San Luis and the DCPP could impede emergency response vehicles (Class II)**
MITIGATION MEASURE	Implement Mitigation Measure U-2a.
Location	As in Mitigation Measure U-2a.
Monitoring / Reporting Action	As in Mitigation Measure U-2a.
Effectiveness Criteria	As in Mitigation Measure U-2a.
Responsible Agency	As in Mitigation Measure U-2a.
Timing	As in Mitigation Measure U-2a.
IMPACT S-5	**Seismic activity could compromise the integrity of the OSG Storage Facility (Class II)**
MITIGATION MEASURE	Implement Mitigation Measure G-3a.
Location	As in Mitigation Measure G-3a.
Monitoring / Reporting Action	As in Mitigation Measure G-3a.
Effectiveness Criteria	As in Mitigation Measure G-3a.
Responsible Agency	As in Mitigation Measure G-3a.
Timing	As in Mitigation Measure G-3a.

Table D.12-4. Mitigation Monitoring Program – System and Transportation Safety

IMPACT S-7	Residual contamination would be present on the OSGs with the potential for radiation exposure during offsite transport (Class III)
MITIGATION MEASURE	S-8a: Alternate OSG Barge Loading Site. In order to avoid exposing the public to residual OSG contamination, the Applicant shall utilize the DCPP Intake Cove for OSG barge loading should offsite transport and storage be required.
Location	DCPP, Port San Luis
Monitoring / Reporting Action	Use alternate barge loading site
Effectiveness Criteria	Avoidance of impact
Responsible Agency	CPUC, Port San Luis Harbor District, NRC
Timing	Prior to OSG removal

D.12.7 References

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Attachment to Section D.12.7 – OSG Storage Facility Aircraft Hazards

Summary of Air Traffic in the Vicinity of the Proposed OSG Storage Facility

The SLO County Airport is located 12 miles east of the DCPP site. The airport serves approximately 52 scheduled landings and departures per day of commercial commuter flights, provided primarily by turbo-prop aircraft that seat no more than 41 people with a gross weight of no more than 30,000 pounds. The airport also serves approximately 10,000 total landings and departures of private aircraft per month. These consist mostly of aircraft that seat no more than 8 people, with an average gross weight of less than 12,500 pounds. There are no specific air traffic restrictions over DCPP.

Federal flight corridor (V-27) is located approximately 5 miles east of the OSG Storage Facility that is used for aircraft flying between Santa Barbara and Big Sur areas, with an estimated 20 flights per day. The majority of the aircraft using this route is above 10,000 feet. Occasionally, this corridor is used also for traffic into San Luis Obispo Airport and, in this case, has traffic that passes as close as 1 mile of the OSG Storage Facility site at an elevation of 3,000 feet. Typically this portion of the route is only used for aircraft to align for instrument landing. The more commonly used approach route for visual landings passes 8 miles from the proposed OSG Storage Facility site on the far side of the San Luis Range.

Military training route (VR-249) runs parallel to the coast with its center approximately 2 miles off shore of DCPP. This training route is not frequently used with an estimated number of flights of less then 60 per year. Use of VR-249 requires a minimum of 5 miles visibility, with flights required to maintain their altitude between sea level and 10,000 feet.

A municipal airport near Oceano is located 15 miles east-southeast of the DCPP site, which accommodates only small (12,500 pounds or less) private planes. The traffic at this airport is estimated to be no more than 2,200 flights per month. The Camp San Luis Obispo airfield is located 8 miles northeast of the DCPP site, but it is now shown as helicopter use only.

There are currently no over flight prohibitions in and around the DCPP. Pilots are provides with the following advisory:[9]

> *Special Notice… Flight Restrictions. Effective immediately until further notice, pursuant to 14 CFR Section 99.7, special security instructions, pilots conducting flight operations within the territorial airspace of the U.S. are advised to avoid the airspace above or in proximity to all nuclear power plants. Pilots should not circle or loiter in the vicinity of such facilities. Pilots who do so can expect to be interviewed by law enforcement personnel at their destination airport and the pilot's name may be added to the transportation security administration incident reporting system.*

Aircraft crashes were assessed in PG&E's ISFSI SAR (2002a) in accordance with the guidance of NUREG-0800, Section 3.5.1.6, Aircraft Hazards. Although this guidance applies to power reactor sites, the analysis of aircraft crash probabilities on the site is not dependent on the nature of the site other than size of the facility involved and, thus, the guidance of NUREG-0800 can be applied to the proposed OSG Storage Facility site. This approach is similar to that used in the evaluation of potential

[9] Federal Aviation Administration (FAA) Notices to Airmen (NOTAMs).

aircraft accidents at the DCPP ISFSI (SLO County, 2004) and proposed Skull Valley Spent Fuel Storage Facility (Atomic Safety and Licensing Board, 2003).

As specified in NUREG-0800 (Item II.1 of Section 3.5.1.6), the probability of aircraft crashes is considered to be negligibly low by inspection and does not require further analysis if three criteria are met. Criterion 1 specifies that the plant-to-airport distance (D) must be greater than 10 statute miles, and the projected annual number of operations must be less than 1000 times the distance-squared (D^2). San Luis Obispo Airport is at a distance of 12 miles, with annual flight totals of approximately 139,000, which is less than $1000(12)^2$ or 144,000. The airport at Oceano is 15 miles away, with flight totals of no more than approximately 26,400 per year, which is less than $1000(15)^2$ or 225,000. Vandenberg Air Force Base is 35 miles away and flight totals there are not expected to be more than $1000(35)^2$ or 1,225,000 per year (or more than 3,300 each day). Although based on current data, Criterion 1 is met for the airways that are in the vicinity of the proposed OSG Storage Facility, hazards have been analyzed below.

Criterion 2 specifies that the facility must be at least 5 statute miles from the edge of military training routes. There is a military training flight corridor (VR-249) that is within approximately 2 miles of the proposed OSG Storage Facility site. This route is also evaluated below.

Criterion 3 specifies that the facility must be at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern. There is a federal airway (V-27) whose edge would be within approximately 1 mile east of the OSG Storage Facility site. As a result, this route is evaluated below.

Evaluation of Air Traffic Hazards to the DCPP OSG Storage Facility

To determine the probability of an aircraft crash into the proposed OSG Storage Facility, a four-factor formula of the "Aircraft Hazards" portion of NUREG-0800 was utilized. This approach has regularly been used to calculate the risks of aircraft crashing into NRC-regulated facilities.

The analysis contained in the PG&E ISFSI SAR (2002a) was used as the basis for this analysis with some modification. The PG&E ISFSI SAR utilized some conservative and non-conservative estimates for many of the parameters used to estimate aircraft crash probabilities. These values have been refined in this analysis to provide a consistent set of parameters for each aircraft category. Following NUREG-0800, the following equation was used to calculate the probability:

$$P = C \times N \times A/w$$

where:

C	=	the aircraft's historic accident rate (in accidents per mile flown);
N	=	the number of flights per year;
A	=	the effective area of the facility (in square miles); and
w	=	the width of the airway (in miles).

For circumstances where the site is located outside of the airway, the airway width (w) is estimated as the airway width plus twice the distance of the site to the airway edge.

The PG&E analysis was completed per DOE-STD 3014-96 to determine effective crash area. In this analysis conservative factors were used for maximum skid distance and maximum wingspan. Based on the available information on aircraft type, size, and the location of the site these factors are conservative, especially when considering the terrain in the immediate vicinity of the proposed OSG Storage Facility site.

The effective crash area was estimated by PG&E and is defined as:

Aeff = Af + As

where:

Af = (WS + R) (HcotΦ) +(2*L*W*WS)/R+ (L*W)
As = (WS + R)*S

where:

Af = effective fly-in area;
As = effective skid area;
WS = aircraft wingspan; (reference Table B-16 of DOE Standard)
R = length of diagonal of the facility,
H = facility height;
cotΦ = mean on the cotangent of the aircraft impact angle;
L = length of facility;
W = width of facility;
S = aircraft skid distance; (reference Table B-18 of DOE Standard)

A sample calculation for a large military aircraft would be:

$$
\begin{aligned}
Af &= (223 + 226)(30)(10.2) + (2*160*160*223)/226 + (160*160) \\
&= 206798 ft^2/(5280\ ft/mile)^2 \\
&= 0.0074 \text{ sq miles}
\end{aligned}
$$

and

$$
\begin{aligned}
As &= (WS + R)*S \\
&= (223 + 226)(700) \\
&= 0.0113 \text{ sq miles}
\end{aligned}
$$

For calculating "As" the skid distance is based on the layout of the facility which is surrounded on three sides by hills and is embedded up against one of the hills. The location limits the potential crash angle and possible skid distance. The fourth side is protected by a drop off in terrain with a slope of greater than 1:1, with little likelihood that an aircraft approaching from this side would substantially impact the OSG Storage Facility site. The maximum distance on the unprotected side is estimated at less than 700 feet. Because the site is protected and limited from skidding aircraft on three sides, the use of 700 feet is conservative.

$$
\begin{aligned}
Aeff &= Af + As \\
&= 0.0074 + 0.0113 \\
&= 0.0187 \text{ sq miles}
\end{aligned}
$$

The above calculations generally utilize worst case values for military aircraft. Effective facility areas would be substantially smaller for commercial and general aviation aircraft. Based on the data in DOE STD 3014-96, worst-case values for aircraft wing span, and effective facility area are as follows:

Aircraft Class	Wing Span (feet)	Skid Distance (feet)	Cotangent Impact Angle	Facility Diagonal (feet)	Facility Height (feet)	Facility Length (feet)	Facility Width (feet)
Military Aircraft	223	700	9.7	226	30	160	160
Commercial Aviation	98	700	10.2	226	30	160	160
General Aviation	50	60	8.2	226	30	160	160

Given these aircraft class specific values, the facility effective area is presented below.

Aircraft Class	Af (sq miles)	As (sq miles)	Aeff (sq miles)
Military Aircraft	0.0074	0.0113	0.0187
Commercial Aviation	0.0053	0.0081	0.0134
General Aviation	0.0038	0.0006	0.0044

The values presented above are generally conservative as they represent the largest values for each aviation category. Thus potential estimates of aircraft strike probabilities would be considered conservative worst-case values.

Federal Flight Corridor V-27. The use of V-27 for local aircraft is usually limited to instrument landings for aircraft arriving from the south and instrument departures to the south from runway 19 at the San Luis Obispo Airport. There are approximately 52 scheduled commercial aircraft landings and takeoffs per day, most of which occur during daylight hours. It is estimated that approximately 50% of the commercial traffic comes from or departs to the south with 10% of the commercial aircraft using this route under instrument conditions. For private aircraft usage, there are approximately 10,000 total landings and takeoffs per month at the San Luis Obispo Airport of which 50% were also assumed to originate to the south.

Because of limited instrument landing capability and qualification, conservatively only approximately 5% would be flying under instrument conditions and on this route. As a result, N for commercial aircraft would be equal to approximately 949 flights per year (52 x 0.5 x 0.1 x 365) and for general aviation to 3,000 flights per year (10,000 x 0.5 x 0.05 x 12).

Published holding patterns exist for arrivals at CREPE and CADAB intersections and for missed approaches, at Morro Bay VOR. The CREPE Intersection is 11 miles and the CADAB Intersection 21 miles from the proposed OSG Storage Facility site. Both holding patterns place the aircraft further from the OSG Storage Facility site and therefore do not need to be considered. The OSG Storage Facility site distance to the Morro Bay VOR is approximately 6 miles and the holding pattern places the aircraft closer to the OSG Storage Facility. Because the Morro Bay VOR holding pattern is used for missed approaches, it is conservatively estimated that 5% of all instrument landing approaches are missed and each aircraft remains in the holding pattern for ten passes. For commercial traffic N is increased by 237 flights (949/2 x 0.05 x 10) and general aviation by 750 flights (3,000/2 x 0.05 x 10).

Per NUREG-0800, C for commercial aircraft is provided as 4×10^{-10}/mile, which is the value used by PG&E. However, data from the Bureau of Transportation Statistics (BTS) shows that the actual accident rate for US carriers has been approximately 1×10^{-8}/mile for the period 1995-2000 (BTS 2003). The NUREG-0800 rate is based on aircraft accidents during flight and excludes activities such as takeoff, landing, ascent and descent. Therefore, for this part of the analysis addressing aircraft from San Luis Obispo Airport, the more conservative BTS data for all flight operations was utilized.

For general aviation no specific data is provided in NUREG-0800; thus PG&E assumed a value of 1×10^{-8}/mile in its analysis. However, DOE-STD 3014-96 provides relative generic maximum crash rates for general and commercial aviation. Based on the ratio of these values with the actual BTS crash data, a general aviation crash rate of 6×10^{-7}/mile was estimated for use in this analysis. In addition, the effective area of the facility, A, was also modified to reflect the smaller impact area associated with general aviation aircraft.

The width of the V-27 airway is 8 miles and the center is approximately 5 miles from the site. As a result, (w) is estimated to be 18 miles (8-mile width plus two times the distance from the site to the airway edge of 2 times 5 miles).

For commercial flights:

$$\begin{aligned} P1c &= C \times N \times A/w \\ &= (1 \times 10^{-8}) \times (949 + 237) \times (0.0134)/(18) \\ &= 8.84 \times 10^{-9} \end{aligned}$$

For general aviation flights:

$$\begin{aligned} P1g &= C \times N \times A/w \\ &= (6 \times 10^{-7}) \times (3{,}000 + 750) \times (0.0044)/18 \\ &= 5.45 \times 10^{-7} \end{aligned}$$

Total local aircraft crash potential:

$$\begin{aligned} P1 &= P1c + P1g \\ &= 8.84 \times 10^{-9} + 5.45 \times 10^{-7} \\ &= 5.53 \times 10^{-7} \end{aligned}$$

For commercial traffic flying on V-27 and not landing locally, V-27 is a federal flight route from the Santa Barbara area northwest to the Big Sur area. Most of the aircraft on this route are normally flying at altitudes above 10,000 feet, with some smaller aircraft at elevations as low as 3,500 feet. Per the FAA Standards Office, the number of aircraft on this route is conservatively estimated at 20 per day or 7,300 per year. Using the same data as above and adjusting for the number of flights:

$$\begin{aligned} P2 &= C \times N \times A/w \\ &= (4 \times 10^{-10}) \times (7300) \times (0.0134)/(18) \\ &= 2.18 \times 10^{-9} \end{aligned}$$

Military Aircraft Flying on VR-249. VR-249 is a military training route, which requires 5 miles visibility and the ceilings above 3,000 feet. The aircraft may be traveling between sea level and 10,000 feet. The route is used very infrequently and is estimated to have approximately 50 flights a year. In the area of the DCPP OSG Storage Facility this route is provided for normal flight modes and is not expected to include any high-stress maneuvers. The majority of the aircraft flying this route over the past 12 months were F-18s. In addition, there have been a limited number of C-130, F-16 and EA6B aircraft and some helicopters using this route. For this calculation, N is conservatively taken to be 75 flights. The center of the route is approximately 2 miles off shore; therefore, (w) is set at 1 mile in this calculation. There was no data provided in the NUREG for military aircraft that would support this route and as a result, the in flight crash probability for F-16s accepted for the Skull Valley Spent Fuel Storage Facility of 2.736×10^{-8} was used.

P3 = C x N x A/w
= $(2.736 \times 10^{-8}) \times (75) \times (0.0187)/(1)$
= 3.84×10^{-8}

Military Ordinance on Aircraft on VR-249. Based on information provided by the Naval Air Station at Lenore, which flies a majority of the flights on VR-249, aerial bombs are not carried. However, other ordinance such as air-to-air missiles and cannon/machine guns might be carried on a very small number of the military aircraft on this route. Accidental firings of air-to-air missiles or aircraft guns have not been reported. In addition, air-to-air ordinance does not have a large explosive charge and would not be expected to cause major damage to non-aircraft targets.

VR-249 is a visual route, which requires a minimum of 5 miles of visibility and minimum ceilings of 3,000 feet. Aircraft using this route normally remain offshore and do not fly directly over the Diablo Canyon Power Plant or the proposed OSG Storage Facility site. Based on the type of ordinance, the miniscule probability of an accidental discharge, and the visual requirements of the route the potential for any possible interaction between the ordinance and the proposed OSG Storage Facility is not credible.

Summary of Aircraft Hazards

The landing patterns and distances to the local airports would not significantly affect the probability of a crash at the proposed OSG Storage Facility site. However, there a designated a military training flight corridor (VR-249) within the limits of Criterion 2 of NUREG-0800. As result, the total aircraft hazard probability at the DCPP OSG Storage Facility site is equal to the sum of the individual probabilities calculated above.

Total = P1 + P2 + P3
= $(5.53 \times 10^{-7}) + (2.18 \times 10^{-9}) + (3.84 \times 10^{-8})$
= 5.94×10^{-7}

Based on the above calculation, the total aircraft hazard probability is determined to be approximately 5.94×10^{-7}/year, which is less than the threshold of 1×10^{-6}/year specified in DOE-STD-3014-96 for acceptable frequency of aircraft impact into a facility from all types of aircraft. Given the conservative nature of the methodology used to estimate this frequency, as well as the fact that it is highly unlikely that a general aviation aircraft, which constitutes the highest accident probability, would breach OSG Storage Facility, the probability of potential impacts associated with aircraft accidents would be considered less than significant.