DIABLO CANYON INDEPENDENT SPENT FUEL STORAGE INSTALLATION SAFETY EVALUATION REPORT

Prepared for

U.S. Nuclear Regulatory Commission
Contract No. NRC–02–02–012

Prepared by

J. Stamatakos, A. Chowdhury, B. Dasgupta, D. Dunn, A. Ghosh, D. Gute, B. Hill, S. Hsiung, P. Mackin, C. Manepally, G. Ofoegbu, D. Pomerening, O. Povetko, R. Sewell, M. Smith

Center for Nuclear Waste Regulatory Analyses
San Antonio, Texas

October 2003

TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

FIGURE

TABLES

ACRONYMS

ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
AWS	American Welding Society
BWR	Boiling Water Reactor
CNWRA	Center for Nuclear Waste Regulatory Analyses
COC	Certificate of Compliance
CTF	Cask Transfer Facility
DDE	Double Design Earthquake
DE	Design Earthquake
DCPP	Diablo Canyon Power Plant
DOE	U.S. Department of Energy
DSHA	Deterministic Seismic Hazard Analysis
EIS	Environmental Impact Statement
FAA	Federal Aviation Administration
FEMA	Federal Emergency Management Agency
FHB/AB	Fuel-Handling Building and Auxiliary Building
FSAR	Final Safety Analysis Report
GSI	Geological Strength Index
HE	Hosgri Earthquake
ILP	Long Period
ISFSI	Independent Spent Fuel Storage Installation
JCS	Joint Compressive Strength
JRC	Joint Roughness Coefficient
LTSP	Long-Term Seismic Program
M	Magnitude (Earthquake)
MLLW	Mean Lower Low Water
MPC	Multi-Purpose Canister
MSL	Mean Sea Level
MTU	Metric Tons of Uranium
NRC	U.S. Nuclear Regulatory Commission
NFPA	National Fire Protection Association
PGA	Peak Ground Acceleration
PG&E	Pacific Gas and Electric Company
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PSHA	Probabilistic Seismic Hazard Analysis
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAI	Requests for Additional Information
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SFPE	Society of Fire Protection Engineers
SSC	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
TEDE	Total Effective Dose Equivalent
TNT	Trinitrotoluene
ZPA	Zero Period Acceleration

ACKNOWLEDGMENTS

This report was prepared to document work performed by the Center for Nuclear Waste Regulatory Analyses for the U.S. Nuclear Regulatory Commission under Contract No. NRC–02–02–012. The activities reported here were performed on behalf of the U.S. Nuclear Regulatory Commission Office of Nuclear Material Safety and Safeguards, Spent Fuel Project Office. The report is an independent product of the Center for Nuclear Waste Regulatory Analyses (CNWRA) and does not necessarily reflect the views or regulatory position of the U.S. Nuclear Regulatory Commission.

The authors thank K. Murphy for technical support; G. Cragnolino, B. Dasgupta, L. Howard, H.L. McKague, and B. Sagar for technical reviews; and W. Patrick for programmatic review of this report. The authors also appreciate C. Patton, A. Ramos, J. Gonzalez, K. Murphy, C. Cudd, B. Long, J. Pryor, and A. Woods for editorial support in the preparation of the document.

QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT

DATA: No CNWRA-generated original data are contained in this report.

ANALYSES AND CODES: The computer codes ORIGEN-ARP and MCNP Version 4C2 were used to perform confirmatory source term and dose rate calculations described in this report. These computer codes are controlled under the CNWRA Quality Assurance program. Scientific Notebook No. 565E contains material relevant to this report.

EXECUTIVE SUMMARY

On December 21, 2001, the Pacific Gas and Electric (PG&E) Company submitted a license application in accordance with 10 CFR Part 72 to the U.S. Nuclear Regulatory Commission (NRC) to construct and operate an onsite independent spent fuel storage installation (ISFSI). The application consists of the following documents:

(1) A **License Application**, in which the applicant describes itself and provides general and financial information

(2) A **Safety Analysis Report**, in which the applicant describes its plans for building, operating, maintaining, and decommissioning the proposed Facility

(3) An **Emergency Plan**, in which the applicant describes its plan for resolving any emergencies that may happen during the Facility's operation

(4) A **Safeguards and Physical Security Plan** (this document is not released to the public), in which the applicant describes its plans for ensuring that the Facility and nuclear material are appropriately protected

(5) An **Environmental Report**, in which the applicant provides the information the NRC staff uses in performing its environmental review of the proposed Facility

The NRC staff documents its review and conclusions on the safety-related aspects of the license application in this Safety Evaluation Report (SER) This SER provides the staff evaluation concerning the first four documents of the Diablo Canyon License Application. This executive summary provides a brief overview and summary of the SER.

The facility that PG&E proposes to build (called the Diablo Canyon ISFSI) would store spent nuclear fuel used to generate power at the two units of the PG&E Diablo Canyon commercial nuclear power plant. The spent nuclear fuel is proposed to be stored in large metal and concrete containers called storage casks. This method of storing spent nuclear fuel is called dry-cask storage technology and is distinct from wet storage, which is a method of storing the spent nuclear fuel in large pools of water. Based on the existing inventory of spent nuclear fuel in the pools and the expected generation of spent nuclear fuel, PG&E estimates that the current wet storage at the Diablo Canyon Power Plant (DCPP) will reach capacity prior to refueling operations in 2006.

The Nuclear Waste Policy Act of 1982 mandated that the U.S. Department of Energy (DOE) assume responsibility for the permanent disposal of spent nuclear fuel. The DOE has identified Yucca Mountain, Nevada, as its proposed site for disposal of high-level waste, including spent nuclear fuel. In accordance with the current DOE schedule, the proposed Yucca Mountain repository will not be able to accept high-level waste prior to 2010, pending approval of the Yucca Mountain license application by the NRC. Thus, spent nuclear fuel will need to remain at the DCPP site until the proposed Yucca Mountain repository is operational or until another interim storage facility is in place to accept spent nuclear fuel. Therefore, PG&E proposed to use the onsite ISFSI dry-storage technology to meet its needs for additional capacity to store spent nuclear fuel beyond 2006.

According to the PG&E license application, the ISFSI is to be located on the same site as the DCPP. The Diablo Canyon site consists of 300 hectares [750 acres] of land located in San Luis Obispo County, California, near the Pacific Ocean. The site is approximately 19 km [12 mi] west-southwest of the city of San Luis Obispo, California. The proposed ISFSI Facility includes four main categories of structures, systems, and components (SSCs). These categories are the (1) dry-cask storage system, (2) storage pads, (3) onsite cask transporter, and (4) Cask Transfer Facility (CTF).

Description of the Proposed Diablo Canyon ISFSI Facility

The dry-cask storage system that PG&E proposes to use at the Diablo Canyon ISFSI Facility is the Holtec International HI-STORM 100 System. The HI-STORM 100 System is a canister-based storage system that stores spent nuclear fuel in a vertical orientation (the cask and the fuel rods inside are, in effect, standing up). The HI-STORM 100 System consists of three parts:

(1) Multi-purpose canisters (MPCs), which contain the fuel
(2) HI-STORM 100SA Overpacks, which contain the MPCs during storage
(3) HI-TRAC 125 Transfer Casks, which contain the MPCs during loading, unloading, and transfer

The MPCs provide the confinement system for the spent nuclear fuel. Each MPC is a welded, cylindrical canister in which the fuel is sealed. The HI-TRAC 125 Transfer Cask also provides radiation shielding and structural protection of the MPC during transfer operations. The transfer cask is a multiwalled cylindrical vessel with a water jacket attached to the exterior. The HI-TRAC transfer cask will be used to move the MPC from the fuel-handling building and auxiliary building at the DCPP to the CTF where they will be placed in the HI-STORM 100SA Overpacks. The storage overpacks provide radiation shielding and structural protection of the MPC during storage. The HI-STORM 100 System can be used to store either pressurized water reactor fuel assemblies or boiling water reactor fuel assemblies. The HI-STORM 100 System is a passive system that does not rely on any active cooling systems to remove spent nuclear fuel decay heat. The HI-STORM 100 System is anchored in a vertical position to the reinforced concrete storage pad.

The HI-STORM 100 System has been approved by the NRC for use under the general license provisions of 10 CFR Part 72, Subpart K (Certificate of Compliance No. 1014, Amendment 1, July 15, 2002, Docket No. 72-1014) (U.S. Nuclear Regulatory Commission, 2002b). NRC staff evaluated the cask system for general use for dry storage. This evaluation is documented in the NRC Holtec International HI-STORM 100 System Safety Evaluation Report, which was issued with the certificate of compliance, the regulatory document by which NRC allows general use of an approved storage or transportation cask. To demonstrate that the HI-STORM 100 System was acceptable for use at the Diablo Canyon ISFSI Facility, PG&E evaluated the HI-STORM 100 System against the parameters and conditions specific to the Facility. NRC staff reviewed the PG&E evaluation. As discussed in this SER, staff finds that the HI-STORM 100 System is acceptable for use at the Diablo Canyon ISFSI in accordance with the site-specific license provisions of 10 CFR Part 72.

The cask transporter is to be purchased as a commercial-grade item and qualified by testing prior to use to satisfy the single-failure-proof lift system criterion. The load-bearing components of the cask transporter are designed to prevent damage to the spent nuclear fuel and spent

nuclear fuel storage cask during transport, lifting, and MPC transfer operations during all normal, off-normal, and accident conditions. It is designed to transport the MPC in the transfer cask from the DCPP to the CTF. At the CTF, the transporter will be used to lower each MPC into an overpack. The transporter will then be used to move the loaded overpack from the CTF to the storage pad.

The CTF is a cylindrical, steel-lined structure embedded in the rock close to the storage pads. The CTF is designed to prevent damage to the spent nuclear fuel and cask system components during lifting and MPC transfer operations during normal, off-normal, and accident conditions. Operations in the CTF are performed as follows. The empty storage overpack is placed on the lift platform and lowered into the CTF using three jack screws. The cask transporter, with MPC in the HI-TRAC 125 Transfer Cask, is driven over the overpack, and the transporter and transfer cask are anchored to the surrounding rock. The cask transporter then lowers the MPC into the overpack. The lid is then placed onto the overpack and the overpack is lifted by the jack screws at the CTF. The transporter with the loaded overpack is then detached from the CTF and supports and is driven to the storage pad.

The HI-STORM 100 System will be anchored to reinforced concrete storage pads that will be built on top of bedrock. The storage pads for the Diablo Canyon ISFSI provide the necessary embedment for the anchorage of the HI-STORM 100 System casks. The pads are designed to ensure a stable and level support surface for the storage cask during normal, off-normal, and a accident conditions. Each of the seven pads at the facility is designed to hold 20 casks.

Safety of Facility

In its evaluation of the application, the NRC staff determined that PG&E showed that its proposed Facility and the HI-STORM 100 System cask design are structurally sound and will ensure that the spent fuel will remain within the cask and maintain a sound structure during all phases of operation for both normal operating conditions and accidents. PG&E included analyses of all plausible natural and man-made phenomena, many of which have already been accepted by the U.S. Nuclear Regulatory Commission (2002a,b). The regulations at 10 CFR §72.40(c) indicate that a reevaluation of a site is not required if it is covered under previous licensing actions, except where new information is discovered that could alter the original site evaluation findings. In most cases, the staff has not discovered new information that alters the applicability of the current DCPP licensing basis. In cases where new information was found, PG&E has provided additional analysis and design information to obtain staff acceptance. After reviewing the applicant analyses, the NRC staff concluded that the PG&E Facility and HI-STORM 100 System design are structurally safe and will meet regulatory requirements.

The NRC staff also determined that PG&E has shown that the spent nuclear fuel within the storage casks will remain subcritical (that is, unable to sustain a nuclear chain reaction) during all phases of operation for both normal and credible accident conditions. PG&E provided radiation dose estimates for the surrounding public and the workers at the Facility. The HI-STORM 100 System storage canister will be welded closed to prevent leakage of radioactive material. The canister is surrounded by a thick wall of concrete and steel to shield the area outside of the cask from direct radiation during storage.

The amount of radiation to which a person is exposed is called a dose. PG&E has estimated that members of the public near the proposed Facility would receive doses below the NRC regulatory requirements, which for normal conditions of operation is 0.25 mSv/yr [25 mrem/yr] and for credible accidents is 0.05 Sv/yr [5 rem/yr]. PG&E also calculated radiation dose rates within the vicinity of individual casks to demonstrate that workers at the proposed Facility will not receive doses that exceed 0.05-Sv/yr [5 rem/yr], the NRC annual regulatory limits for workers at nuclear facilities. These radiation dose limits have been established by the NRC to prevent any undue risk and to ensure the safety of all members of the public and workers at a nuclear facility. PG&E also described its radiation protection program, which employs an as low as reasonably achievable (ALARA) radiation protection principle. The operating PG&E Facility would also monitor radiation doses received by the workers and dose rates within the vicinity of the storage pad to verify that radiation dose limits are not exceeded. The NRC staff reviewed PG&E analyses and concluded that the PG&E Facility and HI-STORM 100 System design are radiologically safe and will meet regulatory requirements.

PG&E was required to demonstrate that all of the important parts of its proposed Facility would continue to perform their designed functions during normal conditions and during any of the accidents that might reasonably be expected to occur. The NRC staff concluded that, as required by 10 CFR Part 72, PG&E has provided acceptable analyses of the design and performance of these "structures, systems, and components important to safety" under credible, off-normal and accident scenarios. Based on its evaluation of these events, the staff concluded that they do not pose a credible hazard to the Facility.

The staff further concluded that the PG&E analyses of off-normal and accident events demonstrate that the proposed Facility will be sited, designed, constructed, and operated so that during all credible off-normal and accident events, public health and safety will be adequately protected and the capability to retrieve fuel from the Facility will be preserved.

Other Requirements

To demonstrate its financial qualifications, PG&E identified anticipated sources of funds to construct its Facility. Appropriate license conditions have been developed and stated in this SER providing reasonable assurance of the applicant's financial qualifications.

The NRC staff also found the PG&E emergency plan and safeguards and physical security plans to be acceptable. The emergency plan appropriately described the PG&E program for responding to onsite emergencies. It also described plans for seeking offsite assistance, if needed. The safeguards and physical protection plan were also found to meet NRC requirements.

References

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report.* Amendment 1. Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2002a.

U.S. Nuclear Regulatory Commission. *Certificate of Compliance No. 1014, Amendment No 1, for the HI-STORM Cask System.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2002b.

SAFETY EVALUATION REPORT CONCERNING THE DIABLO CANYON INDEPENDENT SPENT FUEL STORAGE INSTALLATION

INTRODUCTION

On December 21, 2001, the Pacific Gas and Electric Company (PG&E) submitted a license application in accordance with 10 CFR Part 72 to the NRC to construct and operate an onsite independent spent nuclear fuel storage installation (ISFSI). The onsite ISFSI Facility will store spent nuclear fuel and associated nonfuel hardware from Units 1 and 2 of the Diablo Canyon Power Plant (DCPP). The proposed Diablo Canyon ISFSI will use MPCs placed inside overpacks of the HI-STORM 100 System to store the spent nuclear fuel. The HI-STORM 100 System is anchored in a vertical position on the storage pad.

In support of its application, PG&E submitted the following documents, which contain the information specified in 10 CFR Part 72, Subpart B, License Application, Form, and Contents:

(1) The License Application, which contains
- General and financial information required by 10 CFR §72.22
- Proposed technical specifications required by 10 CFR §72.26
- Applicant's technical qualifications required by 10 CFR §72.28
- Preliminary decommissioning plan required by 10 CFR §72.30
- Emergency Plan required by 10 CFR §72.32.

(2) The Safety Analysis Report (SAR) for the Diablo Canyon ISFSI Facility required by 10 CFR §72.24.

(3) The Environmental Report for the Diablo Canyon ISFSI Facility required by 10 CFR §72.34.

(4) The Security Plan for the Diablo Canyon ISFSI Facility, which includes the safeguards contingency plan, as required by 10 CFR §72.180 and §72.184.

This report documents the results of the safety evaluation review conducted by the staff. Documents reviewed by staff include (i) the Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2002), (ii) associated calculation packages, (iii) responses to staff requests for additional information, and (iv) other supporting documentation. The technical review was carried out according to the applicable NRC regulations in 10 CFR Parts 20 and Part 72 and was supported by NUREG–1567, Standard Review Plan for Spent Fuel Dry Storage Facilities (U.S. Nuclear Regulatory Commission, 2000a); NUREG–1536, Standard Review Plan for Dry Cask Storage Systems (U.S. Nuclear Regulatory Commission, 1997); NUREG–0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (U.S. Nuclear Regulatory Commission, 1987); Regulatory Guide 3.61, Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask, Revision 1 (U.S. Nuclear Regulatory Commission, 1989a); and Regulatory Guide 3.62, Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Nuclear Fuel Storage Casks, Revision 0 (U.S. Nuclear Regulatory Commission, 1989b).

This SER documents the staff review of the design, operation, and other safety aspects of the proposed Diablo Canyon ISFSI Facility, as described in the above submittals, except for the

Environmental Report. The Environmental Report is the subject of a separate Environmental Impact Statement.

The staff assessment in this SER is based on the regulatory requirements of 10 CFR Part 72. In its review, staff evaluated the (1) characteristics of the site, (2) facility operations and operating systems, (3) design and design criteria for the facility and its SSC important to safety, (4) programs that support protection of public health and safety and worker health and safety, (5) impact of potential off-normal and accident events on SSC important to safety, (6) financial qualifications of the applicant, and (7) proposed technical specifications.

The applicant has identified the HI-STORM 100 System as the dry-cask storage system that will be used at the ISFSI Facility. The HI-STORM 100 System is a canister-based storage system that stores spent nuclear fuel in a vertical orientation. It consists of three discrete components: the MPC, the HI-TRAC 125 Transfer Cask, and the HI-STORM 100 System storage overpack. The MPC is the confinement system for the stored fuel. The HI-TRAC 125 Transfer Cask provides radiation shielding and structural protection of the MPC during storage. The HI-STORM 100 System is passive and does not rely on any active cooling systems to remove spent nuclear fuel decay heat. The Diablo Canyon ISFSI Facility will use the currently approved HI-STORM 100 System with a modified cask anchoring system to be approved as part of this site-specific license. In evaluating the use of this cask at the ISFSI Facility, staff reviewed the HI-STORM 100 System Final Safety Analysis Report (Holtec International, 2000) and the related NRC SER (U.S. Nuclear Regulatory Commission, 2000b,c) to determine whether or not the ISFSI Facility site parameters are enveloped by the cask design parameters considered in those reports and whether the HI-STORM 100 System is acceptable for use at the Diablo Canyon ISFSI Facility site. Staff also verified that the ISFSI Facility cask storage pads and areas are designed to support the static load of the stored cask adequately and that the radiological limits of 10 CFR §72.104 are met.

REFERENCES

Holtec International. *Final Safety Analysis Report for the HI-STORM 100 Cask System.* Docket No. 72-1014. Marlton, NJ: Holtec International. August 2000.

Pacific Gas and Electric Company. *Diablo Canyon ISFSI Safety Analysis Report.* Amendment 1. Docket No. 72-26. Avila Beach, CA: Pacific Gas and Electric Company. October 2002.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.* NUREG–0800. Washington, DC: U.S. Nuclear Regulatory Commission. 1987.

U.S. Nuclear Regulatory Commission. *Standard Format and Content for a Topical Safety Analysis Report for Spent Fuel Storage Casks.* Regulatory Guide 3.61. Washington, DC: U.S. Nuclear Regulatory Commission. 1989a.

U.S. Nuclear Regulatory Commission. *Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Nuclear Fuel Storage Casks.* Regulatory Guide 3.62. Revision 0. Washington, DC: U.S. Nuclear Regulatory Commission. 1989b.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Dry Cask Storage Systems.* Final Report. NUREG–1536. Washington, DC: U.S. Nuclear Regulatory Commission. 1997.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Spent Fuel Dry Storage Facilities.* Final Report. NUREG–1567. Washington, DC: U.S. Nuclear Regulatory Commission. 2000a.

U.S. Nuclear Regulatory Commission. *Certificate of Compliance No. 1014, Amendment No. 0.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2000b.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2000c.

1 GENERAL DESCRIPTION

1.1 Conduct of Review

On December 21, 2001, the Pacific Gas and Electric (PG&E) Company submitted a license application in accordance with 10 CFR Part 72 to the U.S. Nuclear Regulatory Commission (NRC) to construct and operate an onsite independent spent fuel storage installation (ISFSI) (Pacific Gas and Electric Company, 2001a). Included in the PG&E license application is the Safety Analysis Report (SAR). Amendment 1 of the SAR (Pacific Gas and Electric Company, 2002) was submitted in October 2002 and incorporates responses to staff requests for additional information.[1,2] Staff review of the SAR as documented in this Safety Evaluation Report (SER) is based on the information provided through Amendment 1 of the SAR, including responses to the staff requests for additional Information.

Chapter 1 of the SAR for the Diablo Canyon ISFSI explains the need for the Diablo Canyon ISFSI and provides a general description of the site, the major components and operations of the ISFSI, and the colocated Diablo Canyon Power Plant (DCPP). The objective of this chapter of the SAR is to familiarize the reader with the pertinent features of the installation.

1.1.1 Introduction

The proposed Diablo Canyon Facility is to be an ISFSI that will use dry-cask storage technology. In accordance with 10 CFR §72.42, the ISFSI will be initially licensed for 20 years. Before the end of this license term, the applicant may submit an application to renew the license. All spent nuclear fuel will be transferred offsite, and the ISFSI will be ready for decommissioning, after a permanent repository is operational.

The Diablo Canyon ISFSI will be colocated with the DCPP on PG&E-owned property, which is located on the California coast approximately 10 km [6 mi] northwest of Avila Beach, California. The DCPP consists of two nuclear-generating stations having spent nuclear fuel pools that can store spent nuclear fuel generated from refueling operations until 2006. The Diablo Canyon ISFSI will provide additional spent nuclear fuel storage capacity to DCPP beyond 2006, when the current wet pool storage will reach full capacity. Where possible, the Diablo Canyon ISFSI SAR utilizes site-specific information previously presented in the DCPP Final Safety Analysis Report (FSAR). The DCPP FSAR was accepted by the NRC when it granted the operational license of the DCPP in accordance with 10 CFR Part 50. The DCPP FSAR, which is maintained through periodic revisions in accordance with 10 CFR §50.71(e), was last updated in 2001 (Pacific Gas and Electric Company, 2001b).

---

[1]Hall, J.R. *Request for Additional Information for the Diablo Canyon Independent Spent Fuel Storage Installation Application.* Letter (August 29) to L.F. Womack, Diablo Canyon Power Plant. Washington, DC: U.S. Nuclear Regulatory Commission. 2002.

[2]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Spent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

The two reactors of DCPP share a common fuel-handling building and auxiliary building as well as components of auxiliary systems. Each reactor has a dedicated fuel-handling system and spent nuclear fuel pool. Both reactors share a single 125-ton-[113,398-kg]-capacity crane for fuel-handling activities. Each reactor core contains 193 fuel assemblies, and both units are currently operating on 18 to 21-month refueling cycles. Typically, 76 to 96 spent nuclear fuel assemblies are permanently discharged from each unit during a refueling.

The Diablo Canyon ISFSI will consist of the storage pads, a Cask Transfer Facility (CTF), an onsite cask transporter, and the dry-cask storage system. The Diablo Canyon ISFSI will use the currently approved HI-STORM 100 System with a modified cask anchoring system to be approved as part of this site-specific license.

The Diablo Canyon ISFSI is designed to hold up to 140 storage casks. Based on the current fuel strategy and the principal use of the multi-purpose canister (MPC)–32 that contains a maximum of 32 pressurized water reactor fuel assemblies, the Diablo Canyon ISFSI with a storage pad capacity of 140 casks will be capable of storing the spent nuclear fuel generated by the two DCPP reactors during the term of their current operating licenses. In addition, to accommodate spent nuclear fuel generated during the ISFSI licensed period, as well as any damaged fuel assemblies, debris, and nonfuel hardware, PG&E may use three other MPC designs from the HI-STORM 100 System, including the MPC–24, MPC–24E, and MPC–24EF designs. All four MPC designs use the same storage overpack and are licensed by the current Certificate of Compliance No. 1014 (U.S. Nuclear Regulatory Commission, 2002) for the HI-STORM 100 System. The applicant proposes to begin construction of the Diablo Canyon ISFSI in 2003 and plans to begin operation in 2006.

1.1.2 General Description of the Location

The Diablo Canyon ISFSI will be located within the PG&E owner-controlled area at the DCPP site, which consists of approximately 300 hectares [750 acres] of land located in San Luis Obispo County, California. The area is directly southeast of Montana de Oro State park and is approximately 19 km [12 mi] west-southwest of the city of San Luis Obispo, California, the county seat and nearest significant population center. Only a few individuals reside within 8 km [5 mi] of the DCPP site. The nearest residential community is Los Osos, approximately 13 km [8 mi] north of the plant site. A number of other cities, as well as some unincorporated residential areas, exist along the coast and inland. However, these population areas are more than 13 km [8 mi] from the DCPP site.

A security fence defines the Diablo Canyon ISFSI protected area within the owner-controlled area, which is surrounded by a fence. The DCPP site is located near the mouth of Diablo Creek, and a portion of the site is bounded by the Pacific Ocean. All coastal properties north of Diablo Creek, extending north to the southerly boundary of Montana de Oro State Park and reaching inland approximately 1 km [0.5 mi] are owned by PG&E. Coastal properties south of Diablo Creek and reaching inland approximately 1 km [0.5 mi] are owned by Eureka Energy Company, a wholly owned subsidiary of PG&E. PG&E has complete authority to control all activities within the site boundary, and this authority extends to the mean high water line along the ocean. On land, there are no activities unrelated to DCPP operation within the owner-controlled area. The DCPP site is not traversed by any public highway or railroad. Normal access to the site is from the south by private road through the owner-controlled area, which is fenced and posted by PG&E.

The DCPP occupies a coastal terrace that ranges in elevation from 18 to 46 m [60 to 150 ft] above mean sea level and is approximately 300 m [1,000 ft] wide. The coastal areas surrounding the DCPP are well drained, primarily through Diablo Creek, and groundwater is at least 52 m [170 ft] below the surface of the ISFSI pad. Winter is the rainy season, more than 80 percent of the average annual rainfall of approximately 41 cm [16 in] occurs during the winter months. The average annual temperature of the site area is approximately 13 °C [55 °F], with a variation between approximately 0 °C [32 °F] minimum and 36 °C [97 °F] maximum.

Staff finds that the site and Diablo Canyon ISFSI Facility descriptions have sufficient detail to allow familiarization with the site characteristics of the proposed ISFSI.

1.1.3 General Systems Description

The major SSC of the Diablo Canyon ISFSI will include the storage pads, CTF, onsite transporter, and dry-cask storage system. The dry-cask storage system that has been identified for use at the ISFSI is the HI-STORM 100 System (the cask system). The cask system is a canister-based storage system that stores spent nuclear fuel in a vertical orientation. It consists of three discrete components: the MPC, the HI-TRAC 125 Transfer Cask, and the HI-STORM 100 System Overpack. The MPC is the confinement system for the stored fuel. The HI-TRAC 125 Transfer Cask provides radiation shielding and structural protection of the MPC during transfer operations, while the storage overpack provides radiation shielding and structural protection of the MPC during storage. The HI-STORM 100 System is passive and does not rely on any active cooling systems to remove spent nuclear fuel decay heat. As discussed in Section 1.1.1, the Diablo Canyon ISFSI will use the currently approved HI-STORM 100 System with a modified cask anchoring system to be approved as part of this site-specific license.

The spent nuclear fuel will be loaded into the MPC at the colocated DCPP. Before transfer to the ISFSI, the canister lid will be welded in place, and the canister will be drained, vacuumed dried, filled with an inert gas, sealed, and leak tested. The MPC will be then put into the HI-TRAC 125 Transfer Cask. A transporter will be used to move the HI-TRAC 125 Transfer Cask and MPC assembly from the fuel-handling building at DCPP to the CTF, which will be adjacent to the ISFSI storage pads. The transfer cask will be lifted off the transport vehicle and placed in the CTF. The MPC will be transferred from the HI-TRAC 125 Transfer Cask into the storage overpack. The storage overpack, loaded with the canister, will then be closed and moved to the storage area again using the cask transporter. The loaded overpack will be placed on a concrete pad in a vertical orientation, and then it will be anchored to the concrete storage pad.

A general description of the cask system and its operation is provided in the Diablo Canyon ISFSI SAR. A detailed description of the cask system is given in the FSAR for the HI-STORM 100 System (Holtec International, 2000) and in Holtec HI-STORM 100 System License Amendment Request 1014-1 (Holtec International, 2001). Staff finds that the description of the storage cask system to be used at the ISFSI is sufficiently detailed to allow familiarization with its design.

1.1.4 Identification of Agents and Contractors

Section 1.4 of the SAR identifies the organizations responsible for providing the licensed spent nuclear fuel storage and transfer systems and engineering, design, licensing, and operation of

the ISFSI. Holtec International is responsible for the design of the HI-STORM 100 System, transporter, design criteria for the storage pads, and CTF. PG&E has overall responsibility for engineering, site preparation, and construction of the ISFSI storage pads and CTF, using specialty contractors as necessary. The applicant is also responsible for operating the ISFSI and providing quality assurance services.

The staff finds that agents and contractors responsible for the design and operation of the installation have been adequately identified.

1.1.5 Material Incorporated by Reference

Many chapters of the SAR include a reference section that identifies documents referred to in those chapters.

Staff finds that material incorporated by reference, including topical reports and docketed material, have been appropriately identified in the SAR.

1.2 Evaluation Findings

Staff finds that the site and Diablo Canyon ISFSI facility descriptions presented in Chapter 1 of the SAR have sufficient detail to allow familiarization with the pertinent site-related features of the proposed Diablo Canyon ISFSI.

1.3 References

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System).* Volumes I and II. HI–2002444. Docket No. 72-1014. Marlton, NJ: Holtec International. 2000.

Holtec International. *License Amendment Request 1014-1, Revision 2*, July 2001, including Supplements 1 through 4 dated August 17, 2001; October 5, 2001; October 12, 2001; and October 19, 2001. Marlton, NJ: Holtec International. 2001.

Pacific Gas and Electric Company. *Diablo Canyon ISFSI Safety Analysis Report.* Docket No. 72-26. Avila Beach, CA: Pacific Gas and Electric Company. December 2001a.

Pacific Gas and Electric Company. *Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update.* Revision 14. Avila Beach, CA: Pacific Gas and Electric Company. November 2001b.

Pacific Gas and Electric Company (PG&E). *Diablo Canyon ISFSI Safety Analysis Report.* Amendment 1. Docket No. 72-26. Avila Beach, CA: Pacific Gas and Electric Company. October 2002.

U.S. Nuclear Regulatory Commission. *10 CFR Part 72 Certificate of Compliance No. 1014, Amendment 1, for the HI-STORM 100 Cask System.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. July 15, 2002.

2 SITE CHARACTERISTICS

2.1 Conduct of Review

Chapter 2 of the Safety Analysis Report (SAR) discusses the geographical location of the Diablo Canyon independent spent fuel storage installation (ISFSI) Facility and meteorological, hydrological, seismological, and geological characteristics of the site and the surrounding area. It also describes the population distribution within and around the Diablo Canyon ISFSI, land and water uses, and associated site activities. Chapter 2 of the SAR also evaluates site characteristics with regard to safety and identifies assumptions that need to be applied when evaluating safety, establishing installation design, and providing design bases in other evaluations in the SAR.

The information and analyses in SAR Chapter 2 were reviewed with respect to the applicable siting evaluation regulations in 10 CFR Part 72, Subpart E, and 10 CFR §72.122(b). Where appropriate, findings of regulatory compliance were made for the 10 CFR Part 72 requirements that are fully addressed in Chapter 2 of the SAR. Because compliance with some regulations were determined by an integrated review of several sections in Chapter 2 or other chapters within the SAR, a finding of regulatory compliance was not made in each major section unless the specific regulatory requirement was fully addressed. Findings of technical adequacy and acceptability were made for each section in Chapter 2, however, as they related to the regulatory requirements.

In addition to the information presented in the SAR (Pacific Gas and Electric Company, 2002), much of the information pertaining to the site characteristics was derived from the power plant Final Safety Analysis Report (FSAR) (Pacific Gas and Electric Company, 2001). The Diablo Canyon Power Plant (DCPP) FSAR update is maintained through periodic revisions in accordance with 10 CFR §50.71(e). Furthermore, in response to License Condition Item 2.C.(7) of the Unit 1 Operating License issued in 1980, Pacific Gas and Electric (PG&E) was required to reevaluate the seismic design for the DCPP.

The reevaluation included a 10-year study of the geologic, tectonic, and seismological characteristics of the DCPP site. That 10-year study was documented in a report titled Diablo Canyon Long-Term Seismic Program (Pacific Gas and Electric Company, 1988, 1991). The U.S. Nuclear Regulatory Commission (NRC) concluded (U.S. Nuclear Regulatory Commission,1991) that the analyses presented in the Diablo Canyon long-term seismic program (LTSP) report were sufficient to satisfy License Condition Item 2.C.(7).

2.1.1 Geography and Demography

This section contains the review of Section 2.1, "Geography and Demography," of the SAR. Subsections discussed include (i) Site Location, (ii) Site Description, (iii) Population Distribution and Trends, and (iv) Land and Water Uses. The staff reviewed the discussion on geography and demography with respect to the following regulatory requirements:

- 10 CFR §72.24(a) requires a description and safety assessment of the site on which the ISFSI is to be located, with appropriate attention to the design bases for external events. Such assessment must contain an analysis and evaluation

of the major structures, systems, and components of the ISFSI that bear on the suitability of the site when the ISFSI is operated at its design capacity. If the proposed ISFSI is to be located on the site of a nuclear power plant or other licensed facility, the potential interactions between the ISFSI and such other facility—including shared common utilities and services—must be evaluated.

- 10 CFR §72.90(a) requires site characteristics that may directly affect the safety or environmental impact of the ISFSI to be investigated and assessed.

- 10 CFR §72.90(b) requires proposed sites for the ISFSI must be examined with respect to the frequency and the severity of external natural and man-induced events that could affect the safe operation of the ISFSI.

- 10 CFR §72.90(c) requires design-basis external events must be determined for each combination of proposed site and proposed ISFSI design.

- 10 CFR §72.90(d) requires that proposed sites with design basis external events for which adequate protection cannot be provided through ISFSI design shall be deemed unsuitable for the location of the ISFSI.

- 10 CFR §72.90(e) requires that pursuant to Subpart A of Part 51 of Title 10 for each proposed site for an ISFSI, the potential for radiological and other environmental impacts on the region must be evaluated with due consideration of the characteristics of the population, including its distribution, and of the regional environs, including its historical and aesthetic values.

- 10 CFR §72.90(f) requires the facility to be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of floodplains.

- 10 CFR §72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI must be identified.

- 10 CFR §72.98(c) requires that those regions identified pursuant to paragraph 10 CFR §72.98(a) and (b) must be investigated as appropriate with respect to (1) the present and future character and the distribution of the population, (2) consideration of present and projected future uses of land and water within the region, and (3) any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR §72.100(a) requires that the proposed site must be evaluated with respect to the effects on populations in the region resulting from the release of radioactive materials under normal and accident conditions during operation and decommissioning of the ISFSI; in this evaluation, both usual and unusual regional and site characteristics shall be taken into account.

2.1.1.1 Site Location

Section 2.1.1 of the SAR, "Site Location," and relevant literature cited in the SAR describe the site location. The Diablo Canyon ISFSI Facility is to be located within the PG&E-owned area at the DCPP.

The DCPP site consists of 300 hectares [750 acres] of land located along the Pacific Ocean in San Luis Obispo County, California. The site is approximately 19 km [12 mi] west-southwest of the city of San Luis Obispo, California, and approximately 10 km [6 mi] northwest of Avila Beach, California. The SAR reports that the Diablo Canyon ISFSI is to be located at latitude 35°12'52″ N and longitude of 120°51'00″ W, with Universal Transverse Mercator coordinates of 695,698 easting and 3,898,723 northing, Zone 10.

The staff reviewed the description of the site location and found it acceptable because it clearly describes the geographic location of the site, including its relationship to political boundaries and natural anthropogenic features. The maps provided in the SAR are acceptable because they provide sufficient detail to review the Diablo Canyon ISFSI Facility. This information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements in 10 CFR §72.24(a), §72.90(a), §72.90(e), and §72.98(a) with respect to this topic.

2.1.1.2 Site Description

Section 2.1.2 of the SAR, "Site Description," describes the site with maps to delineate the site boundary and controlled area. The Diablo Canyon ISFSI Facility site is to be located within the PG&E-owned area at the DCPP.

The location and orientation of the Diablo Canyon ISFSI Facility structures with respect to nearby roads and waterways are shown on various maps and plots, and there is no obvious way in which traffic on adjacent transportation links can interfere with ISFSI Facility operations. The proposed Diablo Canyon ISFSI site is located within the site boundaries of the existing PG&E DCPP site. PG&E owns the land along the coast, north 9.5 km [5.9 mi] to the Montana de Oro State Park and south approximately 13 km [8 mi] to Avila Beach and inland between 0.8 km [0.5 mi] and 2.9 km [1.8 mi]. The area surrounding the DCPP site is encumbered by grazing leases. Access to the ISFSI and the DCPP-controlled areas is restricted by fencing. The PG&E owner-controlled area is not traversed by any public roads or railroads. Ingress and egress of site personnel will be controlled. Access to the site is from the south, at Avila Beach, along a 10.5-km [6.5-mi] private road through the PG&E-owned site.

The DCPP owner-controlled area rests on a coastal terrace and adjacent coastal uplands. Elevations of the DCPP site range from 18 to 425 m [60 to 1,400 ft] above mean sea level (MSL). The ISFSI site is located on bedrock between hillsides at an elevation of 94.5 m [310 ft]. Drainage of the area is through Diablo Creek, located just north of the ISFSI pad site.

The staff reviewed the site description and relevant literature cited in the SAR. The staff finds that the site description is adequate because the descriptive information and maps clearly delineate the site boundary, controlled area, general natural and man-made features, topography, and surface hydrologic features. The maps have a sufficient level of detail and are

of appropriate scale and legibility required for the review of the site and the Diablo Canyon ISFSI Facility. The information is also acceptable to determine distances between the ISFSI Facility and nearby facilities and cities. This information is acceptable for use in other sections of the SAR to develop the design bases of the ISFSI Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR §72.90(a), §72.90(e), and §72.98(a) with respect to this topic.

2.1.1.3 Population Distribution and Trends

Section 2.1.3 of the SAR, "Population Distribution and Trends," and relevant literature cited in the SAR describes the population distribution and trends. The population data used in the SAR were derived from the 2000 census and from estimates of future population provided by the California Department of Finance. The population distribution and trends description in the SAR based on information from the DCPP FSAR update.

The 2000 census data show the population within 80.5 km [50 mi] of the ISFSI site to be 424,000 people. The same census data show that 23,700 people live within a 16-km [10-mi] radius of the ISFSI site. The population within a 9.6-km [6-mi] radius of the site, defined by PG&E as the low population zone, is approximately 100 residents. The nearest residence is approximately 2.4 km [1.5 mi] from the ISFSI site. Information in the SAR indicates that the maximum number of people in the low population zone is 5,000. This population estimate is based on the maximum number of day-use visitors to the Montana de Oro State Park, based on information from the California Department of Parks and Recreation.

PG&E indicates that projected population growth near the ISFSI Facility will be limited based on projected trends in the population data and on the assumption that the land use will not change in character within the next 25 years. This assumption is supported by information on land and water uses, as described in Section 2.1.4 of the SAR. Land use is expected to remain either agricultural or as wilderness within state parks or national forests.

The staff reviewed the information presented in the SAR and has determined that the population distribution and trends in the region have been adequately described and assessed. The source of the population data used in the SAR is appropriate, and the basis for population projections is reasonable. The staff found that 10 CFR §72.98(c)(1) is met because the region has been appropriately investigated with respect to the present and future character and distribution of the population. This information is also acceptable for use in other sections of the SAR to develop the design bases of the ISFSI Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR §72.90(e), §72.98(a), and §72.100(a), with respect to this topic.

2.1.1.4 Land and Water Uses

Section 2.1.4 of the SAR, "Uses of Nearby Lands and Waters," describes land and water uses in the region surrounding the DCPP site. The SAR notes that the area between the DCPP site and U.S. Route 101 is dominated by the San Luis Range, which consists of rugged topography with elevation up to 550 m [1,800 ft]. Land use in the San Luis Range is limited to beef cattle grazing, minor dairy cattle grazing, wild and domestic goats, and support of other wildlife. A large portion of this land area is contained within the Los Padres National Forest.

Farming within surrounding regions of San Luis Obispo County is limited to the narrow coastal valleys. Wine grapes are the principal cash crop. The only dairy production is at the California Polytechnic State University, approximately 19 km [12 mi] northeast of the Diablo Canyon ISFSI site. The region also supports sport and commercial deep-sea fishing. Fishing vessels harbor at Moor Bay, located 16 km [10 mi] north of the Diablo Canyon ISFSI site, and at Port San Luis Harbor, located 10 km [6 mi] south of the Diablo Canyon ISFSI site.

PG&E has identified two public groundwater suppliers within 16 km [10 mi] of the site. These suppliers are the Avila Beach County Water and Sewer District and the San Miguelito Mutual Water and Sewer Company. Property owners north and east of the DCPP also capture surface and spring water for limited domestic use. PG&E captures surface water from Crowbar Canyon, located 1.6 km [1 mi] north of the DCPP. DCPP also gets water from an ocean water desalinization plant, which was built at the DCPP in 1985.

The staff reviewed the description of the land and water use in the SAR and found that it has been adequately described and assessed. The region has been investigated as appropriate with respect to consideration of present and projected future uses of land and water within the region. This information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR §72.98(a), §72.98(b), and §72.98(c) with respect to this topic.

2.1.2 Nearby Industrial, Transportation, and Military Facilities

Section 2.2 of the SAR, "Nearby Industrial, Transportation, and Military Facilities," and relevant literature cited in the SAR describe each nearby facility and identify potential hazards from all these facilities. This information is necessary to evaluate credible scenarios involving manmade facilities that may endanger the proposed facility site. The staff reviewed nearby industrial, transportation, and military facilities with respect to the following regulatory requirements:

- 10 CFR §72.24(a) requires a description and safety assessment of the site on which the ISFSI is to be located, with appropriate attention to the design bases for external events. Such assessment must contain an analysis and evaluation of the major structures, systems, and components of the ISFSI that bear on the suitability of the site when the ISFSI is operated at its design capacity. If the proposed ISFSI is to be located on the site of a nuclear power plant or other licensed facility, the potential interactions between the ISFSI and such other facility—including shared common utilities and services—must be evaluated.

- 10 CFR §72.94(a) requires that the region must be examined for both past and present man-made facilities and activities that might endanger the proposed ISFSI. The important potential man-induced events that affect the ISFSI design must be identified.

- 10 CFR §72.94(b) requires that information concerning the potential occurrence and severity of such events must be collected and evaluated for reliability, accuracy, and completeness.

- 10 CFR §72.94(c) requires that appropriate methods must be adopted for evaluating the design-basis external man-induced events, based on the current state of knowledge about such events.

- 10 CFR §72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI must be identified.

- 10 CFR §72.98(b) requires that the potential regional impact caused by the construction, operation, or decommissioning of the ISFSI must be identified. The extent of regional impacts must be determined on the basis of potential measurable effects on the population or the environment from ISFSI activities.

- 10 CFR §72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR §72.98(a) and 10 CFR §72.98(b) must be investigated as appropriate with respect to (1) the present and future character and the distribution of population, (2) consideration of present and projected future uses of land and water within the region, and (3) any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR §72.100(a) requires that the proposed site must be evaluated with respect to the effects on populations in the region resulting from the release of radioactive materials under normal and accident conditions during operation and decommissioning of the ISFSI; in this evaluation both usual and unusual regional and site characteristics shall be taken into account.

- 10 CFR §72.100(b) requires that each site must be evaluated with respect to the effects on the regional environment resulting from construction, operation, and decommissioning of the ISFSI; in this evaluation, both usual and unusual regional and site characteristics must be taken into account.

The identification of potential hazards includes identification of facilities and determination of credible scenarios that may endanger the proposed Facility. Industry in the vicinity of the Diablo Canyon ISFSI mainly consists of food processing and crude oil refining (Pacific Gas and Electric Company, 2002). Vandenberg Air Force Base is the largest industrial complex near the proposed facility and is located approximately 56 km [35 mi] south-southwest of the proposed Facility. This base is the site for launching missiles and polar-orbiting satellites and also the designated alternate landing site for space shuttles. Coastal shipping lanes are approximately 32 km [20 mi] offshore. The Port San Luis tanker loading pier is approximately 9.6 km [6 mi] east-southeast of the proposed site. This pier is no longer active because tanker traffic into Port San Luis has been discontinued (Pacific Gas and Electric Company, 2002). The local tanker terminal at Estero Bay ceased operations in 1994 because of lack of tanker traffic. For the same reason, the Avila Beach pier closed in 1998. Some petroleum products and crude oil are stored at Estero Bay, which is approximately 16 km [10 mi] from the DCPP site. This storage site is sufficiently far from the site to have a negligible impact and thus is beyond the distance of regulatory concern.

PG&E identified the potential crash of aircraft onto the proposed site as a potential hazard (Pacific Gas and Electric Company, 2002). Civilian aircraft with a potential to crash at the proposed Facility include aircraft taking off and landing at San Luis Obispo Regional Airport and Oceano Airport, and aircraft flying on federal airway V-27. Military aircraft that have a potential to crash at the proposed Facility include aircraft flying the military training route VR-249 (Pacific Gas and Electric Company, 2002). Additionally, PG&E considered hazards associated with launch and landing of the space shuttle, rockets, and missiles at Vandenberg Air Force Base.

The staff finds that all nearby facilities that may present a hazard to the proposed Facility have been adequately identified and demonstrate compliance with regulatory requirements in 10 CFR §72.24(a). The potential hazards from these facilities are evaluated in Chapter 15 of this SER.

2.1.3 Meteorology

The staff has reviewed the information presented in Section 2.3 of the SAR, "Meteorology." Subsections discussed below include (i) Regional Climatology, (ii) Local Meteorology, and (3) Onsite Meteorological Measurement Program. The staff reviewed the discussion on meteorology with respect to the following regulatory requirements:

- 10 CFR §72.24(a) requires a description and safety assessment of the site on which the ISFSI is to be located, with appropriate attention to the design bases for external events. Such assessment must contain an analysis and evaluation of the major structures, systems, and components of the ISFSI that bear on the suitability of the site when the ISFSI is operated at its design capacity. If the proposed ISFSI is to be located on the site of a nuclear power plant or other licensed facility, the potential interactions between the ISFSI and such other facility—including shared common utilities and services—must be evaluated.

- 10 CFR §72.90(a) requires site characteristics that may directly affect the safety or environmental impact of the ISFSI be investigated and assessed.

- 10 CFR §72.90(b) requires proposed sites for the ISFSI to be examined with respect to the frequency and severity of external natural and man-induced events that could affect the safe operation of the ISFSI.

- 10 CFR §72.90(c) requires design-basis external events to be determined for each combination of proposed site and proposed ISFSI design.

- 10 CFR §72.90(d) requires that proposed sites with design-basis external events for which adequate protection cannot be provided through ISFSI design shall be deemed unsuitable for the location of the ISFSI.

- 10 CFR §72.90(e) requires that, pursuant to Subpart A of Part 51 of Title 10, for each proposed site for an ISFSI, the potential for radiological and other environmental impacts on the region must be evaluated with due consideration of the characteristics of the population, including its distribution, and of the regional environs, including its historical and aesthetic values.

- 10 CFR §72.90(f) requires the facility to be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of floodplains.

- 10 CFR §72.92(a) requires that natural phenomena that may exist or that can occur in the region of a proposed site be identified and assessed according to their potential effects on the safe operation of the ISFSI. The important natural phenomena that affect the ISFSI design must be identified.

- 10 CFR §72.92(b) requires that records of the occurrence and severity of those important natural phenomena must be collected for the region and evaluated for reliability, accuracy, and completeness. The applicant shall retain these records until the license is issued.

- 10 CFR §72.92(c) requires that appropriate methods must be adopted for evaluating the design-basis external natural events based on the characteristics of the region and the current state of knowledge about such events.

- 10 CFR §72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI must be identified.

- 10 CFR §72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR §72.98(a) and (b) be investigated as appropriate with respect to (1) the present and future character and the distribution of population, (2) consideration of present and projected future uses of land and water within the region, and (3) any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR 72.122(b) requires that (1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. (2)(i) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect (A) appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (B) appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (2)(ii) The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent nuclear fuel or high-level waste or onto structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety. (4) If the ISFSI is located over an

aquifer that is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

2.1.3.1 Regional Climatology

Section 2.3.1, "Regional Climatology," in the SAR briefly describes the regional climatology for the region surrounding the DCPP. The description is based on a summary of regional climatology from Section 2.3 of the DCPP FSAR update. Because the Diablo Canyon ISFSI site is 0.35 km [0.22 mi] from the DCPP, the regional climatic conditions at the ISFSI site are the same as those at the DCPP.

Climate at the DCPP is characterized by moderate temperatures and precipitation, with small diurnal and seasonal changes. Dry conditions are prevalent from May through September. The rainy season is from October through April. The mean annual temperature at San Luis Obispo is 13 °C [55 °F], and the average yearly precipitation is approximately 56 cm [22 in].

The staff reviewed the description of the regional climate in the SAR and found that it has been adequately described and assessed. The staff review found the description of the regional climate in the SAR to be acceptable because it is based largely on general information provided for DCPP, information that is also applicable to the ISFSI site. The regulations at 10 CFR §72.40(c) indicate that a reevaluation of a site is not required when the regional climatology is covered under previous licensing actions. The staff has previously accepted the regional climatology description for DCPP, and the staff has discovered no new information that might alter the relevance of the regional climatology description for the proposed ISFSI location. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.90(a), §72.90(b), and §72.122(b) with respect to this topic.

2.1.3.2 Local Meteorology

Section 2.3.2 of the SAR, "Local Meteorology," in the SAR describes and characterizes the local meteorology of the DCPP site. The description is based on a summary of local meteorological information from Section 2.3 of the DCPP FSAR update. Because the Diablo Canyon ISFSI site is 0.35 km [0.22 mi] from the DCPP, the local meteorological conditions at the ISFSI site are considered to be the same as those at the DCPP.

The SAR indicates that the maximum recorded temperature for the site was 36 °C [97 °F] in October 1997 and that the minimum recorded temperature was just below 0 °C [32 °F] in December 1990. The mean annual temperature is approximately 13 °C [55 °F], based on measurement from the DCPP primary meteorological tower.

Solar insolation data collected by the California Polytechnic State University, Department of Water Resources, and cataloged in the California Irrigation Management Information System, are applicable to the Diablo Canyon ISFSI Facility site. These data are measured at a location approximately 19.3 km [12 mi] northeast of the proposed Facility. For data collected between 1986 and 1999, the maximum measured insolation for a 24-hour period was 766 g-cal/cm^2 per day {371 W/m^2 [118 BTU/hr-ft^2]} and, for a 12-hour period, 754 g-cal/cm^2 per day {365 W/m^2

[116 BTU/hr-ft^2]}. The daily (24-hour) average for the period of record was 430 g-cal/cm^2 per day {208 W/m^2 [66 BTU/hr-ft^2]}.

The SAR also notes that the average annual precipitation at the DCPP is 40 cm [16 in]. The maximum daily precipitation is 8.33 cm [3.28 in]. The maximum yearly precipitation, recorded at San Luis Obispo, was 138.5 cm [54.53 in] in 1969. The highest wind gust recorded at the meteorological tower was 135 km/hr [84 mph]. The prevailing wind direction is from the northwest.

The staff reviewed the description of the local meteorology in the SAR and found that it has been adequately described and assessed. The staff review found the description of the local meteorology in the SAR to be acceptable because it is based largely on general information provided for DCPP, information that is also applicable to the ISFSI site. The regulations at 10 CFR §72.40(c) indicate that a reevaluation of a site is not required when the local meteorology is covered under previous licensing actions. The staff has previously accepted the local meteorology description for DCPP, and the staff has discovered no new information that might alter the relevance of the local meteorology description for the proposed ISFSI location. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR 72.92(a), 72.98(a), 72.98(c)(3), and 72.122(b) with respect to this topic.

2.1.3.3 Onsite Meteorological Measurement Program

Section 2.3.3 of the SAR, "Onsite Meteorological Measurement Program," describes the onsite meteorological measurement program at the DCPP. PG&E noted that the onsite meteorological monitoring system supporting the DCPP will also be used to support the proposed ISFSI Facility. The system consists of two independent subsystems and is designed to conform with NRC Regulatory Guide 1.23 (U.S. Nuclear Regulatory Commission, 1976).

The staff reviewed the description of the onsite meteorological measurement program in the SAR and found that it has been adequately described and assessed. The staff review found the description of the onsite meteorological measurement program in the SAR to be acceptable because it is based largely on general information provided for DCPP, information that is also applicable to the ISFSI site. The regulations at 10 CFR §72.40(c) indicate that a reevaluation of a site is not required when the onsite meteorological measurement program is covered under previous licensing actions. The staff has previously accepted the onsite meteorological measurement program description for DCPP, and the staff has discovered no new information that might alter the relevance of the onsite meteorological measurement program description for the proposed ISFSI location. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.24(a), §72.92(a), §72.98(a), §72.98(c)(3), and §72.122(b) with respect to this topic.

2.1.4 Surface Hydrology

The staff has reviewed the information presented in Section 2.4 of the SAR, "Surface Hydrology." Subsections discussed in the following text include (1) Hydrologic Description,

(2) Floods, (3) Probable Maximum Flood (PMF) on Streams and Rivers, (4) Potential Dam Failures (seismically induced), (5) Probable Maximum Surge and Seiche Flooding, (6) Probable Maximum Tsunami Flooding, (7) Ice Flooding, (8) Flood Protection Requirements, and (9) Environmental Acceptance of Effluents. The staff reviewed the discussion on surface hydrology with respect to the following regulatory requirements:

- 10 CFR §72.24(a) requires a description and safety assessment of the site on which the ISFSI is to be located, with appropriate attention to the design bases for external events. Such assessment must contain an analysis and evaluation of the major structures, systems, and components of the ISFSI that bear on the suitability of the site when the ISFSI is operated at its design capacity. If the proposed ISFSI is to be located on the site of a nuclear power plant or other licensed facility, the potential interactions between the ISFSI and such other facility—including shared common utilities and services—must be evaluated.

- 10 CFR §72.40(a) requires that except as provided in paragraph (c) of this section, the Commission will issue a license under this part upon a determination that the application for a license meets the standards and requirements of the Act and the regulations of the Commission

- 10 CFR §72.90(a) requires that site characteristics that may directly affect the safety or environmental impact of the ISFSI be investigated and assessed.

- 10 CFR §72.90(b) requires that proposed sites for the ISFSI must be examined with respect to the frequency and severity of external natural and man-induced events that could affect the safe operation of the ISFSI.

- 10 CFR §72.90(c) requires that design-basis external events must be determined for each combination of proposed site and proposed ISFSI design.

- 10 CFR §72.90(d) requires that proposed sites with design-basis external events for which adequate protection cannot be provided through ISFSI design shall be deemed unsuitable for the location of the ISFSI.

- 10 CFR §72.90(e) requires that, pursuant to subpart A of Part 51 of CFR Title 10, for each proposed site for an ISFSI, the potential for radiological and other environmental impacts on the region must be evaluated with due consideration of the characteristics of the population, including its distribution, and of the regional environs, including its historical and aesthetic values.

- 10 CFR §72.90(f) requires the facility must be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of flood plains.

- 10 CFR §72.92(a) requires that natural phenomena that may exist or that can occur in the region of a proposed site must be identified and assessed according to their potential effects on the safe operation of the ISFSI. The important natural phenomena that affect the ISFSI design must be identified.

- 10 CFR §72.92(b) requires that records of the occurrence and severity of those important natural phenomena must be collected for the region and evaluated for reliability, accuracy, and completeness. The applicant shall retain these records until the license is issued.

- 10 CFR §72.92(c) requires that appropriate methods must be adopted for evaluating the design-basis external natural events based on the characteristics of the region and the current state of knowledge about such events.

- 10 CFR §72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI must be identified.

- 10 CFR §72.98(c) requires that those regions identified pursuant to 10 CFR §72.98(a) and 10 CFR §72.98(b) must be investigated as appropriate with respect to (1) the present and future character and the distribution of population, (2) consideration of present and projected future uses of land and water within the region, and (3) any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR 72.122(b) requires that (1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. (2)(i) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect (A) appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (B) appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (2)(ii) The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent nuclear fuel or high-level waste or onto structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety. (4) If the ISFSI is located over an aquifer that is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

2.1.4.1 Hydrologic Description

Section 2.4.1 of the SAR, "Hydrologic Description," describes and characterizes the surface hydrological conditions and features pertaining to the proposed ISFSI site. The applicant

provides a general description and maps of the drainage basin in the region surrounding the proposed ISFSI site, summarizes the seasonal flow characteristics for Diablo Creek, and indicates sources of water supply for DCPP. The applicant documents similar hydrologic information pertaining to the Diablo Canyon area in the DCPP FSAR update, and the applicant indicates that much of that information pertains also to the ISFSI location, because the hydrologic characteristics in the Diablo Canyon area do not vary significantly in the general vicinity of the ISFSI and power plant facilities.

The surface hydrologic conditions most relevant to the proposed ISFSI include those influencing storm run-offs within the Diablo Creek watershed and those influencing wave run-up caused by disturbances in the water level of the Pacific Ocean. The area encompassing the drainage basin for Diablo Creek is approximately 13 km^2 [5 mi^2]. The drainage basin is elongated roughly east-west, with highest elevation at 555 m [1,819 ft] above MSL. The maximum flow in Diablo Creek during the flood of January 18–25, 1969, has been estimated at 12,176 l/s [430 cfs]. Typical flows are described as being nearer the minimum flow value of 12.5 l/s [0.44 cfs].

The staff reviewed information concerning the surface hydrologic features of the Diablo Canyon vicinity, including information on the location, size, and flow rates of streams and other significant water features; topographic maps and watershed characteristics; relevant oceanographic and tidal data; proposed alterations of natural drainage features; and other applicable information. The staff also reviewed meteorological data significant to the characteristics and hazards associated with surface hydrology.

The staff reviewed the description of the surface hydrologic features in the SAR and found that these features have been adequately described and assessed. The staff review found the description of the surface hydrologic features in the SAR to be acceptable because it is based largely on general information provided for DCPP, information that is also applicable to the ISFSI site. The regulations at 10 CFR §72.40(c) indicate that a reevaluation of a site is not required when the surface hydrologic features are covered under previous licensing actions. The staff has previously accepted the surface hydrologic features description for DCPP, and the staff has discovered no new information that might alter the relevance of the surface hydrologic features description for the proposed ISFSI location. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.24(a), §72.90(a), §72.90(b), §72.90(c), §72.90(d), §72.90(f), §72.92(a), §72.92(b), §72.92(c), §72.98(a), §72.98(c)(3), and §72.122(b) with respect to this topic.

2.1.4.2 Floods

Section 2.4.2 of the SAR, "Floods," discusses the potential for and effects of floods at the proposed ISFSI site. The applicant identifies flood considerations from the DCPP FSAR update that are applicable to the proposed ISFSI location. The applicant flood design considerations pertain primarily to local floods along Diablo Creek. PG&E states that the canyon confining Diablo Creek is more than sufficient to channel any conceivable flood without hazard to the proposed ISFSI. Additionally, any damming of water caused by channel blockage resulting from a landslide downstream of the ISFSI would not be sufficient to flood the ISFSI area, and there are no dams or natural features along Diablo Creek that would hinder run-off for a significant period of time. The applicant also indicates that run-off is efficiently drained at the proposed

ISFSI site by the adjacent natural and constructed drainage features. Ponding caused by intense local precipitation can occur at the proposed ISFSI site, but the applicant indicates that such ponding would not be significant, would be temporary, and would not adversely impact ISFSI operation or spent nuclear fuel confinement. Similarly, the applicant notes that overflow of nearby constructed reservoirs would drain toward Diablo Creek and the Pacific Ocean and would not adversely impact the ISFSI.

The staff review of this information focused on the potential for and effects of locally intense precipitation at the proposed ISFSI. (Section 2.1.4.3 discusses the staff review of information concerning the potential and effects of flooding along streams and rivers.) Values for the probable maximum precipitation (PMP) for various time periods are provided in Table 2.4-1 of the DCPP FSAR update, with the 1-, 6-, and 24-hour PMPs reported as 10.9 cm [4.3 in], 23.1 cm [9.1 in], and 42.16 cm [16.6 in]. The staff also considered local and regional precipitation data, and historical flood data, as well as local grading and drainage provisions at the proposed ISFSI location.

The staff reviewed the description of floods resulting from locally intense precipitation in the SAR and found that flooding has been adequately described and assessed. The staff review found the description in the SAR of floods resulting from locally intense precipitation to be acceptable because the description is based largely on general information provided for DCPP, information that is also applicable to the ISFSI site. The regulations at 10 CFR §72.40(c) indicate that a re-evaluation of a site is not required when the description of floods caused by locally intense precipitation is covered under previous licensing actions. The staff has previously accepted the description of floods caused by locally intense precipitation for the DCPP, and the staff has discovered no new information that might alter the relevance of the description of floods caused by locally intense precipitation for the proposed ISFSI location. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.24(a), §72.90(a), §72.90(b), §72.90(c), §72.90(d), §72.90(f), §72.92(a), §72.92(b), §72.92(c), §72.98(a), §72.98(c)(3), and §72.122(b) with respect to this topic.

2.1.4.3 Probable Maximum Flood on Streams and Rivers

Section 2.4.3 of the SAR, "Probable Maximum Flood (PMF) on Streams and Rivers," discusses probable maximum flooding at the proposed ISFSI site caused by stream flooding. Diablo Creek is identified as the only significant channel within the drainage basin encompassing the proposed ISFSI site. In the DCPP FSAR update, the PMF upstream of DCPP is reported to have a peak discharge of approximately 195,386 l/s [6,900 cfs], with a total volume of approximately 5.3×10^6 m^3 [4,300 acre-ft] for a 24-hour storm. The applicant indicates that the drainage capacity of Diablo Creek, including the portion defined by the 3.1-m [10-ft] diameter culvert passing under the 500-kV switchyard (even if temporarily clogged) is sufficient to shed the PMF volume directly into the Pacific Ocean, with no water retention. Even if the culvert becomes plugged indefinitely, the applicant indicates that local floodwaters would pass through water-diversion features with adequate freeboard around switchyards and would not cause any flooding of the proposed ISFSI.

The staff review of this information focused on the potential for and the effects of stream overflows at the proposed ISFSI location. Section 2.1.4.2 discusses the staff review of

information concerning the potential and effects of flooding caused by locally intense precipitation. The value of PMF chosen for the proposed ISFSI is the same as that determined in the DCPP FSAR update and is based on shedding a 24-hour PMP that occurs over the entire Diablo Creek drainage basin, under the assumptions that all culverts along Diablo Creek are plugged and that antecedent ground wetness is such that flood run-off is high. The staff also considered local and regional precipitation data, historical flood data, characteristics of the Diablo Creek drainage basin, and the specific locations and elevations of interest for the proposed ISFSI.

The staff reviewed the description of the PMFs on streams and rivers in the SAR and found that the PMF has been adequately described and assessed. The staff review found the description of the PMFs on streams and rivers caused by locally intense precipitation in the SAR to be acceptable because it is based largely on general information provided for DCPP, information that is also applicable to the ISFSI site. The regulations at 10 CFR §72.40(c) indicate that a re-evaluation of a site is not required when the description of the PMFs on streams and rivers caused by locally intense precipitation is covered under previous licensing actions. The staff has previously accepted the description of the PMFs on streams and rivers caused by locally intense precipitation for DCPP, and the staff has discovered no new information that might alter the relevance of the description of the PMFs on streams and rivers caused by locally intense precipitation for the proposed ISFSI location. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.24(a), §72.90(a), §72.90(b), §72.90(c), §72.90(d), §72.90(f), §72.92(a), §72.92(b), §72.92(c), §72.98(a), §72.98(c)(3), and §72.122(b) with respect to this topic.

2.1.4.4 Potential Dam Failures (Seismically Induced)

Section 2.4.4 of the SAR, "Potential Dam Failures (Seismically Induced)," discusses the potential for flooding at the proposed ISFSI site as a result of seismically induced dam failures. The applicant states that there are no dams in the watershed area of the proposed ISFSI and that any seismically induced failures outside the watershed area would not affect the ISFSI. Similar information from the applicant concerning potential seismically induced dam failures is provided in the DCPP FSAR update.

The staff reviewed the applicant information concerning potential seismically induced dam failures and found it to be acceptable because no dams or reservoirs exist whose failure could cause flooding of the ISFSI Facility. Thus, seismically induced dam failure is not considered to be a credible threat to the proposed ISFSI site. Furthermore, the regulations at 10 CFR §72.40(c) indicate that a reevaluation of the potential for seismically induced dam failure is not required when the potential for seismically induced dam failure is covered under previous licensing actions. The staff has previously accepted the assessment of seismically induced dam failures in the FSAR for the DCPP, and the staff has discovered no new information that would alter the finding that seismically induced dam failures do not pose a credible threat to the proposed ISFSI. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.24(a), §72.90(a), §72.90(b), §72.90(c), §72.90(d),

§72.90(f), §72.92(a), §72.92(b), §72.92(c), §72.98(a), §72.98(c)(3), and §72.122(b) with respect to this topic.

2.1.4.5 Probable Maximum Surge and Seiche Flooding

Section 2.4.5 of the SAR, "Probable Maximum Surge and Seiche Flooding," discusses flooding from a maximum surge or seiche at the proposed ISFSI site. The applicant states that, because of the elevation of the ISFSI, there is no credible scenario that would create any flooding from a maximum surge or seiche. Additional information from the applicant concerning surge and seiche flooding is provided in the DCPP FSAR update.

The staff reviewed the applicant information concerning probable maximum surge and seiche flooding and found it to be acceptable because the ISFSI pad and transporter route (which bound the vertical extent of the overall ISFSI Facility) are located at elevations (see Section 2.1.4.6) that are significantly higher than the DCPP. The regulations at 10 CFR §72.40(c), indicate that a reevaluation of a site is not required when the probable maximum surge and seiche flooding is covered under previous licensing actions. The staff has previously accepted the DCPP analysis of probable maximum surge and seiche flooding and has discovered no new information in this safety evaluation that alters the previous findings. The staff concludes that the probable maximum surge and seiche flooding do not pose credible threats to the proposed ISFSI. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.24(a), §72.90(a), §72.90(b), §72.90(c), §72.90(d), §72.90(f), §72.92(a), §72.92(b), §72.92(c), §72.98(a), §72.98(c)(3), and §72.122(b) with respect to this topic.

2.1.4.6 Probable Maximum Tsunami Flooding

Section 2.4.6 of the SAR, "Probable Maximum Tsunami Flooding," discusses probable maximum tsunami flooding at the proposed ISFSI site. The proposed ISFSI pad is to be located at an elevation of +94.5 m [+310 ft] relative to MSL, which is +95.3 m [+312.6 ft] relative to mean lower low water level (MLLW). The lowest portion of the ISFSI cask transporter route is to be at an elevation of approximately 24.4 m [+80 ft] MSL. The applicant states that, because of the elevation of the ISFSI, a maximum tsunami would not cause any flooding to the ISFSI. The applicant cites data and analysis from the DCPP FSAR update as supporting basis for this statement. In the DCPP FSAR update, the maximum combined wave run-up from a distantly generated tsunami is reported as +9.1 m [30 ft] MLLW, and the maximum combined wave run-up for near shore tsunamis is reported as +10.5 m [+34.6 ft] MLLW.

PG&E indicates that little new offshore geologic or geophysical data for central coastal California have been gathered since the earlier studies conducted in the 1970s and 1980s. More information about the occurrence and generation of tsunamis in other parts of California and the world have recently become available. The SAR and responses to staff request for additional information provide an overview of available new information, including recent worldwide tsunami data, recent results from tsunami inundation mapping studies for locations in California, and information concerning sub-aerial and submarine landslide potential along central coastal California, and provide the applicant assessment of implications of that new information relative to the proposed ISFSI and to tsunami scenarios potentially affecting the

Diablo Canyon site. Considering the new information, the applicant concludes that the run-up from tsunami scenarios would not exceed the DCPP design-basis tsunami and would be significantly below the Diablo Canyon ISFSI site (pad location) and the transporter route. The applicant also discusses the potential for tsunami scenarios to reactivate movement along the existing Patton Cove landslide, but concludes that tsunamis would not cause a major reactivation of the slide.

The staff reviewed the applicant information concerning probable maximum tsunami flooding and found it to be acceptable because the ISFSI pad and transporter route (which bound the vertical extent of the overall ISFSI Facility) are located at elevations significantly higher than the DCPP. The regulations at 10 CFR §72.40(c) indicate that a reevaluation of a site is not required when the probable maximum tsunami flooding is covered under previous licensing actions. The staff has previously accepted the DCPP analysis of probable maximum tsunami flooding, the DCPP license requires regular updating of the FSAR in consideration of new information, and a special condition of the DCPP license requires updating of seismic information as part of the LTSP. Therefore, the staff found that, relative to probable maximum tsunami flooding, the proposed ISFSI is adequately covered in accordance with the previous licensing actions for DCPP. The staff has discovered no new information in this safety evaluation that alters the previous findings. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.24(a), §72.90(a), §72.90(b), §72.90(c), §72.90(d), §72.90(f), §72.92(a), §72.92(b), §72.92(c), §72.98(a), §72.98(c)(3), and §72.122(b) with respect to this topic.

2.1.4.7 Ice Flooding

Section 2.4.7 of the SAR, "Ice Flooding," discusses the potential for ice flooding at the proposed ISFSI site. The applicant indicates that flooding caused by ice melt events is not credible because of the mild climate and infrequency of freezing temperatures in the region.

In reviewing the SAR in regard to ice flooding, the staff considered regional climatology, local meteorology, historical meteorological data, surface hydrology, and other information relevant to the potential for ice-jam flood formation, wind-driven ice ridges, and ice-producing forces that may affect the proposed ISFSI site. The staff also reviewed information provided in the DCPP FSAR update regarding ice flooding.

The staff review found the applicant information regarding ice flooding to be acceptable because, based on current knowledge and data, the proposed ISFSI site is not subject to ice-flooding hazards, and thus, ice flooding is not considered to be a credible threat to the proposed ISFSI site. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.24(a), §72.90(a), §72.90(b), §72.90(c), §72.90(d), §72.90(f), §72.92(a), §72.92(b), §72.92(c), §72.98(a), §72.98(c)(3), and §72.122(b) with respect to this topic.

2.1.4.8 Flood Protection Requirements

Section 2.4.8 of the SAR, "Flood Protection Requirements," discusses flood protection requirements for the proposed ISFSI. The applicant indicates that no cooling water canals, reservoirs, rivers, or streams are used in the operation of the proposed ISFSI. Because the ISFSI does not depend on such water sources, low flow conditions or diversion of water flow are not considered. The applicant states that there are no credible hydrological scenarios that could adversely affect the proposed ISFSI, and thus, other than normal engineering provisions for grading and drainage of storm run-off, no specialized hydrological considerations or flood protection requirements are needed.

The staff reviewed the applicant information pertaining to flood protection requirements and found it to be acceptable because the analyses pertaining to surface hydrology and flooding indicate that enclosed areas of the proposed ISFSI [such as the Cask Transfer Facility (CTF)] are not subject to significant flooding or uncontrolled moisture intrusion, and exposed areas of the proposed ISFSI (such as the ISFSI pad) are not subject to levels of flooding that would adversely affect the ISFSI. In addition, ISFSI operating procedures prevent cask transport during any times of severe weather that could cause flooding along the transporter route.

2.1.4.9 Environmental Acceptance of Effluents

Section 2.4.9 of the SAR, "Environmental Acceptance of Effluents," discusses the potential for release and transport of radionuclides from the proposed ISFSI site via the hydrologic system. PG&E states that no radioactive wastes are created by the HI-STORM 100SA Cask System while in storage, while in transport, while at the CTF, or because of accidents. Because the ISFSI will not produce radioactive waste that can be incorporated into surface run-off, PG&E also asserts that surface run-off from the proposed ISFSI will have no radioactive contamination and will not adversely affect the surrounding ecosystem. Moreover, PG&E asserts that there is no public use of any surface waters or groundwater sources from the DCPP site. Thus, PG&E concludes that because no radioactive waste will be produced by the ISFSI and because surface and groundwater are not used by the public, detailed analysis of the acceptance of effluents via surface waters or groundwater as a result of ISFSI operation is not necessary.

The staff reviewed the applicant information concerning the potential for release and transport of radionuclides from the proposed ISFSI site via the hydrologic system and found it to be acceptable. The applicant has shown that the multipurpose canister (MPCs) will contain the waste. Consequently, no waste will be incorporated into the surface or groundwater systems. Thus, staff concludes that there will be no radioactive effluents. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.24(a)(m), §72.90(a), §72.92(a)(c), §72.98(b), and §72.122(b) with respect to this topic.

2.1.5 Subsurface Hydrology

The staff has reviewed the information presented in Section 2.5, "Subsurface Hydrology," of the SAR. The staff reviewed the discussion on subsurface hydrology with respect to the following regulatory requirements:

- 10 CFR §72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI be identified.

- 10 CFR §72.98(b) requires that the potential regional impacts caused by the construction, operation, or decommissioning of the ISFSI must be identified. The extent of regional impacts must be determined on the basis of potential measurable effects on the population or the environment from ISFSI activities.

- 10 CFR §72.98(c) requires that those regions identified pursuant to 10 CFR 72.98(a) and (b) must be investigated as appropriate with respect to (1) the present and future character and distribution of population, (2) consideration of present and projected future uses of land and water within the region, and (3) any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR 72.122(b) requires that (1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. (2)(i) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect (A) appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (B) appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (2)(ii) The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent nuclear fuel or high-level waste or onto structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety. (4) If the ISFSI is located over an aquifer that is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

Section 2.5 "Subsurface Hydrology," of the SAR discusses the characteristics of groundwater at the DCPP, ISFSI pad, and CTF sites. The descriptions are based on a summary of groundwater characteristics provided in the DCPP FSAR. The DCPP FSAR is maintained through periodic revisions in accordance with 10 CFR §50.71(e). Because the Diablo Canyon ISFSI site is 0.35 km [0.22 mi] from the DCPP, the regional characteristics of groundwater conditions at the ISFSI site are the same as those at the DCPP.

The SAR concludes that the groundwater table beneath the entire DCPP site is influenced by sea level at the coastline, drainages such as Diablo Creek, and the topography of the hills east

and southeast of the DCPP. Several monitoring wells at the DCPP also note perched groundwater, especially above impermeable clay beds in the Obispo Formation. Groundwater is found in the alluvium of Diablo Creek, within the terrace deposits along the coast, and in the fractured bedrock of the Obispo Formation.

The SAR notes that the groundwater table at the ISFSI and CTF site is largely controlled by Diablo Creek. The elevation of the creek adjacent to the ISFSI site is approximately 30 m [100 ft] above MSL. The elevation of the ISFSI pad site and CTF is approximately 95 m [310 ft] above MSL. Thus, there is 65 m [210 ft] distance from the groundwater table to the ISFSI site. In addition, PG&E concludes in the SAR that the intermittent clay beds observed within the Obispo Formation would act to temporarily impede groundwater flow from the ISFSI pads to the water table.

PG&E also concludes in the SAR that groundwater quality or quantity will not be affected by the proposed ISFSI in any way. Construction and operation of the ISFSI Facility, including the CTF, will not include use of groundwater. In addition, there is no public use of the local groundwater. The occurrences of perched groundwater were shown not to impact the ISFSI Facility construction or operation.

The staff review found the description of the regional characteristics of groundwater in the SAR to be acceptable because it is based largely on general information provided for the DCPP that is also applicable to the ISFSI site. The regulations at 10 CFR §72.40(c) indicate that a reevaluation of a site is not required when the regional characteristics of groundwater has been covered under previous licensing actions. The staff has previously accepted the applicant description of the subsurface hydrology for the DCPP, and the staff has discovered no new information that might alter the relevance of the description of the subsurface hydrology for the proposed ISFSI location. New information presented in the SAR concerning groundwater characteristics relative to the proposed ISFSI pad and CTF shows that groundwater characteristics at the site will not be adversely affected by the ISFSI Facility nor will groundwater conditions at the DCPP site impact construction and operation of the ISFSI. Therefore, the staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the ISFSI Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR §72.98(c)(2) and §72.122(b) with respect to this topic.

2.1.6 Geology and Seismology

Section 2.6 of the SAR, "Geology and Seismology," describes the geological and seismological setting of the DCPP. This review corresponds to the following sections of the SAR: 2.6.1, "Basic Geologic and Seismic Information;" 2.6.2, "Vibratory Ground Motion;" 2.6.3, "Surface Faulting;" 2.6.4, "Stability of Subsurface Materials;" and 2.6.5, "Slope Stability." The staff reviewed the geology and seismology of the site with respect to the following regulatory requirements:

- 10 CFR §72.90(a) requires that site characteristics that may directly affect the safety or environmental impact of the ISFSI must be investigated and assessed.

- 10 CFR §72.90(b) requires that proposed sites for the ISFSI must be examined with respect to the frequency and severity of external natural and man-induced events that could affect the safe operation of the ISFSI.

- 10 CFR §72.90(c) requires that design basis external events must be determined for each combination of proposed site and proposed ISFSI design.

- 10 CFR §72.90(d) requires that proposed sites with design-basis external events for which adequate protection cannot be provided through ISFSI design shall be deemed unsuitable for the location of the ISFSI.

- 10 CFR §72.92(a) requires that natural phenomena that may exist or that can occur in the region of a proposed site be identified and assessed according to their potential effects on the safe operation of the ISFSI. The important natural phenomena that affect the ISFSI design must be identified.

- 10 CFR §72.92(b) requires that records of the occurrence and severity of those important natural phenomena be collected for the region and evaluated for reliability, accuracy, and completeness. The applicant shall retain these records until the license is issued.

- 10 CFR 72.92(c) requires that appropriate methods be adopted for evaluating the design-basis external natural events based on the characteristics of the region and the current state of knowledge about such events.

- 10 CFR §72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI be identified.

- 10 CFR §72.98(b) requires that the potential regional impact caused by the construction, operation, or decommissioning of the ISFSI be identified. The extent of regional impacts must be determined on the basis of potential measurable effects on the population or the environment from ISFSI activities.

- 10 CFR §72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR §72.98(a) and (b) be investigated as appropriate with respect to (1) The present and future character and the distribution of population, (2) Consideration of present and projected future uses of land and water within the region, and (3) Any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR §72.102(b) requires that west of the Rocky Mountain Front (west of approximately 104° west longitude), and in other areas of known potential seismic activity, seismicity will be evaluated by the techniques of Appendix A of Part 100 of this chapter. Sites that lie within the range of strong near-field ground motion from historical earthquakes on large capable faults should be avoided.

- 10 CFR §72.102(c) requires that sites other than bedrock sites must be evaluated for their liquefaction potential or other soil instability caused by vibratory ground motion.

- 10 CFR §72.102(d) requires that site-specific investigations and laboratory analyses show that soil conditions are adequate for the proposed foundation loading.

- 10 CFR §72.102(e) requires that in an evaluation of alternative sites, those that require a minimum of engineered provisions to correct site deficiencies are preferred. Sites with unstable geologic characteristics should be avoided.

- 10 CFR §72.102(f) requires that the design earthquake (DE) for use in the design of structures must be determined as follows: (1) for sites that have been evaluated in accordance with the criteria of Appendix A of 10 CFR Part 100, the DE must be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant, (2) regardless of the results of the investigations anywhere in the continental United States the DE must have a value for the horizontal ground motion of no less than 0.10g with the appropriate response spectrum.

- 10 CFR 72.122(b) requires that (1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. (2)(i) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect (A) appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (B) appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (2)(ii) The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent nuclear fuel or high-level waste or onto structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety. (4) If the ISFSI is located over an aquifer that is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

2.1.6.1 Basic Geologic and Seismic Information

Basic geologic and seismic characteristics of the site and vicinity are presented in Section 2.6.1 of the SAR, "Basic Geological and Seismological Information." Much of the information presented by PG&E is a summary of information developed for the DCPP FSAR and LTSP.

In the SAR, PG&E summarizes the additional analyses conducted at the site to support the applicability of the DCPP geologic and seismic information to the proposed ISFSI Facility. In particular, PG&E examined the site geology from aerial photography, made direct field observations and updated site geologic maps and cross sections, gathered and interpreted subsurface data from borehole and trenches beneath the ISFSI pad and CTF sites, and collected and interpreted seismic refraction data at the site to obtain compression and shear wave velocity profiles for that site-response analysis.

Regional Geology and Tectonics

Section 2.6.1.3 of the SAR summarizes the regional geology and tectonic setting of the DCPP site. Detailed discussion is provided in the FSAR update and in the Diablo Canyon LTSP.

The DCPP lies within the tectonic plate boundary between the Pacific and North American tectonic plates. Along this plate boundary, relative motions between the Pacific and North American plates are largely manifest as dextral strike-slip motion along the San Andreas strike-slip fault system. Although the main trace of the San Andreas fault lies 77 km [48 mi] to the east of the DCPP site, the width of the plate boundary encompasses much of coastal California from the main fault trace westward to the Santa Maria Basin, several kilometers offshore.

Tectonically, the plate boundary in central California consists of a melange of crustal blocks or exotic and suspect terranes. These terranes were accreted to North America during the latter half of the Mesozoic [170–65 Ma] and are currently undergoing complex dextral strike-slip and transpressional deformation. A large area of transpression occurs south of the DCPP where the San Andreas fault makes a left bend and interacts with the left lateral Garlock fault zone. The result is the uplifted Transverse ranges.

The DCPP site lies within what has been defined as the Los Osos domain (Lettis, in press), a triangular crustal block that is undergoing north-northeast contraction in response to right-lateral motion on the San Andreas fault to the east and clockwise rotation and uplift of the Western Transverse mountains to the south. The western edge of the Los Osos domain is delineated by the Hosgri fault, which is located offshore, approximately 5 km [3 mi] to the west of the DCPP. The Hosgri fault is the largest active fault near the DCPP. As a result of analyses documented in the Diablo Canyon LTSP, an earthquake with a moment magnitude ($\mathbf{M_W}$) of 7.2 on the Hosgri fault was determined to be the controlling earthquake in the development of the design-basis ground motion spectra for the DCPP.

There is some debate in the scientific literature whether the Hosgri fault zone is a nearly vertical strike-slip fault or a thrust or reverse fault that could dip eastward underneath the DCPP site. McIntosh, et al. (1991) use reflection seismic data (profile RU-3, which crosses the Hosgri fault zone at a near 90° angle) to conclude that the Hosgri fault consists of blind thrusts and high-angle fault strands. McIntosh, et al. (1991) interpret the high-angle strands of the fault to have originally formed as basinward-dipping normal or transform faults in the early Miocene [22.7–16.6 Ma] transtensional stress regime, coinciding with the formation of the Santa Maria Basin. In the middle and late Miocene [16.6–5.3 Ma], the stress regime changed to one of compression, and the normal faults were rotated to their present orientation. According to the McIntosh, et al. (1991) interpretation, low-angle thrust faulting has occurred on the Hosgri fault since the late Miocene. McIntosh et al. (1991), however, also note right-lateral earthquake focal

mechanisms along the Hosgri fault from Gawthrop (1975) and concede that strike-slip motion on the fault is permissible within their interpretation.

More recent scientific studies conclude that the Hosgri fault is a strike-slip fault with only minor vertical motion. For example, McLaren and Savage (2001) use earthquake data to show that the Hosgri fault is a strike-slip fault with 1–3 mm/yr [0.04–0.12 in/yr] of horizontal motion. In the McLaren and Savage (2001) interpretation, the transtensional strain is partitioned between strike-slip motion on the Hosgri fault and contraction and uplift of the Los Osos domain along a number of smaller reverse faults. Two of the reverse faults nearest the DCPP are the Los Osos fault and the Southwestern Boundary fault zone. These two faults bound a crustal block in the Los Osos domain called the Irish Hills subblock. Marine terraces near the DCPP document minor uplift of the Irish Hills subblock in the Late Quaternary [last 750,000 years] with vertical displacement rates of 0.02 and 0.4 mm/yr [0.008 and 0.16 in/yr], nearly an order of magnitude smaller than the horizontal slip rates on the Hosgri fault.

This tectonic interpretation summarized by McLaren and Savage (2001)—strike-slip faulting on Hosgri fault with smaller amounts of reverse faulting on other faults near the DCPP—is not substantially different from the tectonic interpretation used by PG&E in the development of the design-basis ground motion spectra for the DCPP and accepted by the NRC (U.S. Nuclear Regulatory Commission, 1991). Therefore, staff conclude that there is no new information on the tectonic and seismic sources near the DCPP that would require PG&E to update the technical bases for earthquake sources in the existing 10 CFR Part 50 license for the DCPP. The new information on the regional geology and tectonics does not alter the original site-evaluation findings.

Site Stratigraphy

Section 2.6.1.4 of the SAR summarizes the stratigraphy of the DCPP site, including the detailed stratigraphy beneath the proposed ISFSI pad and CTF. The detailed stratigraphic analyses of strata at the DCPP were conducted by PG&E and reported in the SAR to support two aspects of the Diablo Canyon ISFSI license application. First, PG&E used the information to show that the bedrock stratigraphy beneath the proposed ISFSI site was the same as the bedrock stratigraphy under the DCPP. This condition provides the technical bases for use of the existing design-basis ground motion spectra, originally developed for the DCPP license application. Second, PG&E used the stratigraphic information to develop appropriate rock property data for the slope-stability analyses of rocks in the hillslope above the proposed pad site, the pad site itself, and the hillslopes along the transport route.

The SAR shows that both the DCPP and the proposed ISFSI site are underlain by the early-to-middle Miocene [22.7–11.2 Ma] Obispo Formation. This formation consists of fine-grained and massively bedded zeolitic tuff and a thick sequence of interbedded marine sandstone, siltstone, and dolomite. The Obispo Formation has been subdivided in the SAR into three laterally interfingering units based on lithologic changes. These units are, from east to west across the site: (i) Unit Tof_a—a massively bedded diatomaceous siltstone and tuffaceous sandstone; (ii) Unit Tof_b—a medium to thickly bedded dolomite, dolomitic sandstone, dolomitic siltstone; and sandstone; and (iii) Unit Tof_c—a thin-to-medium bedded shale, claystone, and siltstone. The Obispo Formation is intruded in places by late Miocene diabase and gabbro sills and dikes. None of these intrusive rocks is known to be under the DCPP or the proposed ISFSI pad sites. Overlying the Obispo Formation are Quaternary [1.6 Ma to the present] deposits

consisting of coastal marine terrace platforms, debris and colluvial fans, and alluvium. Quaternary deposits do not exist at the proposed ISFSI pad site because the proposed pad site is located in an area of extensive borrowing operations in the 1970s. Quaternary deposits removed from this site were used by PG&E as fill at the adjacent switchyard location.

The proposed ISFSI pad and CTF will be founded on dolomitic sandstone and dolomite of Unit Tof_b. The proposed cutslope above the pad site is also underlain by dolomitic sandstone and dolomite. Thin clay beds, 5.0–10 cm [2–4 in] thick, are present within both the dolomitic sandstone and dolomite. The clay beds are composed of kaolinite, ganophyllite, and sepiolite. They are generally parallel to bedding but laterally discontinuous. Based on trenches and boring, some of the clay beds have been correlated across distances up to 60 m [200 ft], while other clay beds appear to pinch out laterally within 15–30 m [50–100 ft]. The clay beds are more common in the dolomite than in the dolomitic sandstone. Presence of the clay beds is important to slope stability analyses and are discussed in greater detail in Section 2.1.6.4, "Stability of Subsurface Materials," of this SER.

The staff review found the description of the basic geologic and seismic information in the SAR to be acceptable because it is based largely on general information provided for the DCPP that is also applicable to the ISFSI site. The regulations at 10 CFR §72.40(c) indicate that a re-evaluation of a site is not required when covered under previous licensing actions. Staff have discovered no new information that would alter the original site evaluation findings. In addition, the staff reviewed information specific to the proposed ISFSI site in Section 2.6.1.1 of the SAR and found it acceptable because the basic geologic and seismic characteristics of the site and vicinity have been adequately described in detail to allow investigation of seismic characteristics of the Diablo Canyon ISFSI Facility. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR §72.92(a), §72.92(b), §72.102(e), and §72.122(b) with respect to this topic.

2.1.6.2 Ground Vibration

Section 2.6.2 of the SAR, "Vibratory Ground Motions," discusses the development of design bases associated with credible levels of vibratory ground motions that may be experienced at the proposed ISFSI site. The applicant provided (i) a general overview of the approach taken in developing design-basis ground motions for the proposed ISFSI Facility; (ii) a summary of the existing licensing-basis ground motions for the DCPP; (iii) a comparison of factors influencing rock site response and ground motions at the DCPP power block site and the proposed ISFSI site; (iv) description of the derivation and results of rock design spectra for proposed ISFSI pads, casks, and cask anchorage, and CTF; (v) description of the derivation and results of ISFSI long-period (ILP) rock design spectra and time histories used as input for analyses of ISFSI pad sliding, seismic stability of slopes, and transporter seismic stability; and (vi) description of soil site response along the transporter route and ground motion input used for the analysis of transporter stability.

The applicant cites 10 CFR 72.102(f) as the basis for determining ISFSI DE ground motions and indicates that seismic analyses for the proposed ISFSI use ground motions that meet or exceed DCPP ground motions based on SSE criteria of 10 CFR Part 100, Appendix A. The approach followed by the applicant consisted of: (i) using the existing DCPP SSE ground motions;

(ii) confirming the applicability of DCPP ground motions to the ISFSI site by showing similarity in site conditions and other factors influencing ground-shaking hazard; (iii) determining ground response spectra at longer vibration periods than those developed for the DCPP, accounting for near-source effects such as directivity; and (iv) developing spectra-compatible time histories for use in analyses and design.

The applicant indicates that DCPP design-basis ground motions are discussed in the DCPP FSAR update and notes the following three design spectra relevant to DCPP, the DE, the double design earthquake (DDE), and the Hosgri earthquake (HE). The existing seismic qualification basis for the DCPP (applicable also to future additions and modifications at the plant) consists of the DE and DDE (used for original design at DCPP) and the HE (used for seismic re-evaluation at DCPP) combined with the respective analytical methods, acceptance criteria, and conditions defining their implementation. In addition to these design-basis ground motions, response spectra have been developed as part of the LTSP for use in verifying the adequacy of seismic margins of specific structures, systems, and components at DCPP (Pacific Gas and Electric Company, 1988). The applicant uses the Diablo Canyon LTSP spectra as an additional basis for seismic analysis and design of the proposed ISFSI.

For the DCPP and proposed ISFSI locations, the applicant compares site conditions and distances to the seismic source having dominant contribution to local ground-shaking hazard. In comparing site conditions, the applicant notes that DCPP (power block) and the proposed ISFSI are both sited within the same continuous, thick sequence of sandstone and dolomite beds defining a single unit of geologic formation. Additionally, the applicant considers shear-wave velocity profiles for both sites and finds that shear-wave velocities at the ISFSI site are within the range of values obtained for the DCPP site. The applicant states that both sites can be classified as rock having similar ranges of shear-wave velocities. In comparing source-to-site distances, the applicant cites results of the Diablo Canyon LTSP seismic hazard analysis that indicated the Hosgri fault zone, at a distance of 4.5 km [2.8 mi] from DCPP, to be the controlling seismic source. The applicant notes that the ISFSI is just slightly farther, 245 to 366 m [800 to 1,200 ft], from the Hosgri fault zone than DCPP and states that the distances to the controlling seismic source are essentially the same for both locations. Based on similarity in site conditions and distance to the controlling seismic source, the applicant concludes that the DCPP ground motions are applicable to the ISFSI design.

For analysis and design of the ISFSI pads, casks, and cask anchorage, and the CTF, the applicant notes that the DCPP spectra for the DE, DDE, HE, and LTSP are applicable. These spectra are defined, up to vibration periods of 1.0 second (DE), 1.0 second (DDE), 0.8 second (HE), and 2.0 seconds (LTSP). Details regarding design criteria for these ground motions are evaluated in Chapters 4 and 5 of this SER. For design of ISFSI cask anchorage, the HE spectrum is used for vibration periods up to 0.8 second (i.e., the longest period for which the HE spectrum is defined), and the LTSP is used for vibration periods up to 2.0 seconds. Using the spectral-matching approach, time histories were developed consistent with the resulting (composite) spectrum. The Standard Review Plan criteria of Section 3.7.1 of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981) were used in deriving the spectral match, with the exception of the requirement for a minimum power spectral density for an NRC Regulatory Guide 1.60 spectral shape, which the applicant determines not to be appropriate for use with the HE or LTSP spectra. The applicant states that the objective of a minimum power spectral density is met by requiring the spectrum of each time history to be less than 30 percent above and 10 percent below the target spectrum.

For analysis and design pertaining to ISFSI pad sliding, slope stability, and transporter stability, cases of interest include vibration periods beyond 2.0 seconds, and the applicant thus develops the ILP spectra and associated time histories for use in these cases. The applicant indicates that ILP spectra represent 84th-percentile values of horizontal and vertical spectra that extend out to a vibration period of 10 seconds. In developing ILP spectra and associated time histories, the applicant addresses near-source effects associated with fault rupture directivity and tectonic deformation (fault fling). The applicant summarizes various considerations and assumptions incorporated into the development of ILP horizontal and vertical spectra. These assumptions and considerations include (i) assuming strike-slip faulting as the dominant slip mechanism on the Hosgri fault; (ii) identifying and using a rupture scenario with maximum directivity effects; (iii) using near-source models that incorporate directivity effects; (iv) defining the 5-percent-damped ILP horizontal and vertical spectra, for periods less than 2 seconds, as the envelope of the DDE, HE, and LTSP spectra; (v) using the 84th-percentile spectral shape to extrapolate the enveloped horizontal spectrum over the period range of 2 to 10 seconds; (vi) using a conservative vertical-to-horizontal ratio of 2/3 to extrapolate the enveloped vertical spectrum over the period range of 2 to 10 seconds; (vii) increasing the 5-percent damped horizontal and vertical spectra to ensure that they envelope corresponding HE spectra at 4- and 7-percent damping; (viii) increasing the fault-normal spectrum, over the period range of 0.5 to 3.0 seconds, to account for directivity effects near 1-second period for earthquakes having magnitude less than 7.2; and (ix) using an 84th-percentile ground-motion model of the effects of tectonic fling on the fault-parallel horizontal spectrum. The applicant provides plots of ILP ground motions, including results for fault-normal and fault-parallel spectra.

The applicant develops five sets of time histories compatible with the ILP ground-motion spectra. Representative empirical motions were used as starting bases for the spectral matching procedure. Although the Standard Review Plan spectral matching requirements of Section 7.1 of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981) recommend the use of at least 75 frequencies for matching, the applicant used 104 frequencies (29 more than recommended in NUREG–0800) as an enhancement to cover the broader period range. The requirements that no more than five ordinate frequencies fall below the (5-percent-damped) target spectrum, and no ordinates fall below 0.9 times the target spectrum, were applied. The average spectra for 2-, 4-, 5-, and 7-percent damping, produced from the five sets of time histories, enveloped target spectra for corresponding damping levels.

For analysis of transporter stability, the applicant considers the effects of soil site response on ground motions along the transporter route. The applicant notes that approximately one-third of the transporter route is founded on the same bedrock unit as the DCPP and the ISFSI. Hence, the ILP ground motions are considered applicable to that portion of the route. For the remaining two-thirds of the transporter route that crosses over surficial deposits, the applicant considers the effects of site response. The applicant performed soil response analyses at selected locations, with the finding that surface peak ground acceleration (PGA) values can be amplified by a factor of approximately 1.5 to 2.0 times the PGA value on bedrock. The applicant does not make any direct comparison of spectral results but suggests that the ILP motions for bedrock are also applicable to transporter stability analysis for the portions of the route on surficial deposits. Additionally, the applicant performed analyses that indicate that the transporter would remain stable underground accelerations twice the ILP accelerations. The applicant further evaluated the risk of scenarios where an earthquake produces twice ILP ground motions coincident with a transport operation and concluded that such scenarios are not credible (i.e., they have an annual frequency less than 10^{-7}/yr).

In reviewing the applicant analysis of vibratory ground motions, the staff considered factors related to the principal elements of seismic hazard analysis: (i) seismotectonic modeling; (ii) ground-motion prediction and near-source wave propagation; and (iii) dynamic site response. Documentation and data from the DCPP FSAR update and the Diablo Canyon LTSP seismic hazard study were reviewed, including information on (i) historical seismicity; (ii) seismogenic faults and area sources; (iii) parameters (e.g., activity rates, slip rates, b-values, maximum magnitudes) derived from seismological, geological, and geophysical investigations; (iv) applicable ground-motion relations, including models that incorporate directivity effects; (v) empirical time histories, and methods for developing time histories that match target spectra; (vi) controlling seismic sources and earthquake scenarios; and (vii) geotechnical parameters related to site response. The staff also considered new information concerning current data and available methods developed since the time of the Diablo Canyon LTSP report.

The staff review of the applicant information found it to be acceptable because, where applicable, it makes use of ground motions that meet or exceed the SSE ground motions for DCPP that have already been accepted by the NRC. The regulations at 10 CFR 72.40(c) indicate that a reevaluation of a site is not required if it is covered under previous licensing actions, except where new information is discovered that could alter the original site evaluation findings. The NRC has previously accepted the DCPP analyses of vibratory ground motions, with the Diablo Canyon LTSP as an applicable license condition. The staff has discovered no new information that alters the applicability of the current DCPP licensing basis for vibratory ground motions. Furthermore, the staff found that, in verifying the applicability of DCPP ground motions to the proposed ISFSI, the applicant has used parameters and methods of evaluation consistent with the current state of knowledge. The staff notes that, in extending the existing DCPP motions to longer vibration periods, for the purpose of developing the ILP spectra, the applicant has used an approximate approach. The applicant has nonetheless incorporated assumptions and criteria (e.g., 84th-percentile ground motions, near-source scenario that produces the maximum effects of directivity, enveloping of spectra, and other factors) that are clearly conservative and consistent with the requirements of 10 CFR Part 100, Appendix A. Therefore, the staff concludes that the ILP spectra are acceptable for use in design analyses. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR §72.92(a), §72.92(b), §72.102(e), and §72.122(b) with respect to this topic.

The staff notes that license conditions pertaining to vibratory ground motions for DCPP are considered to be also applicable for continued acceptability of ISFSI ground motions.

2.1.6.3 Surface Faulting

The description and characterization of surface faulting at the proposed Diablo Canyon ISFSI site are in Section 2.6.3, "Surface Faulting," of the SAR. Much of the information presented by PG&E is a summary of information developed for the DCPP FSAR and LTSP. In addition, potentially active faults at the DCPP site and the surrounding region were also identified and characterized by PG&E in the LTSP. From the geological and geophysical studies, PG&E showed that there is no evidence for capable faults at the DCPP site. Detailed mapping of marine terraces coupled with paleoseimologic studies (e.g., trenches, test pits, and boreholes) summarized in the SAR show that there was no significant active faulting at the DCPP site in the last 120,000 years.

The principal geologic structure at the DCPP site is the Pismo syncline. This structure folds strata of the Miocene [22.7 to 5.3 Ma] Monterey and Obispo and Pliocene [5.3 to 1.6 Ma] Pismo formations. Folding occurred in the late Pliocene or earliest Quaternary [1.6 Ma to the present]. Maps of Quaternary marine terraces across the fold axis show continuity such that folding and associated faulting must have ended at least 500,000 years ago. Thus, in the present tectonic regime, the Irish Hills subblock appears to be uplifting *en masse* without additional internal deformation.

Excavation of trenches near the proposed ISFSI site revealed minor bedrock faults with displacements up to 3 m [10 ft]. The faults are similar to minor bedrock faults encountered beneath the DCPP foundation and to bedrock faults found throughout the Obispo and Monterey formations in the Irish Hill subblock. The SAR concludes that development of these bedrock faults is either related to the growth of the Pismo syncline or to intrusion of the diabase into the Obispo formation. Both folding and intrusion have been demonstrated to have occurred in the Miocene [before 5.3 Ma].

The staff review found the description of surface faulting in the SAR to be acceptable because it is based in part on general information provided for DCPP that is also applicable to the ISFSI site. The regulations at 10 CFR §72.40(c) indicate that a reevaluation of a site is not required when the potential for surface faulting is covered under previous licensing actions. The staff has discovered no new information that would alter the original site evaluation findings. Information concerning the potential surface faulting hazard that is specific to the proposed ISFSI pad and CTF show that surface faulting at the site is not a credible hazard and will not adversely impact the ISFSI Facility nor will surface faulting at the DCPP site impact construction and operation of the ISFSI. Therefore, the staff concludes that this information is also acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR §72.90(b), §72.90(c), §72.90(d) §72.92(a), §72.92(b), §72.92(c), 72.98(b), 72.98(c)(3), and 72.122(b) with respect to this topic.

2.1.6.4 Stability of Subsurface Materials

The staff has reviewed the applicant information provided in SAR Sections 2.6.1, "Geologic, Seismologic and Geotechnical Investigations;" 2.6.4, "Stability of Subsurface Materials;" 2.6.5, "Slope Stability;" 4.1, "Location and Layout;" 4.2, "Storage Systems;" and the supporting calculations and data reports (provided as attachments to the SAR). Staff also reviewed documents[1] submitted by the applicant to respond to staff requests for additional information.

Geotechnical Site Characterization

The following geotechnical and geological studies were performed at the site by the applicant to obtain information needed for the design of the ISFSI Facility: (i) Surface geologic mapping; (ii) continuous rock coring, down-hole velocity measurements, and caliper and televiewer borehole logging; (iii) shallow seismic refraction; (iv) trenching, *in-situ* measurement of

[1]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

rock-discontinuity properties, and sampling of clay beds exposed in the trenches; and (v) laboratory testing of intact rock and clay-bed samples. The studies are documented in PG&E Calculation 52.27.100.731, "Analysis of Bedrock Stratigraphy and Geologic Structure at the DCPP ISFSI Site" (also cited as GEO.DCPP.01.21, Revision 2). Based on these studies, the applicant concluded that the ISFSI site and the hill slope to the south and southeast of the site are underlain by a thick sequence of sandstone and dolomite, within which there are several discontinuous intercalations of clay beds and friable rock. The hill slope above the ISFSI site is underlain by dolomite, but the ISFSI site is underlain by sandstone. The bedding planes in the sandstone and dolomite are generally tight and bonded. The other geologic structures in the bedrock include folds, faults, joints, and fractures.

The staff accepts the applicant description of the geologic structure and stratigraphy of the site (see Section 2.1.6.1, "Basic Geology and Seismic Information," of this SER). The geologic features of the site essential for assessing the stability of the subsurface materials are (i) the clay beds, because the clay-bed material is significantly weaker than the surrounding rock; and (ii) the rock discontinuities, specifically joints and fractures.

Characterization of the Clay Beds

The geometry (i.e., thickness, lateral persistence, and structural attitude) and engineering properties (i.e., compressibility and shear strength) of the clay beds are essential to the stability of the subsurface materials. The applicant analysis of trench and road-cut exposures and borehole encounters of the clay beds indicates that the clay beds have a maximum thickness of approximately 10 cm [4 in] and lateral persistence of approximately a few tens of meters (tens to a few hundred feet). The effective lateral persistence of the clay beds (i.e., the lateral distance over which a slip on a clay bed would not encounter any rock-to-rock contact) is likely smaller than the estimated persistence because rock-to-rock contacts occur through clay beds that have a thickness smaller than approximately 0.64 cm [0.25 in].

The compressibility of the clay beds is not important because the thinness of the clay beds and the fact that the clay is overconsolidated (through previous erosion and excavation of the overburden material) implies that any subsequent compression of the clay beds would be small and of no significance to the stability of the subsurface materials. The applicant evaluation of the shear strength of the clay beds through laboratory testing is documented in SAR Section 2.6.5.1.2.3, "Material Properties," and in PG&E Data Report G, "Soil Laboratory Test Data." Three types of laboratory shear-strength tests were performed, namely, triaxial compression, monotonic direct shear, and cyclic direct shear. The applicant lumped the results from the three types of tests (SAR Figure 2.6-50) to obtain Eq. (2-1) for the shear strength of the clay bed

$$\tau_f = 38.3 + \sigma_{fc} \tan(15°) \qquad (2\text{-}1)$$

where τ_f is the total strength (kPa) and σ_{fc} is the overburden pressure {kPa [1 kPa ≈ 21 psf]}. Because the clay-bed thickness is small and the clay-bed material is much weaker than the overlying and underlying rocks, the stress conditions during any potential failure of the clay bed are appropriately represented by the direct-shear test loading conditions but not by the triaxial compression test. The monotonic direct-shear test represents the behavior of the clay under long-term static loading conditions, whereas the cyclic direct shear represents the behavior

during seismic loading conditions. Lumping the three sets of data, therefore, is inappropriate. It can be shown by separating the applicant data into three sets (each set consisting of data from one type of test) that the cyclic direct-shear data would be appropriately represented by Eq. (2-1), but the following equation is more appropriate for representing the monotonic direct-shear data

$$\tau_f = 143.6 + \sigma_{fc} \tan(6°) \tag{2-2}$$

Equation (2-1) may be used for the analysis of stability during seismic loading conditions, but Eq. (2-2) is more appropriate for long-term static stability analyses. An examination of the two equations indicates that using Eq. (2-1) for static stability analysis would result in overestimating the applicable shear strength of the clay bed for values of overburden pressure greater than 0.65 MPa [13.5 ksf] such as for clay beds at depths greater than approximately 29.4 m [96.4 ft].

The staff accepts the applicant conclusion that the geometry of a clay bed can be approximately represented by a thin horizontal bedding plane that extends laterally approximately 30 to 60 m [100 to 200 ft]. The staff also accepts that the shear strength of a clay bed under seismic-loading conditions is approximately represented by Eq. (2-1). As explained subsequently in "Stability Against Bearing-Capacity Failure Under Dynamic Loading," the clay beds need not be considered in the evaluation of static-bearing capacity of the foundation because the foundation loading under static conditions is vertical and would not induce any shear stress in the horizontal clay beds. The difference between Eqs. (2-1) and (2-2), therefore, has no impact on foundation-stability evaluations. Furthermore, any uncertainty introduced by using Eq. (2-1) instead of Eq. (2-2) to evaluate the static stability of slopes would be adequately accounted for by the safety factor required to demonstrate stability under static conditions. The staff concludes that the applicant characterization of the clay beds is based on accepted techniques. Therefore, the information presented is adequate for use in other sections of the SAR to perform additional safety analyses and demonstrate compliance with regulatory requirements in 10 CFR §72.90(a), §72.90(b), §72.90(c), §72.90(d), §72.92(a), §72.92(b), §72.92(c) §72.102(c), §72.102(d) and 72.122(b).

Characterization of Rock Discontinuities and the Discontinuous Rock Mass

Rock Discontinuities: The *in-situ* friction angle of the discontinuities is estimated using Barton's equation (Barton and Choubey, 1977) in PG&E Calculations 52.27.100.730, "Development of Strength Envelops for Sallow Discontinuities at DCPP ISFSI Using Barton's Equations" (also cited as GEO.DCPP.01.20). At the slope face, the stress-relieved rock tends to dilate along preexisting discontinuities as asperities on the joint surfaces override one another. The nonlinear shear strength in Barton's joint model is dependent on the joint roughness coefficient (JRC), joint compressive strength (JCS), and a base friction angle. The JRC was estimated from field and laboratory assessments either by measurement or by comparison with a standard roughness profile (Barton and Choubey, 1977). The JCS was assumed to be 25 percent of the unconfined compressive strength of intact rock. The analysis of laboratory test data of unconfined compressive strength for dolomite and sandstone is given in PG&E Calculations 52.27.100.727, "Determination of Mean and Standard Deviation of Unconfined Compressive Strength for Hard Rock at DCPP ISFSI, Based on Laboratory Tests" (also cited as GEO.DCPP.01.17). The mean and standard deviations of JRC and JCS were evaluated for joint, bedding planes, and faults in dolomite and sandstone. The basic friction angle was estimated from laboratory direct-shear tests as described in PG&E Calculation 52.27.100.728,

"Determination of Basic Friction Angle Along Rock Discontinuities at DCPP ISFSI, Based on Laboratory Tests" (also cited as GEO.DCPP.01.18). Shear strength failure envelopes were developed for joints, bedding planes, and faults in dolomite and sandstone for a range of JCS, JRC, and base friction angles varied about their mean values. The *in-situ* friction angle was determined from the inclination of the tangent line on the shear-strength failure envelope at the midpoint of the normal stress range of 0–0.15 MPa [21.75 psi]. The normal stress on the wedge surfaces is not expected to exceed approximately 0.2 MPa [29 psi] considering the height of the slope {52.3 ft [5.9 m]}, density of the rock {140 pcf [0.022 MN/m^3]} and average orientation of the discontinuities. The range of normal stress used for the analysis is, therefore, consistent with the applicable normal stress at the site. The *in-situ* friction angles varied between 17.5–54° for dolomite and 16–46° for sandstone. The average friction angle of 34° for dolomite or 30° for sandstone is higher than the base friction angle of 28°, which is expected considering the effect of surface roughness on the shear strength of discontinuities (e.g., Barton and Choubey, 1977).

The staff concluded that the applicant evaluation of the *in-situ* strength of rock discontinuities is based on standard methods (Hoek, 2000a; Barton and Choubey, 1977). Therefore, the information presented is adequate for use in other sections of the SAR to perform additional safety analyses and demonstrate compliance with regulatory requirements in 10 CFR §72.90(a), §72.90(b), §72.90(c), §72.90(d) §72.92(a), §72.92(b), §72.92(c) §72.102(c), §72.102(d), and 72.122(b).

Strength of the Discontinuous Rock Mass: The strength of the discontinuous rock mass modeled as a continuum is needed to evaluate the stability of the rock mass against potential failure modes for which the rock-mass behavior can be considered as similar to the behavior of a homogeneous continuum. The staff evaluated the applicant estimation of rock-mass strength at the ISFSI pad site presented in PG&E calculation 52.27.100.729, "Development of Strength Envelope for Jointed Rock Mass at DCPP ISFSI Using Hoek-Brown Equation," (also cited as GEO.DCPP.01.09) and supporting documents. The applicant used the Hoek-Brown criterion (Hoek, 2000b) to develop the shear strength envelopes and estimate the shear strength parameters of the jointed rock mass in dolomite and sandstone. The rock-mass strength was utilized in foundation design for ISFSI pads and CTF and the analysis of slopes above the ISFSI pads.

The Hoek-Brown criterion defines a nonlinear empirical relationship between principal stresses associated with failure of moderately to heavily jointed rock mass. The failure criterion considers the characteristics of the intact rock and the geologic structures and surface conditions of discontinuities. The rock-mass strength is estimated using two intact-rock strength parameters, namely, uniaxial compressive strength, σ_{ci} ,and material index, m_i , and one rock-mass quality parameter referred to as the Geological Strength Index (GSI).

The unconfined compressive strengths of dolomite and sandstone were evaluated from laboratory tests on intact rock as discussed in PG&E Calculation "Determination of Mean and Standard Deviation of Unconfined Compressive Strength for Hard Rock at DCPP ISFSI, Based on Laboratory Tests" (also cited as GEO.DCPP.01.17). The average value of σ_{ci} was determined to be 31.1 MPa [4,517 psi] with standard deviation of 14.7 MPa [2,131 psi] for dolomite based on test results of 12 samples and 21.8 MPa [3,165 psi] with standard deviation of 9.3 MPa [1,348.9 psi] for sandstone based on test results of five samples.

The applicant estimated GSI and m_i from visual observation by individual members of a team consisting of two to three field geologists. The value of m_i was assigned by comparing the mineralogical and sedimentological characteristics of the exposed rock in exploratory trenches against the estimated values of m_i given by Hoek (2000b) for similar rock types. The means and standard deviations of m_i values for dolomite were estimated as 15.4 and 2.0, and for sandstone, 17.8 and 1.0. The Hoek-Brown constant, m_i, is usually determined by statistical analysis of sets of triaxial tests conducted on intact rock core samples (Hoek, 2000b). The applicant estimate for m_i for sandstone compares well with the estimated value of 14.3 from laboratory tests (Hoek and Brown, 1980). However, the applicant estimate for dolomite is significantly larger than the laboratory-determined value of 6.8 (Hoek and Brown, 1980). An m_i value of 6.8 for dolomite, compared with the value of 15.4 used by the applicant, implies that the rock-mass friction angle calculated by the applicant (considering the values GSI discussed subsequently) could be smaller by approximately 5°. This uncertainty in the rock-mass friction angle is acceptable as will be discussed subsequently in this section.

The applicant estimated the GSI by visual inspection of the rock exposures in the exploratory trenches and road cuts. The GSI was estimated by comparing the rock-mass structure and discontinuity surface conditions against the GSI classification developed by Hoek (2000b). The means and standard deviations of GSI for dolomite were 55.7 and 9.3, and for sandstone, 64.8 and 3.1.

The applicant performed several spreadsheet calculations and developed nonlinear shear strength failure curves with different combinations of mean, lower-bound, and upper-bound values of the three parameters. The lower-bound and upper-bound values are smaller and larger than the mean value by one standard deviation.

The applicant considered a straight-line failure envelope, with a friction angle of 50° and cohesion of zero, which represented a lower bound for all the calculated rock-mass shear strength curves for dolomite and sandstone. The applicant indicated that the rock-mass shear and compressive strengths are significantly smaller than the uniaxial compressive strength of intact rock determined from laboratory test data. The estimated rock-mass shear strength shows a zero uniaxial compressive strength and a steeply rising shear-strength envelope. The applicant did not provide any triaxial test data for the ISFSI-site intact rock. The staff, therefore, used intact-rock shear strength envelopes available in the literature (e.g., Goodman, 1983) in combination with the applicant unconfined compressive strength data to assess the applicant rock-mass strength information. Using a friction angle of 37.2° and unconfined compressive strength of 21.8 MPa [3,165 psi] for sandstone, a cohesion of 5.4 MPa [783 psi] was determined using the relation given in Goodman (1983)

$$q_u = 2c \tan(45 + 0.5\phi) \qquad (2\text{-}3)$$

where q_u is the unconfined compressive strength, c is the cohesion, and ϕ is the friction angle. Similarly, using a friction angle of 35.5°(Goodman, 1983) and unconfined compressive strength of 31.1 MPa [4,517 psi] for dolomite, a cohesion of 8.12 MPa [1,177 psi] was calculated for

intact dolomite rock. The straight-line failure envelope of the rock mass defined by a friction angle of 50° and zero cohesion is compared with the resulting intact-rock strength envelopes for sandstone and dolomite defined by the c and ϕ values. The rock-mass strength envelope would intersect the intact-rock envelopes at a normal stress of 12.5 MPa [1,813 psi] for sandstone and 16.7 MPa [2,422 psi] for dolomite, which suggests that the rock-mass strength obtained using the friction angle of 50° would not exceed the strength of intact sandstone for values of normal stress smaller than 12.5 MPa [1,813 psi] or the strength of intact dolomite for values of normal stress smaller than 16.7 MPa [2,422 psi]. These normal stresses are substantially higher than the normal stress of 1.3 MPa [188.5 psi] on a typical failure plane at a maximum depth of 60 m [200 ft]. The applicant characterization of the rock-mass strength, therefore, satisfies the condition that the rock-mass strength should not exceed the intact rock strength.

The rock-mass strength was evaluated further by considering a disturbance factor, D, that was recently introduced by Hoek, et al. (2002) to account for rock-mass weakening from blast damage or stress relaxation. The value of D varies between 0 for the undisturbed case to 1.0 for the disturbed case. The rock properties evaluated using the Hoek-Brown criterion in PG&E Calculation 52.27.100.729 represent rock-mass strength for the undisturbed condition. Stress relief from the removal of overburden, such as by excavation, would cause a disturbance of the rock mass. Hoek, et al. (2002) recommended a D value in the range of 0.7–1.0 and suggested that assuming an undisturbed *in-situ* rock condition for slopes would result in an overestimation of the applicable rock-mass strength. The rock mass underlying the ISFSI site and the hill-slope area close to the ISFSI site has been subjected to such a stress relief because of the previous excavation of the slope for borrow material. Additional stress relief is expected to result from future excavation needed for the ISFSI construction.

Additional information provided by the applicant[2] indicated that any effects of stress relief from the 1971 excavation at the site are implicitly included in the measured geologic parameters used as input into the analyses to determine the Hoek-Brown parameters for the rock-mass. The applicant also indicated that the proposed excavation at the ISFSI site would be performed using large earth-moving equipment and explained that a zero value for D is appropriate for slopes excavated using such techniques. The applicant obtained an expert opinion from E. Hoek[3] to support its view regarding the applicable value of D. The staff accepts that a zero value for D is appropriate for the ISFSI site rock mass, considering that any stress-relief effect from the 1971 excavation has been implicitly accounted for and the excavation technique proposed by the applicant would not cause any appreciable rock-mass disturbance.

The staff concludes that the applicant determination of an equivalent-continuum strength for the discontinuous rock mass is based on standard methods (Hoek, 2000b; Hoek, et al., 2002). Therefore, the information presented is adequate for use in other sections of the SAR to perform additional safety analyses and demonstrate compliance with regulatory requirements in

---

[2]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[3]Ibid.

10 CFR §72.90(a), §72.90(b), §72.90(c), §72.90(d), §72.92(a), §72.92(b), §72.92(c), §72.102(c), §72.102(d) and 72.122(b).

Stability of Cask-Storage Pad Foundation

The proposed cask-storage design (SAR Section 4.1, "Location and Layout") consists of seven contiguous reinforced concrete pads covering an area of approximately 152.4 m [500 ft] by 32.0 m [105 ft]. Each storage pad is 2.29 m [7.5 ft] thick and is designed to accommodate up to 20 storage casks in a 4 by 5 array. Each storage cask is approximately 6.10 m [20 ft] high, has a circular cross section of 3.35 m [11 ft] in diameter, and weighs approximately 163,293 kg [360,000 lb] if loaded. The base of the storage pad, therefore, will be subjected to a maximum static pressure of approximately 0.10 MPa [14.4 psi], considering the weight of a fully loaded pad. The potential dynamic loading for the design of the ISFSI foundation is reviewed in Section 2.1.6.2, "Ground Vibrations," of this SER.

Stability Against Bearing-Capacity Failure Under Static Loading: The allowable bearing pressure for the storage pad under static loading was developed in PG&E Calculation 52.27.100.713 (also designated as GEO.DCPP.01.03). The calculation was performed using two methods. First, considering the subsurface material as a homogeneous rock characterized by an estimated rock quality designation of 18 resulted in an allowable bearing pressure equal to 1.92 MPa [276.5 psi]. Second, the allowable bearing pressure was calculated by considering the subsurface material as a homogeneous soil characterized by zero cohesion and a friction angle of 50°, and using a bearing-capacity equation applicable to a shallow foundation subjected to a centered vertical static load. This approach resulted in an allowable bearing pressure equal to 3.83 MPa [551.6 psi]. Each fully loaded storage pad would be subjected to a centered vertical load resulting in a maximum static-bearing pressure of 0.10 MPa [14.4 psi], considering the dead weight of one concrete pad loaded with 20 casks.

The staff reviewed the applicant evaluation of the stability of the cask storage pads with respect to the potential for bearing-capacity failure under static loading. The applicant evaluation was performed using standard methods. The staff has noted previously in "Strength of the Discontinuous Rock Mass" that the rock-mass friction angle can be smaller by approximately 5° than the value used by the applicant. This potentially smaller value of rock-mass friction angle, however, does not affect the staff assessment of the applicant evaluation of the static-bearing capacity of the storage pads. The allowable bearing pressure remains significantly larger than the maximum static-bearing pressure of 0.10 MPa [14.4 psi] even considering a substantially smaller rock-mass friction angle. Therefore, the staff concludes that the proposed cask-pad design is adequate considering the potential for bearing-capacity failure under static loading. The staff also concludes that the information provided in the SAR regarding the static-bearing capacity of the storage pads is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR §72.102(c), §72.102(d) and §72.122(b).

Stability Against Bearing-Capacity Failure Under Dynamic Loading: The applicant evaluation of the bearing capacity of the storage-pad foundation subjected to dynamic loading from the design-basis earthquake is documented as part of the additional information provided by the

applicant[4] to respond to staff requests. The evaluation considered the two failure modes illustrated in Figure 2-1. These potential failure modes needed to be considered because of the occurrence of near-horizontal clay beds that are significantly weaker than the surrounding rocks. The occurrence of the clay beds does not constitute a problem under static-loading states, because the static load is applied vertically and, as a result, would not cause any significant shear stress in any of the clay beds. Dynamic loading from the design-basis seismic ground motion, however, has horizontal and vertical components. The clay beds within the subsurface materials under the storage pad would, therefore, be subjected to shear stresses during an earthquake. The shear stress in the clay beds may result in a bearing-capacity failure of the storage pad following one of the two failure modes illustrated in Figure 2-1.

The applicant analysis for the horizontal-sliding failure mode [Figure 2-1(a)] is documented in PG&E Calculation GEO.DCPP.03.01, Revision 0, "Determination of Earthquake-Induced Displacements of Potential Sliding Mass Beneath DCPP ISFSI Site." The analysis considered the stability of an approximately rectangular block of rock 155.4 m [510 ft] long, 149.4 m [490 ft] wide, and 15.2 m [50 ft] thick. The top of the block includes seven ISFSI pads carrying 20 fully loaded storage casks each, and the base surface of the block is assumed to pass through a thin but continuous clay bed. The dimensions of the block were determined based on the geometry of the ISFSI pad, the canyon wall to the north of the pad, and the clay beds. The driving force considered in the analysis consists of a static representation of the lateral earthquake force calculated as αW, where W is the total weight of the block and, $\alpha = 0.43$, which is about half of the PGA for the design-basis ground motion. This evaluation of the driving force is consistent with standard practice (e.g., Kramer, 1996, p. 436). Resistance to sliding was considered as arising from the shear resistance of the clay bed at the base of the block and the shear resistance of the rock-mass on the vertical boundary surface of the block, except the canyon face. The shear resistance parameters used in the analysis are consistent with the information reviewed previously under "Geotechnical Site Characterization."

A factor of safety of 1.02 against horizontal sliding [similar to the failure mode illustrated in Figure 2-1(a)] was obtained through the analysis. This factor of safety indicates that a potential horizontal sliding of the ISFSI foundation during a design-basis earthquake cannot be ruled out based on the analysis provided by the applicant. The applicant performed an evaluation of the potential sliding displacement using the Newmark (1965) sliding-block approach. The analysis indicates that the foundation may experience a sliding displacement [Figure 2-1(a)] on the order of a few inches {approximately 1.0 cm [2.5 in]} during a design-basis earthquake. Staff concludes that such horizontal displacement would not have any significant impact on any safety function of either the pads or the storage casks.

The applicant analysis for the rotational failure mode [Figure 2-1(b)] is documented in a PG&E Calculation, "Evaluation of Potential Rotational Failure of Foundation Beneath DCPP ISFSI

---

[4]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

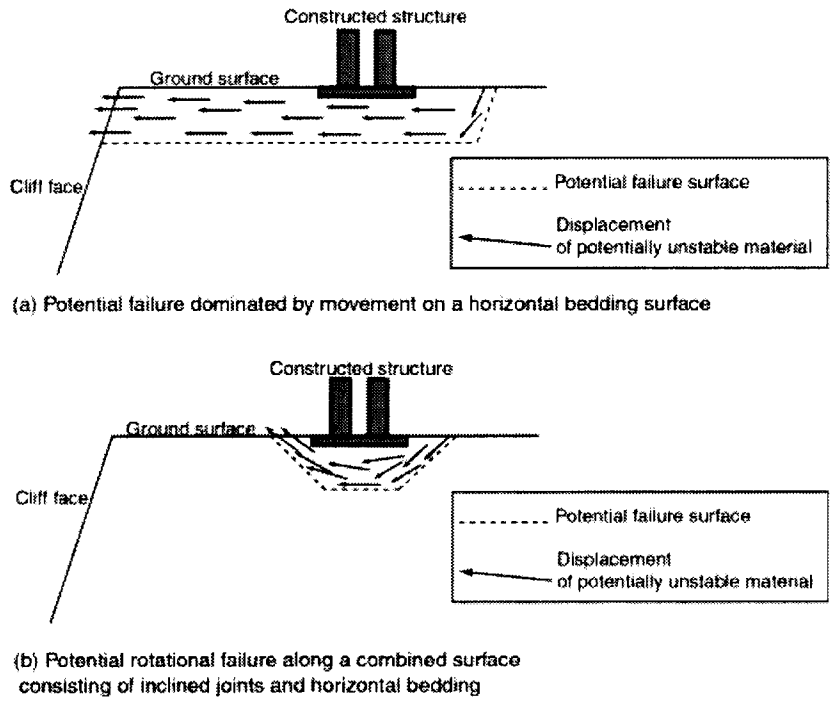

Figure 2-1. Potential failure modes of the subsurface material under the cask-storage pad when subjected to the design-basis seismic ground motion. Both failure modes were considered in the applicant evaluation of the seismic stability of the pad.

Pads in Attachment 2-2."[5] The analysis consists of evaluating the allowable bearing pressure for the pad and comparing it against the bearing pressure that may be imposed on the pad during a design-basis earthquake. The allowable bearing pressure for a shallow foundation such as the storage pad is equal to the ultimate bearing capacity, such as, the value of foundation pressure that would cause a rotational failure [e.g., Figure 2-1(b)], divided by a safety factor.

The applicant evaluated the ultimate bearing capacity of the pad using a standard bearing-capacity formula (e.g., Terzaghi, et al., 1996, p. 260) for a homogeneous soil deposit or a deposit for which strength increases continuously with depth. To apply the formula, the applicant assumed that the subsurface material at the pad site is homogeneous with a strength equal to the strength of the clay beds. The applicant approach is accepted because it would result in a lower-bound ultimate bearing capacity for the pad foundation. The clay-bed strength used in the analysis is consistent with the information reviewed previously under "Geotechnical Site Characterization." The analysis resulted in an ultimate bearing capacity of approximately 0.44 MPa [9,238 psf] for the pad.

---

[5]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

The applicant estimated the bearing pressure imposed by the design-basis earthquake from the results of a finite element model analysis of the pad foundation (PG&E Calculation PGE–009–CALC–003, Revision 2, "ISFSI Cask Storage Pad Seismic Analysis"). The finite element analysis has been reviewed and accepted as documented in Section 5.1.3.4, "Structural Analysis for Reinforced Concrete Structures," of this SER. The analysis indicates that the storage pad would tend to rotate about a horizontal axis during a design-basis earthquake, resulting in a bearing pressure that attains a maximum value along an edge of the pad and decreases to zero near the middle of the pad. Such a rotation would cause a rotational failure of the pad foundation [e.g., Figure 2-1(b)] if the maximum pressure equals the ultimate bearing capacity of the foundation. A maximum bearing pressure of approximately 0.33 MPa [6,950 psf] was determined based on the analysis.

The applicant estimated the pressure on a clay layer at a depth of 7.92 m [26 ft] below the pad by applying a geometric attenuation factor to the maximum foundation bearing pressure. The estimated clay-layer pressure was compared with an allowable bearing pressure calculated as $(1.3/3)P_u$ where P_u is the ultimate bearing capacity of 0.44 MPa [9,238 psf]. The applicant explained that the factor of 1.3 accounts for an increase in dynamic bearing capacity relative to the static value and the factor of one-third is the inverse of the applicable safety factor. The staff did not accept this aspect of the applicant analysis because comparing the pressure at the base of a foundation (the allowable bearing pressure) with the pressure on a clay layer at a depth of 7.92 m [26 ft] below the foundation is inappropriate. The staff, however, determined that the information provided by the applicant is sufficient to complete an evaluation of the stability of the pad foundation during a design-basis earthquake. The allowable bearing pressure can be obtained by applying a safety factor of 1.1, based on U.S. Nuclear Regulatory Commission staff guidance (U.S. Nuclear Regulatory Commission, pp. 3.8.5–7), to the ultimate bearing capacity, which gives 0.40 MPa [8,398 psf] as the allowable bearing pressure for the pad subjected to dynamic loading. Because this allowable bearing pressure is larger than the maximum bearing pressure of 0.33 MPa [6,950 psf], the staff concludes that the information provided by the applicant is sufficient to show that the storage-pad foundation would not experience a rotational failure during a design-basis earthquake.

Therefore, the staff concludes that the proposed cask-pad design is adequate considering the potential for bearing-capacity failure under dynamic loading from the design-basis earthquake. The information provided in the SAR regarding the dynamic bearing capacity of the storage pads is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR §72.102(c), §72.102(d) and §72.122(b).

Staff Evaluation

Staff has reviewed Section 2.6.4, "Stability of Subsurface Materials," of the SAR and concludes that the information presented in this section is adequate for use in other sections of the SAR to develop the design bases of the ISFSI Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR §72.102(c), §72.102(d), and §72.122(b).

2.1.6.5 Slope Stability

The staff has reviewed information presented in the SAR Section 2.6.5, "Slope Stability," which provides an evaluation of the stability of natural and cut slopes as follows: 2.6.5.1, "Stability of the Hill Slope Above the ISFSI;" 2.6.5.2, "Stability of Cut Slopes;" 2.6.5.3, "Slope Stability at CTF Site;" and 2.6.5.4, "Slope Stability Along the Transport Route."

Stability of the Hill Slope Above the Storage Pad

The area to the south and southeast of the proposed storage-pad site is occupied by a hill that rises at an average slope of approximately one vertical to three horizontal units to a maximum elevation of approximately 182.9 m [600 ft] above the pad site. The hill is formed of jointed dolomite and sandstone with discontinuous intercalated of clay beds. The applicant interpretation of geologic data from boreholes, trenches, and outcrops indicates that the stability of the slope would be controlled by potential sliding on compound surfaces formed by a combination of joints, clay beds, and new tension cracks. The clay beds are the weakest materials within the hill rock mass, such that any rock failure would likely initiate through slip on a clay bed. Because the clay beds are discontinuous, any such slip surfaces would have to connect through slip on joints and, if necessary, failure of rock bridges between joints or clay beds.

The stability of the hill slope above the storage pad is of concern for the following reasons. First, a potential failure along a surface that daylights above the pad may result in the casks being impacted by rock blocks and debris, and accumulation of these materials on the storage pads. Second, a potential failure along a surface that daylights to the north of the pad would cause a rotational failure of the pad foundation, similar to the potential failure mode illustrated in Figure 2-1(b). The applicant has chosen not to assess the consequences of such events but instead to establish, through a slope-stability evaluation, that such events are not credible.

The applicant approach to evaluating the stability of the hill slope consists of making a case that (i) a failure of the hill slope during nonseismic (i.e., long-term static) conditions is unlikely; (ii) the displacement of any potentially unstable mass during a potential earthquake would be small and, therefore, of no concern to the safety of the facility; and (iii) adequate rockfall mitigation measures would be provided such that any rockfall generated from any seismically induced slope instability would not impact the storage casks.

Static Stability of the Hill Slope Above the Storage Pad: The applicant evaluation of the long-term stability of the hill slope subjected to static loading is documented in Section 2.6.5.1.2 (pp. 2.6-45 to 2.6-48) of the SAR, in PG&E Calculation 52.27.100.734 (also cited as GEO.DCPP.01.24, Revision 1), and in Attachment 3-1.[6] The evaluation was based on a series of two-dimensional limit-equilibrium analyses. The analysis procedure, referred to as the method of slices, is standard and documented satisfactorily. Three sets of analyses were performed. The first set considered the stability of potentially unstable masses that were predefined based on the interpreted geometry of clay beds. The second set considered the

[6]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

stability along circular slip surfaces located using a random search routine. The third set of analyses also considered the stability of randomly selected circular slip surfaces but differ from the second analysis set because the geometry of the proposed ISFSI excavation was included in the analyses.

The applicant did not include any water pressure in the analyses. The groundwater table is at a depth of approximately 65 m [210 ft] below the storage pad. Furthermore, rainwater is expected to drain freely through the fractured rock, and any perched water that may accumulate on a clay bed is likely to be small and short-lived because the clay beds are discontinuous and surrounded by fractured rock. The staff accepts the applicant justification for not including water pressure in the stability analysis of the hill slope.

The first set of analyses consists of ten cases, each considering the equilibrium of a different potentially unstable mass (PG&E Calculation 52.27.100.734, Figures 1–10). The failure surface in each case is a composite surface that involves slip on one or more clay beds and failure of the fractured rock mass. The strength of the fractured rock was represented using a friction angle of 50° and zero cohesion. The shear strength of the clay beds was assigned using Eq. (2-1).

The results of the analyses are summarized in the SAR Table 2.6-3. Nine analysis cases gave values of safety factor greater than 2.1, and the tenth gave a safety factor of 1.62. The use of a friction angle of 50° for the rock mass is not likely to have a significant effect on the calculated safety factor because the rock-mass sections of the failure surfaces are steeply inclined. Because the normal stress on such steeply inclined surfaces is close to zero, any frictional resistance contributed by such sections of the failure surface would be small, such that changing the friction angle for the rock-mass sections of the failure surfaces would not have a significant effect on the calculated safety factor. Any uncertainty in the rock-mass friction angle, therefore, would not have a significant effect on the calculated safety factor. The staff also determined that the effect of using Eq. (2-1) instead of Eq. (2-2) for the clay-bed strength would be to overestimate the safety factor by a factor of approximately 1.4 considering the strength contribution of sections of the clay beds subjected to a vertical stress greater than 0.65 MPa [94.3 psi], that is, clay-bed sections under a rock column of height greater than 29.3 m [96 ft], which excludes the analyses for slide mass 1a and 1b (FSAR Figure 2.6-47). A minimum safety factor of approximately 1.51 would be obtained by dividing the applicable safety factors in SAR Table 2.6-3 by 1.4, except the safety factor of 1.62 from slide mass 1b.

In the second set of analyses, the rock mass was considered as a homogeneous continuum with a shear strength defined by a friction angle of 50°. The analyses consist of an evaluation of the conditions for limit equilibrium along several randomly selected circular slip surfaces. The analyses, documented in Attachment 3-1,[7] produced a minimum safety factor of 3.26.

As explained previously, the third set of analyses was similar to the second set but also included the geometry of the proposed ISFSI excavation, which was not included in the second set. The analyses also included rock anchors assumed to be installed on the cut slope face

---

[7]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

following a regular pattern and a center-to-center separation of 1.52 m [5 ft]. Each anchor, as represented in the analyses, was 9.14 m [30 ft] long, includes a bond section 3.35 m [11 ft] long, and was prestressed to approximately 340 kN [76,500 lb]. The analyses produced a minimum safety factor of 3.27.

A safety factor in the range of 1.25–1.5 is an acceptable safety margin against failure of a natural slope subjected to static loading (e.g., National Research Council Transportation Research Board, 1978, p. 172). The higher value of 1.5 is preferred for critical slopes (e.g., Hoek and Bray, 1977; U.S. Nuclear Regulatory Commission, 1977). Based on these considerations, the staff concludes that the long-term static stability of the hill slope, including the change in slope geometry that would result from the proposed ISFSI excavation, would be adequate to maintain safety. The information provided in the SAR regarding the long-term static stability of the hill slope is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR §72.102(d) and §72.122(b).

Seismic Stability of the Hill Slope Above the Storage Pad: Slope stability analyses provided by the applicant (SAR Section 2.6.5.1.3, "Seismically Induced Displacements") indicate that rock and debris may be dislodged from the hill slope above the ISFSI during a design-basis earthquake. The applicant provided additional information in SAR Section 2.6.5.1.3, "Seismically Induced Displacements"[8] to make a case that the displacement of any rock volume dislodged from the hill slope during a design-basis earthquake would be small. The applicant also committed to provide a sufficient rockfall mitigation to protect the ISFSI storage casks from rockfall impact that may arise from such dislodged rock. The applicant calculated seismically induced displacements through a series of analyses based on an approach proposed by Newmark (1965). The calculation consists of three steps for a selected potentially unstable mass.

First, the value of yield acceleration, k_y, was calculated through a limit-equilibrium analysis similar to the analysis described previously in "Static Stability of the Hill Slope Above the Storage Pad." The parameter k_y is the horizontal acceleration that would cause the value of the safety factor against sliding of the potentially unstable mass to decrease to 1.0 from the value calculated for the long-term static condition. The acceleration is represented in the analysis as a static horizontal force k_yM, where M is the mass of the potentially unstable mass. The calculation of k_y is documented in PG&E Calculation 52.27.100.734 (also cited as GEO.DCPP.01.24, Revision 1). The calculated values of k_y are given in the SAR Table 2.6-3 for ten different potentially unstable masses and are in the range of 0.19–0.44g.

Second, an average horizontal acceleration time history, a_{ht}, for the potentially unstable mass was calculated [PG&E Calculation 52.27.100.735 (also cited as GEO.DCPP.01.25, Revision 1)]. The average acceleration time history should be based on the ratio F_t/M where F_t is the resultant down-slope force along the potential failure surface (e.g., Kramer, 1996, p. 446). The applicant, instead, calculated the acceleration by averaging the nodal horizontal accelerations in a finite element model of the potentially unstable mass. The applicant argued that this nodal

---

[8]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Design Mitigation Features Information to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399)*. Letter (May 6) to U.S. Nuclear Regulatory Commission (DIL–03–007). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

average is a satisfactory representation of the average acceleration because the rock material is stiff enough that the acceleration of the rock at every point within the potentially unstable mass is similar to the input acceleration time history. Additional information was provided by the applicant in Attachment 4-1[9] to respond to a staff request for a verification of the applicant approach of calculating a_{ht} by averaging nodal accelerations. The additional information indicates that the values of a_{ht} calculated using F_t/M are essentially the same as the values calculated by averaging nodal accelerations.

Third, the difference $a_{ht} - k_y$ for $a_{ht} > k_y$ was integrated twice with respect to time to obtain a displacement, the so-called Newmark displacement. This calculation is documented in the SAR Section 2.6.5.1.3, "Seismically Induced Displacements;" and in PG&E Calculation 52.27.100.736 (also cited as GEO.DCPP.01.26, Revision 1). The calculated displacements are given in SAR Table 2.6-5 and are in the range of 12–94 cm [0.4–3.1 ft]. The applicant interprets the calculated Newmark displacements as representing the magnitude of seismically induced displacements of the potentially unstable masses. Information obtained from other literature (as described subsequently), however, suggests that the Newmark displacements should be interpreted qualitatively when used for the assessment of the stability of a rock slope.

The Newmark displacement calculation is based on an approach proposed in Newmark (1965) for evaluating the potential seismically induced deformations of an embankment dam. The approach is based on the principle that significant displacements of the dam material would begin when the inertial forces on a potentially unstable mass are large enough to overcome the yield resistance and would stop when the inertia forces decrease to values smaller than the yield resistance. The method is implemented by performing a double integration of the acceleration difference $a_{ht} - k_y$ for $a_{ht} > k_y$. The method has been used extensively for earth dams and embankments, but the staff has not found any documentation of a successful use of the method for estimating seismically induced displacements in a rock slope. The method has been discussed in several review articles (e.g., Seed, 1979; Ambraseys and Menu, 1988) and textbooks (e.g., Kramer, 1996; Abramson, et al., 2002). All the applications of the Newmark calculation cited in Seed (1979), Ambraseys and Menu (1988), and Abramson, et al. (2002) are for earth dams and embankments. Similarly, seven of the nine publications on the Newmark method cited in Kramer (1996) are for earth dams and embankments and the other two are for landslides. One of the two landslide examples, [Jibson (1994)], is a review article. Only one, [Wilson and Keefer (1983)], reports an actual application of the method for estimating seismically induced landslide displacements.

This lack of published information on any application of the method to rock slopes accentuates the uncertainty regarding potential interpretations of seismically induced down-slope rock displacements calculated using the method. Materials used for earth dams and embankments typically have high damping, such that any seismically induced motion of such materials is likely to stop as soon as the source of excitation is removed. This behavior may explain why the double integration of the excess acceleration ($a_{ht} - k_y$ for $a_{ht} > k_y$) appears to give a reliable order-of-magnitude estimate of the seismically induced embankment deformations. Rock

---

[9]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

slopes, in comparison, are typically composed of low-damping materials. It is, therefore, conceivable that a rock body dislodged from a slope may continue its down-slope motion even if the condition $a_{ht} > k_y$ is no longer satisfied. In that case, any displacement calculated for such a rock body using the Newmark approach would only be a lower-bound estimate of the seismically induced displacement.

As suggested in Abramson, et al. (2002, p. 408), the Newmark displacements should be interpreted as a qualitative indication of the stability of a slope when subjected to the ground motion specified in the analysis. The State of California proposed the following guideline for such a qualitative interpretation (State of California Division of Mines and Geology, 1997, Chapter 5).

(1) Newmark displacements of 0–10 cm [0–3.94 in] are unlikely to correspond to serious landslide movement and damage.

(2) In the 10–100-cm [3.94–39.4-in] range, slope deformation may be sufficient to cause serious ground cracking or enough strength loss to result in continuing (post-seismic) failure. Determining whether displacements in this range can be accommodated safely requires good professional judgment that takes into account issues such as landslide geometry and material properties.

(3) Calculated displacements greater than 100 cm [39.4 in] are very likely to correspond to damaging landslide movement, and such slopes should be considered unstable.

The State of California guideline suggests a three-category classification of the seismic stability of a slope as (i) likely stable, (ii) likely unstable, and (iii) definitely unstable. The Newmark displacements calculated by the applicant suggest that the hill slope would fall into the second or third category, depending on which potentially unstable mass is considered, if subjected to ground motion from the design-basis earthquake. An application of the State of California guideline to the displacements calculated by the applicant, therefore, suggests a conclusion that the hill slope above the ISFSI would likely be unstable during a design-basis earthquake.

The applicant acknowledges[10,11] that the toe region of a dislodged rock volume, such as the hypothetical slide masses used in the applicant analyses, would be susceptible to calving and progressive raveling and would, as a result, be a source of seismically generated rockfall hazard. The proposed ISFSI design includes the following rockfall mitigation features, however, to reduce the likelihood of any rockfall impact on the storage casks.

---

[10]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Design Mitigation Features Information to Additional NRC Questions for the Diablo Canyon Independent Spent fuel Storage Installation Application (TAC No. L23399).* Letter (May 6) to U.S. Nuclear Regulatory Commission (DIL–03–007). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[11]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

(1) The tower access road (SAR Figure 2.6-1) provides a horizontal bench 3.0–3.7 m [10–12 ft] wide that is expected to reduce the velocity of any falling rock.

(2) A drainage ditch 0.61–1.83 m [2–6 ft] wide and 0.30 m [1 ft] deep will be constructed at the top of the ISFSI excavation (on the hill slope side). The ditch is expected to stop small rock blocks.

(3) A rockfall barrier fence 2.4–3.0 m [8–10 ft] high will be constructed at the top of the ISFSI cut slope. The fence has an impact capacity of 8.1×10^4 m-kg [295 ft-ton]. The applicant analysis indicates that the fence alone would be adequate to protect the ISFSI from rockfall impact.

(4) The ISFSI cut slope includes a mid-slope horizontal bench 7.6 m [25 ft] wide that is an additional energy dissipation barrier against any rockfall that may break through the barrier fence.

(5) The ISFSI pad is set back from the toe of the cut slope by another horizontal bench 12.5 m [41 ft] wide. This bench is a significant energy-absorption buffer against any rockfall that may break through the barriers enumerated previously.

The staff accepts that the proposed rockfall mitigation would be adequate because of the use of several layers of barriers and energy-dissipation features. The staff, therefore, concludes that the information provided in the SAR regarding the safety of the hill slope with respect to seismically induced instability and rockfall is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR §72.102(d) and §72.122(b).

Stability of Cut Slopes

The three cut slopes designated as Eastcut, Backcut, and Westcut are situated along northeast, southeast, and southwest sides of the ISFSI storage-pad site. These slopes will be excavated in dolomite (unit Tof_b-1), sandstone (unit Tof_b-2), and friable dolomite and sandstone (Tof_b-1a and Tof_b-2a). The dolomite and sandstone are jointed and faulted. The stability of the cut slopes in these rock units was evaluated using kinematic and pseudostatic analyses. The kinematic stability analyses on all three slopes provided potential modes of failure controlled by the discontinuities. The slopes with the potential for wedge failure were further analyzed in detail using a pseudostatic method for stability and design of mitigative measures. The friable or altered rocks are essentially not jointed, hence the kinematic and pseudostatic methodologies were not used for analysis of slopes in this medium. The stability of the slope in the altered rock region, which constitutes a limited area in the Backcut, was not analyzed. The applicant, however, discussed the support measures. The cutslope analysis is supported by other PG&E Calculation packages, and those documents are GEO.DCPP.01.05, GEO.DCPP.01.08, GEO.DCPP.01.16, GEO.DCPP.01.17, GEO.DCPP.01.18, GEO.DCPP.01.20, GEO.DCPP.01.21, GEO.DCPP.01.022, and GEO.DCPP.01.23.

Kinematic Analysis: The kinematic analysis of cut slopes inclined at 70° is presented in PG&E Calculation 52.27.100.732, "Kinematic Stability for Cut Slopes at the DCPP ISFSI Site" (also cited as GEO.DCPP.01.22). The analysis identified three modes of failure typically

associated with hard rock slopes, namely, plane, wedge, and toppling failures. The fracture data (joints, faults, and bedding planes) used in the analysis were collected from rock outcrop, exploratory trenches, and boreholes from each cutslope area. The fracture data were analyzed graphically by plotting, clustering, and contouring of poles on stereonets using DIPS Version 5.0 software. Four discontinuity sets were identified on each cutslope face, and average dip and dip direction for each discontinuity set were determined from the stereographic plots of discontinuity data. The discontinuity sets, local faults, slope geometries, and discontinuity friction angles were used for each mode of failure in kinematic analysis using the DIPS Version 5.0 software. The joint friction angle used in the analysis is based on direct shear tests of clean (rock-to-rock contact) and clay-coated rock joints in dolomite and sandstone as described in PG&E Calculation 52.27.100.728, "Determination of Basic Friction Angle Along Rock Discontinuities of DCPP ISFSI, Based on Laboratory Tests" (also cited as GEO.DCPP.01.18, Revision 2). The post-peak friction angles were evaluated to be 35° for clean and 14° for clay-coated joint samples. The applicant used in the kinematic analyses an average friction angle of 28°, evaluated by aggregating the direct-shear strength data from clean and clay-coated joint samples. The kinematic analyses ascertained moderate to high potential for plane failure and moderate potential for wedge failure for Eastcut. The Backcut has low to moderate potential for plane failure and high potential for wedge failure. The Westcut has high potential for toppling failure and low potential for planar and toppling failure. Eastcut and Backcut slopes were further analyzed for wedge stability and support design. Additional rockfall mitigation measures would be provided, as reviewed in this section under "Seismic Stability of Hill Slope Above the Storage Pad."

Staff reviewed the kinematic stability analysis of cut slopes at the ISFSI pad site. The applicant used a standard approach to identify the major joint sets on each cut slope by stereo plots (Hoek and Bray, 1977). The kinematic analysis is also based on standard methodology to identify plane, wedge, and topping failures using stereographich projection approach (Hoek and Bray, 1977; Goodman, 1983). The average joint friction angle of 28° used in the analysis is within the range of friction angles for dolomite (27–31°) and sandstone (25–31°) given in Hoek and Brown (1980). The applicant has also provided detailed description of the DIPS Version 5.0 software used in the analysis, including verification analysis.

Pseudostatic Analysis: The pseudostatic analysis of potential unstable wedges, identified by kinematic analysis, in Eastcut and Backcut slopes are presented in PG&E Calculation 52.27.100.733, "Psuedostatic Wedge Analysis of DCPP ISFIS Cutslopes, CSWEDGE Analysis" (also cited as GEO.DCPP.01.23). The stability analysis was performed using SWEDGE software, which is based on the limit equilibrium methodology described in Hoek and Bray (1977). Probabilistic and deterministic analyses were conducted on Eastcut and Backcut, considering pore water conditions, seismic forces, tension cracks, and rock anchors. Probabilistic analyses determined the potential unstable wedges with high probability of failure, taking into account uncertainty and variability in discontinuity orientation and strength. The deterministic analyses were conducted on these critical wedges to determine the anchor forces required to achieve stability.

Kinematic analysis identified four potential wedges for Backcut and three potential wedges for Eastcut. The proposed geometry for Backcut consists of two 70° cut slopes with a 7.62-[25-ft]-wide bench. The lower cut slope of height 6.25 m [20.5 ft] and upper cut slope of height 9.69 m [31.8 ft] were modeled separately. Because SWEDGE software cannot model a bench profile, a composite slope height of 15.9 m [52.3 ft] inclined at 47° was also analyzed. The

Eastcut was modeled as a single slope of height 7.17 m [23.5 ft] sloping at 70°. Stability of the slope models was analyzed probabilistically by performing 1,000 Monte Carlo realizations on each model considering normal distribution in discontinuity parameters such as dip, dip direction and friction angles, variation of tension crack distances from the slope face, and inclusion or exclusion of seismic forces and joint saturation.

The dimensions of the wedges are controlled by the length of joints. The block dimensions typically range between 0.6 and 0.9 m [2 and 3 ft] and locally up to a maximum of 14.3 m [4 ft] based on the field measurements of the joint continuity. The maximum depth of wedges in the model was 6.1 m [20 ft]. In probabilistic analysis, the dip and dip direction of the joints, obtained from kinematic analysis, was varied between 5 and 10°. The *in-situ* friction angle of the discontinuities estimated using Barton's equation (Barton and Choubey, 1977) is discussed in Section 2.1.6.5 of the SER. In probabilistic analysis, the friction angle was varied between 16° and 46°. For deterministic analysis, the friction angle was varied between 26 and 31°. The range of friction angles was obtained from the probabilistic analysis.

The horizontal force for pseudostatic stability analysis was determined from a seismic coefficient of 0.5. The approach used to evaluate the seismic coefficient is described in PG&E Calculation package GEO.DCPP.01.05. The applicant used a methodology described in Ashford and Sitar (1994) to estimate design seismic coefficient. Although the information provided in the report is insufficient to permit a detailed independent review, the staff finds that the design seismic coefficient of 0.5g used in the pseudostatic analysis is between 50 and 67 percent of the PGA [0.83g] and consistent with the recommendations in Kramer (1996, p. 436) and NUREG–1620 (NRC, 2002).

A water pressure head equivalent to one-half of the slope height was included in the wedge analysis to account for potential temporal accumulation of infiltrated rain water. In general, the cutslope area is dry, and the measured water table is 65 m [210 ft] below the ISFSI pad site. In addition, installation of a drainage system in the cut slopes has been proposed to prevent accumulation of perched water. Staff agrees that consideration of a water table at half the slope height for cutslope stability analysis is a reasonable assumption. Specific information on the drainage system was not provided in the Diablo Canyon SAR; it is, however, expected that the design would include adequate spacing and depth of weeping holes to drain potential accumulation behind the slope face, which will be lined with shotcrete and wire mesh.

The probability of failure of wedges in Backcut varied between 0 and 1. Probability analysis, conducted on 17 models, showed most wedges were stable under dry joint and nonseismic conditions, but the probabilities were high for the cases analyzed with seismic forces or joint saturation or both. The maximum weight of unstable wedges varied between 18.1 MT [40 kips] and 2,030 MT [4,474.6 kips]. Deterministic analysis was conducted on the unstable slopes to design the rock anchor supports with a target factor of safety 1.3. The rock anchor forces per anchor required to support the wedges ranged from 40 to 151.2 KN [9 to 34 kips] for a support pattern of 1.5 by 1.5 m [5 by 5 ft] placed in a staggered manner and inclined at 15° from the horizontal. Minimum anchor length, not including the bond length, required to penetrate the wedges varied between 1.2 and 7.0 m [4 and 23 ft]. The Eastcut slope was analyzed on 7 models, and the calculated probability of failure varied between 0.12 and 1.0. The maximum weight of the wedges ranged between 10.8 and 15.4 MT [23.8 and 34.0 kips] and the anchor force per anchor required to achieve a factor of safety 1.3 was between 37.6 and 40.0 KN [8.4 and 9.0 kips]. The minimum anchor length in Eastcut was 0.9 m [3 ft]. Staff reviewed the

methodology for anchor design of unstable wedges and finds it consistent with the standard practice. Staff also concurred with the factor of safety of 1.3, considering the recommendation by the U.S. Nuclear Regulatory Commission (1977) used in the analysis, and concluded that the proposed methodology is acceptable.

The methodology used for anchor design is described in PG&E Calculation 52.27.100.718, "Determination of Rock Anchor Design Parameters" (also cited as GEO.DCPP.01.08). The anchor design is based on Post-Tensioning Institute (1996) and the American Society of Civil Engineers (1996). The anchor system consists of 2.54-cm [1-in] diameter Cone-Tech Systems Grade 95H bars inserted in 5.08- to 7.6-cm [2- to 3-in] diameter holes of approximately 9 m [30 ft] deep. The anchors would be installed at 15° inclined to the horizontal with 0.61-m [2-ft] square reinforced concrete bearing pads and spaced at 1.52 m [5 ft] in a staggered pattern. The bond length of the anchor system is 3.1 m [10 ft]. The bond length is greater than computed bond length of 2.04 m [6.7 ft], which is based on a maximum design load of 169.0 KN [38 kips] per anchor from the wedge analysis for the seismic case, a factor of safety of two on the ultimate bond stress between rock and grout, and ultimate bond stress of 1.03 MPa [150 psi] for sandstone. The anchors would be stressed at 191.2 kN [43 kips], which is approximately 0.6 times the ultimate load. Staff evaluated the methodology used in the anchor design and concluded that the proposed methodology is acceptable.

The applicant proposed that the anchor design would be modified as follows for any altered or friable rock zones. The anchor system in altered rock would be improved, based on site conditions, with pressure grouting, increased bond length, reduced anchor spacing, extended anchor length beyond the altered zone, and post grouting in the bonded zones. Staff concluded that reinforcing the slope with anchors and protecting the slope face with wire mesh and shotcrete as proposed by the applicant would be adequate to stabilize the slope in the altered rock region.

The effect of the proposed ISFSI excavation on the stability of the hill slope was included in the applicant evaluation of the hill-slope stability. The staff review of this aspect of the slope stability evaluation is documented in "Static Stability of the Hill Slope Above the Storage Pad."

The staff, therefore, concludes that the information provided in the SAR regarding the stability of the cut slopes is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR §72.102(d) and §72.122(b).

Slope Stability at the Cask Transfer Facility Site

There is no slope stability concern for the CTF site. Any potential instability of the subsurface materials under seismic loading conditions is bounded by the analysis reviewed in Section 2.1.6.4, "Stability of Cask-Storage Pad Foundation," which indicates that the subsurface materials are sufficiently stable to withstand the foundation loading.

Slope Stability Along the Transport Route

The slope-stability concerns along the proposed transport route are (1) the potential encroachment of the Patton Cove landslide toward the transport route, (2) potential rockfall hazard from debris flows and toppling failure of rock blocks from the natural slope above the

transport route, and (3) sliding failure of the subsurface material below the transport route. The applicant provided analyses (SAR Section 2.6.5.4) to address these concerns.

Potential Encroachment of the Patton Cove Landslide: Information provided in the SAR (Section 2.6.1.12, "Landslides;" and Section 2.6.5.4, "Slope Stability Along the Transport Route") and in a subsequent clarification of the information by the applicant (Response to Question 2-17)[12] indicate the following: (1) the Patton Cove slide surface intersects the ground surface at least 30.5 m [100 ft] from the transport route, (2) the slide consists of two slide surfaces constrained by a stable bedrock topography as illustrated in (Figures RAI 2-17-2 and 2-17-3),[13] (3) any continued movement on the lower slide surface will not impact the proposed transport route because the headward growth of the landslide is constrained by a buried sea cliff 67.0 m [220 ft] south of the edge of the proposed transport route, and (4) a headward migration of the upper slide surface is not likely because the geometry of the slide is constrained by a horizontal wave-cut bedrock platform. The applicant also indicated that the landslide will be monitored through inclinometer measurements and visual inspections for cracks and that such monitoring will permit a sufficient warning of any impending landslide hazard prior to its occurrence.

The staff accepts the applicant conclusion that an encroachment of the Patton Cove landslide onto the transport route is unlikely because of the buried bedrock topography. Moreover, applicant monitoring of the landslide, which includes a walk-down inspection of the transport route prior to any movement of the transporter, will provide a timely warning of any impending landslide hazard.

Potential Rockfall Hazard from Debris Flows and Toppling Failure of Rock Blocks: The applicant identified two potential sources of rockfall hazard along the transport route (SAR Section 2.6.1.12, "Landslides;" and Section 2.6.5.4, "Slope Stability Along the Transport Route"). The applicant analysis indicates a potential for toppling failure of isolated rock blocks from the hill slope above the transport route. Also, the applicant identified several colluvial or debris-flow swales above the transport route and concluded that debris flows may develop within such swales during severe weather events. Localized debris accumulations up to 1-m [3-ft] thick may occur on the transport route after such a debris flow.

The applicant identified several existing remedial measures to protect the roadway and the transporter from potential rockfall hazards. The remedial measures are (1) drainage ditches on the inboard edge of the road, (2) graded benches for an abandoned leach field system above a portion of the road, and (3) concrete ditches and culverts. The applicant also indicated that any debris accumulation on the road would be removed prior to any movement of the transporter.

The staff accepts the applicant argument that potential rockfall hazards would be mitigated by the existing remedial measures. Such remedial measures are known to be effective for mitigating rockfall hazards along roadways (e.g., Hoek, 2002). Moreover, the applicant has

---

[12]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Spent Fuel Storage Installation (TAC No. L23399).* Avila, Beach, CA: Pacific Gas and Electric Company. 2002.

[13]Ibid.

committed to conduct a walk-down inspection and to implement any necessary cleanup of the transport route prior to any movement of the transporter.

Stability of the Subsurface Material Below the Transport Route Under Static Loading Conditions: The subsurface materials that form the foundation of the transport route may fail by sliding along vectors pointing approximately west to southwest. Information provided by the applicant (e.g., SAR Figures 2.6-7, 2.6-11, 2.6-16, 2.6-17, and 2.6-19) indicates that the subsurface material consists of Quaternary deposits up to approximately 15–27 m [50–90 ft] thick that overly bedrock along a major portion of the transport route, except at the north end where the transport route lies directly on bedrock (sandstone and dolomite with intercalated clay beds). A part of the transport route near the south end will be constructed on engineered fill placed on the Quaternary deposits.

The stability of the transport route will be affected by the potential for a sliding failure of the subsurface material below the roadway. The potential for such failure occurring while the transporter is on the roadway is of concern to the NRC. The occurrence of any such sliding failure would be controlled by slip on clay beds in the northern part of the road or by deformation of the Quaternary deposits along the remaining parts of the road.

Transport Route on Bedrock Foundation: Information provided by the applicant indicates that the transport route lies directly on bedrock from approximately Station 34+50 {i.e., 1,052 m [3,450 ft] from the south end of the route} to approximately Station 53+50 {i.e., 1,631 m [5,350 ft] from the south end}. The applicant further divided this section of the route into two parts: a southern part, from approximately Station 34+50 to 46+10, in which bedding surfaces dip into the slope; and a northern part, from approximately Stations 46+10 to 53+50, in which the bedding surfaces dip out of the slope. Failure of the slope below the transport route is unlikely in the southern part (Stations 34+50 to 46+10) because sliding on the up-slope dipping clay beds is kinematically impossible. Failure of the slope in the northern part (Stations 46+10 to 53+50) could not be ruled out based on kinematic considerations alone.

The applicant presented stability analysis to make a case that the northern part of the roadway from Stations 46+10 to 53+50 would be stable. The geologic cross section and the clay beds and potential sliding masses considered in the analysis are described in Attachment 5-1, Figures TR–1 through TR–4.[14] The location and attitude of the clay beds were inferred based on information from a borehole and bedding-plane measurements at locations close to the cross section. The slope stability analysis, to assess the stability of the transport route during static loading conditions (including the weight of the transporter), is documented in PG&E Calculation GEO.DCPP.01.28, Revision 3, "Stability and Yield Acceleration Analyses of Potential Sliding Masses Along DCPP ISFSI Transport Route"in Attachment 6-1.[15] The clay-bed and rock-mass strength and unit weight used for the analysis are consistent with information reviewed under "Geotechnical Site Characterization." The transporter loading used in the analysis is consistent with information (weight of empty transporter and a loaded HI-TRAC 125 Transfer Cask) given

---

[14]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[15]Ibid.

in a supporting calculation: ENOVA Engineering Calculation 0104-021-C01, "Seismic Stability Analysis of Transporter on Soil."[16] The minimum safety factor of 2.07 obtained from the analysis in Attachment 5-1, Table 5-1[17] indicates the existence of an adequate stability margin for the slope.

A safety factor in the range of 1.25–1.5 is an acceptable safety margin against failure of a natural slope under static loading conditions (e.g., National Research Council Transportation Research Board, 1978, p. 172). The minimum value of 2.07 obtained from the analysis is larger than the range of acceptable values. Based on these considerations, the staff concludes that the long-term static stability of the subsurface material under the transport route, for sections of the transport route that directly overlie bedrock, would be adequate. The information provided in the SAR regarding the long-term static stability of the transport route foundation where the transport route lies directly on bedrock is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR §72.102(d) and §72.122(b).

Transport Route Underlain By Quaternary Deposits: The applicant provided an evaluation of the long-term stability of the subsurface material below the transport route through static slope-stability analyses of three typical cross sections representing the areas of the transport route underlain by Quaternary deposits in Attachment 6-1, Table 2, and Figure TR–1.[18] The analyses are documented in PG&E Calculation GEO.DCPP.01.28, Revision 3, "Stability and Yield Acceleration Analysis of Potential Sliding Masses Along DCPP ISFSI Transport Route" in Attachment 6-1.[19]

The material properties used for the analyses (shear strength and density) were taken from Appendix 2.5C of Volume III of Units 1 and 2 Diablo Canyon Site FSAR, Attachment 6-1 (Table 1 and Attachment B).[20]

The transporter loading was represented as an equivalent line load of approximately 25,538 N/m [1,750 lb/ft] applied uniformly over the transporter footprint. This loading is consistent with information (weight of empty transporter and a loaded HI-TRAC 125 Transfer

---

[16]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Spent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

[17]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[18]Ibid.

[19]Ibid.

[20]Ibid.

Cask) given in a supporting calculation: ENOVA Engineering Calculation 0104–021–C01, "Seismic Stability Analysis of Transporter on Soil."[21]

The calculated factors of safety indicate that there would be a sufficient safety margin against sliding failure of the subsurface material below the transport route during static loading conditions (including the transporter loading), for the sections of the transport route that are underlain by Quaternary deposits. The staff concludes that the long-term static stability of the subsurface material under the transport route, for sections of the transport route underlain by Quaternary deposits, would be adequate. The information provided in the SAR regarding the long-term static stability of the transport route foundation where the transport route lies on Quaternary deposits is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR §72.102(d) and §72.122(b).

Seismic Stability of the Subsurface Material Below the Transport Route: The applicant provided an analysis of the effect of a potential earthquake on the stability of the subsurface materials below the transport route, considering an occurrence of such an earthquake while the transporter is on the transport route. The analysis was performed using four geologic cross sections in Attachment 6-1, Figure TR–1.[22] One cross section represents areas of the transport route supported directly on bedrock, and the other three represent areas of the transport route underlain by Quaternary deposits. The analysis is based on calculating the seismically induced displacements following an approach proposed by Newmark (1965).

As described previously in "Seismic Stability of the Hill Slope Above the Storage Pad," the Newmark approach consists of three key steps: (1) the calculation of the yield acceleration, k_y, (2) the calculation of the average acceleration time history, a_{ht}; and (3) the double integration of $a_{ht} - k_y$ for $a_{ht} > k_y$ to obtain the Newmark displacement.

The calculation of k_y is documented in PG&E Calculation GEO.DCPP.01.28, Revision 3, "Stability and Yield Acceleration Analysis of Potential Sliding Masses Along DCPP ISFSI Transport Route" in Attachment 6-1.[23] This calculation also documents the analysis of the long-term stability of the transport route foundation as described previously in "Stability of the Subsurface Material Below the Transport Route Under Static Loading Conditions." As discussed in that section, the analysis was performed using standard techniques and appropriate values for the transporter loading and for the strength and unit weight of the bedrock and Quaternary deposit materials. The staff, therefore, concludes that the values of k_y

---

[21]Womack, L.F. "Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Spent Fuel Storage Installation (TAC No. L23399)." Avila Beach, CA: Pacific Gas Electric Company. 2002.

[22]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[23]Ibid.

calculated from the analysis in Attachment 6-1, Table 2[24] are appropriate for conducting an assessment of the seismic stability of the transport route.

The calculation of a_{ht} is documented in PG&E Calculation GEO.DCPP.01.29, Revision 3, "Determination of Seismic Coefficient Time Histories for Potential Sliding Masses Along DCPP ISFSI Transport Route" in Attachment 7-2.[25] The calculation was performed through a dynamic finite element analysis using a two-dimensional finite element model of each slope cross section. The base of each model {at a depth of approximately 91.4 m [300 ft] below the toe of the slope} was subjected to an input acceleration time history. The base input time history was calculated from a deconvolution of the design ground motion using a one-dimensional site response analysis code. The applicant verified in PG&E Calculation 52.27.100.744, "Verification of Computer Code QUAD4M" (also cited as GEO.DCPP.01.34) that the free-field ground motion calculated from the finite element model matches the design ground motion satisfactorily. The finite element analysis and the approach used to calculate the input acceleration time history for the finite element model are consistent with standard practice (cf. Ofoegbu and Gute, 2002). The applicant calculated a_{ht} using two methods: first, by averaging the acceleration time history at selected nodal points in each finite element model; and second, by using the ratio F_t/M where F_t is the resultant down-slope force along the potential failure surface and M is the potentially unstable mass. The second approach used by the applicant is the standard approach for calculating the seismic coefficient time history for a deformable medium (e.g., Kramer, 1996, p. 446). The values of a_{ht} calculated using the second approach were used by the applicant to calculate potential seismically induced displacements as explained presently. Staff, therefore, concludes that the values of a_{ht} calculated from the analysis in Attachment 7-2, Figure 18[26] are appropriate for conducting an assessment of the seismic stability of the transport route.

The calculation of Newmark displacements through a double integration of $a_{ht} - k_y$ for $a_{ht} > k_y$ is documented in PG&E Calculation GEO.DCPP.01.30, Revision 3, "Determination of Potential Earthquake-Induced Displacements of Potential Sliding Masses Along DCPP ISFSI Transport Route (Newmark Analysis)" in Attachment 7-1.[27] The calculated Newmark displacements generally lie in the range of 15.2–45.7 cm [0.5–1.5 ft] in Attachment 7-1, Table 2.[28] The calculated displacements should be interpreted as order-of-magnitude estimates of the potential deformation of the transport-route foundation when subjected to the design-basis earthquake and the transporter loading simultaneously. The results indicate that a deformation on the order of tens of centimeters would occur under such conditions. The direction of such deformation would be downward with a lateral component normal to the roadway. The potential implications

---

[24]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Slope Stability Responses to Additional NRC Questions for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter (March 27) to U.S. Nuclear Regulatory Commission (DIL–03–004). Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[25]Ibid.

[26]Ibid.

[27]Ibid.

[28]Ibid.

of such deformation for the stability of the transporter during a potential seismic event are reviewed in Section 15.1.2.6, "Earthquake," of this SER.

The staff concludes that the applicant evaluation of the seismic stability of the transport route foundation is based on standard methods. Therefore, the information presented is adequate for use in other sections of the SAR to perform additional safety analyses and demonstrate compliance with regulatory requirements in 10 CFR §72.90(a), §72.90(b), §72.90(c), §72.90(d), §72.92(a), §72.92(b), §72.92(c), §72.102(c), §72.102(d), and §72.122(b).

Staff Evaluation

The staff has reviewed Section 2.6.5, "Slope Stability," of the SAR and concludes that the information presented in this section is adequate for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR §72.102(c) and §72.102(d).

2.2 Evaluation Findings

The staff has reviewed the site characteristics presented in the SAR and finds that the SAR provides an acceptable description and safety assessment of the site on which the Diablo Canyon ISFSI Facility is to be located, in accordance with 10 CFR §72.24(a). Staff also finds that the proposed site complies with the criteria of 10 CFR Part 72 Subpart E, as required by 10 CFR §72.40(a)(2).

2.3 References

Abramson, L.W., T.S. Lee, S. Sharma, and G.M. Boyce. *Slope Stability and Stabilization Methods*. 2nd Edition. New York City, NY: John Wiley and Sons, Inc. 2002.

Ambraseys, N.N. and J.M. Menu. "Earthquake-Induced Ground Displacements." *Earthquake Engineering and Structural Dynamics*. Vol. 16. pp. 985–1,006. 1988.

American Society of Civil Engineers. "Rock Foundations." Adopted from U.S. Army Corps Manual EM 1,110–1-2908. New York City, NY: American Society of Civil Engineers Press. 1996.

Ashford, S. and N. Sitar. "Seismic Response of Steep Natural Slopes." Report No. UCB/EERC–94/05, Earthquake Engineering Research Center, University of California, Berkeley. Berkeley, CA: Earthquake Engineering Research Center. May 1994.

Barton, N.R. and V. Choubey. "The Shear Strength of Rock Joints in Theory and Practice." *Rock Mechanics*. Vol 10, No. 1-2. pp. 1–55. 1977.

Gawthrop, W.H. "Seismicity of the Central California Coastal Region." U.S. Geological Survey Open File Report 75-134. 1975.

Goodman, R.E. *Introduction to Rock Mechanics*. New York City, NY: John Wiley and Sons. 1983.

Hoek, E. and J.W. Bray. *Rock Slope Engineering.* London, England: Institution of Mining and Metallurgy. 1977.

Hoek, E. "Shear Strength Discontinuities." *Chapter 4: Rock Engineering Course Notes.* Online document <http://www.rocscience.com/roc/Hoek/Hoeknotes. 2000a.

Hoek, E. "Rock Mass Properties." *Chapter 11: Rock Engineering Course Notes.* Online document <http://www.rocscience.com/roc/Hoek/Hoeknotes. 2000b.

Hoek, E. "Practical Rock Engineering." *Chapter 9: Analysis of Rockfall Hazards.* http://www.rocscience.com/roc/Hoek/Hoeknotes2000.htm. 2002.

Hoek, E. and E.T. Brown. *Underground Excavation in Rock.* London, England: Institution of Mining and Metallurgy. 1980.

Hoek, E., C. Carranza-Torres, and B. Corkum. *Hoek-Brown Failure Criterion: 2002 Edition.* Proceedings of the NARMS-TAC 2002, 5th North American Rock Mechanics Symposium and 17th Tunneling Association of Canada Conference, Toronto, Canada, July 7–10, 2002. Vol. 1. R. Hammah, W. Bawden, J. Curran and M. Telesnicki, eds. Toronto, Canada: University of Toronto. pp. 267–271. 2002.

Jibson, R.W. "Predicting Earthquake-Induced Landslide Displacements Using Newmark's Sliding Block Analysis." Transportation Research Record 1411. Washington, DC: Transportation Research Board. pp. 9–17. 1994.

Kramer, S.L. *Geotechnical Earthquake Engineering.* Upper Saddle River, NJ: Prentice Hall. 1996.

Lettis, W.R., K.L. Hanson, J.R. Unruh. M. McLaren, and W.U. Savage. *Quaternary Tectonic Setting of South-Central Coastal California.* Evolution of Sedimentary Basins/Onshore Oil and Gas Investigations-Santa Maria Province. M.A. Keller, ed. *U.S. Geological Survey Bulletin.* In press.

McIntosh, K.D., D.L. Reed, E.A. Silver, and A.S. Meltzer. *Deep Structure and Structural Inversion Along the Central California Continental Margin from Edge Seismic Profile RU–3. Journal of Geophysical Research.* Vol. 96. pp. 6,459–6,473. 1991.

McLaren, M.K. and W.U. Savage. *Seismicity of South-Central Coastal California: October 1987 Through January 1997. Bulletin of the Seismological Society of America.* Vol. 91. pp. 1,629–1,658. 2001.

National Research Council Transportation Research Board. *Landslides Analysis and Control.* Special Report 176. Washington, DC: National Academy of Sciences. 1978.

Newmark, N.M. *Effects of Earthquakes on Dams and Embankments. Geotechnique.* Vol. 15, No. 2. pp. 139–160. 1965.

Ofoegbu, G.I. and G.D. Gute. *Dynamic Soil-Structure Interaction Analysis of a Storage-Cask Foundation Design*. CNWRA 2003-03. San Antonio, TX: Center for Nuclear Waste Regulatory Analyses. 2002.

Pacific Gas and Electric Company. *Final Report of the Diablo Canyon Long Term Seismic Program*. Docket Nos. 50-275, OL–DPR–80 and 50-323, and OL–DPR–82. Avila Beach, CA: Pacific Gas and Electric Company. July 1988.

Pacific Gas and Electric Company. *Addendum to the 1988 Final Report of the Diablo Canyon Long Term Seismic Program*. Docket Nos. 50-275, OL–DPR–80, 50-323, and OL-DPR-82. Avila Beach, CA: Pacific Gas and Electric Company. 1991.

Pacific Gas and Electric Company. *Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update*. Revision 14. Avila Beach, CA: Pacific Gas and Electric Company. November 2001.

Pacific Gas and Electric Company. *Diablo Canyon ISFSI Safety Analysis Report*. Amendment 1. Docket No. 72-26. Avila Beach, CA: Pacific Gas and Electric Company. October 2002.

Post-Tensioning Institute. *Recommendations for Prestressed Rock and Soil Anchors*. Phoenix, AZ: Post-Tensioning Institute. 1996.

Seed, H.B. "Considerations in the Earthquake-Resistant Design of Earth and Rockfill Dams." *Geotechnique*. Vol. 29, No. 3. pp. 215–263. 1979.

State of California Division of Mines and Geology. *Guidelines for Evaluating and Mitigating Seismic Hazards in California. Chapter 5: Analysis and Mitigation of Earthquake-Induced Landslide Hazards*. Special Publication 117. Sacramento, CA: State of California Division of Mines and Geology. 1997.

Terzaghi, K, R.B. Peck, and G. Mesri. *Soil Mechanics in Engineering Practice*. 3rd Edition. New York City, NY: John Wiley and Sons, Inc. 1996.

U.S. Nuclear Regulatory Commission. *Meteorological Programs*. Regulatory Guide 1.23. Washington, DC: U.S. Nuclear Regulatory Commission. 1976.

U.S. Nuclear Regulatory Commission. *Design, Construction, and Inspection of Embankment Retention Systems for Uranium Mills*. Regulatory Guide 3.11. Washington, DC: U.S. Nuclear Regulatory Commission. 1977.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*. NUREG–0800, Rev 1. Washington, DC: U.S. Nuclear Regulatory Commission. 1981.

U.S. Nuclear Regulatory Commission. *Safety Evaluation Report for Diablo Canyon Power Plant, Units 1 and 2*. Supplement Number 34 to NUREG–0675. Washington DC: U.S. Nuclear Regulatory Commission. June 1991.

U.S. Nuclear Regulatory Commission. *Draft Standard Review Plan for the Review of a Reclamation Plan for Mill Trailings Sties Under Title II of the Uranium Mill Tailings Radiation Control Act.* NUREG–1620. Washington, DC: U.S. Nuclear Regulatory Commission. January 2002.

Wilson R.C. and D.K. Keefer. "Dynamic Analysis of a Slope Failure from the 6 August 1979 Coyote Lake, California, Earthquake." *Bulletin of the Seismological Society of America.* Vol. 73, No. 3. pp. 863–877. 1983.

3 OPERATION SYSTEMS

3.1 Conduct of Review

The objective of the operation system review was to evaluate the description presented in the Safety Analysis Report (SAR) (Pacific Gas and Electric Company, 2002) of all operations, including systems, equipment, and instrumentation and to determine if they fulfill the U.S. Nuclear Regulatory Commission (NRC) regulatory requirements for clarity and completeness. Particular emphasis was placed on how operation systems relate to handling and storage of spent nuclear fuel, confinement of nuclear material, and management of expected and potential radiological dose. The review of the operation systems included selected sections of Chapter 3, "Principal Design Criteria," Chapter 4, "ISFSI Design," Chapter 5, "ISFSI Operations," and Chapter 8, "Accident Analysis" of the SAR, and documents cited in the SAR. The review did not include operations within the Diablo Canyon Power Plant (DCPP) Fuel Handling Building and Auxiliary Building (FHB/AB) that are covered in accordance with the 10 CFR Part 50 license for the DCPP.

3.1.1 Operation Description

The description of the operating system was reviewed for conformance with the following regulations:

- 10 CFR §72.24(b) requires a description and discussion of the ISFSI structures with special attention to design and operating characteristics, and principal safety considerations.
- 10 CFR §72.40(a)(5) requires that the proposed operating procedures are adequate to protect health and to minimize danger to life or property.
- 10 CFR §72.40(a)(13) require that the proposed activities can be conducted without endangering the health and safety of the public.
- 10 CFR §72.104(b) and (c) requires the operational restrictions and limits to meet as low as reasonably achievable objectives for radioactive materials in effluent and direct radiation levels associated with ISFSI operations.
- 10 CFR §72.122(f) requires that systems and components must be designed to permit inspection, maintenance and testing.
- 10 CFR §72.122(i) requires that instrumentation systems for dry storage casks be provided in accordance with cask design requirements to monitor conditions that are classified as important to safety over anticipated ranges for normal conditions and off-normal conditions.
- 10 CFR §72.126(b) requires that radiological alarm systems be provided in accessible work area as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given set point.

- 10 CFR §72.126(c) requires means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions and means to measure direct radiation levels in and around the area.

- 10 CFR §72.128(a)(1) requires that the spent fuel waste storage systems be designed with capability to test and monitor components important to safety.

In SAR Section 3.1.2, the applicant describes the general operating functions to be performed in preparing and storing spent nuclear fuel in the HI-STORM 100 System. Additional details are provided in SAR Chapters 3, 4, 5, and 8.

The operations to be performed at the independent spent fuel storage installation (ISFSI) in accordance with 10 CFR Part 72 include the receipt and inspection of incoming empty storage casks. The HI-STORM 100 System storage casks and multipurpose canisters (MPCs) will arrive at the ISFSI site and will be visually inspected. Personnel will then transfer the storage cask to the Cask Transfer Facility (CTF). The empty MPCs will be transferred to the DCPP FHB/AB for fuel loading operations covered by the 10 CFR Part 50 license. The empty MPC will be installed in the HI-TRAC 125 Transfer Cask, inserted into the spent nuclear fuel pool, and loaded with spent nuclear fuel. The transfer activities will use a combination of fixtures and equipment designed by the cask system vendor and equipment specifically designed for the DCPP. After the insertion of fuel, the MPC closure welding and draining, drying, and helium backfilling operations will be performed within the FHB/AB. The MPC and transfer cask will then be placed inside the cask transport frame, rotated to a horizontal position, and removed from the FHB/AB. Subsequent operations include the transport of the MPC in the HI-TRAC 125 Transfer Cask from the DCPP FHB/AB to the CTF, transfer of the MPC containing the spent nuclear fuel from the transfer cask to the storage cask, placement of the loaded storage casks on the storage pads, surveillance of the storage casks, maintenance of the health physics conditions consistent with as low as reasonably achievable (ALARA) requirements and ISFSI technical specifications, maintenance of the ISFSI and storage casks, and inventory documentation management. Ultimately, the spent nuclear fuel casks will be removed from the ISFSI, and the facility will be decommissioned.

The SAR describes in detail the activities that will be performed to ensure that the stored casks do not endanger public health and safety. In summary, these activities include the following actions. After the storage casks are placed on the storage pad, as specified in the Technical Specifications the casks are inspected periodically to ensure that the air vents are not blocked. Security personnel control access to the storage area and identify and assess off-normal and emergency events. Health physics personnel perform dose rate and contamination surveys to ensure that the Technical Specification limits are maintained. Maintenance personnel maintain the facilities including the storage casks, emergency equipment, and transport systems.

The staff reviewed the operating functions described in SAR Chapters 3, 4, 5, and 8 to ensure that the applicant adequately described the appropriate procedures, equipment, and personnel requirements. The SAR identifies the specific equipment and the personnel to accomplish the transfer, storage, and retrieval of the casks in compliance with 10 CFR §72.24(b). The staff determined that the detailed procedure descriptions for operating, inspecting, and testing are consistent with the operation system in compliance with 10 CFR §72.122(f).

The staff found the general description of the proposed ISFSI operations to be adequate. Diablo Canyon ISFSI operations can be conducted without endangering the health and safety of the public and are, therefore, in compliance with 10 CFR §72.40(a)(5) and 10 CFR §72.40(a)(13). Additionally, the SAR provides acceptable descriptions and discussions of the projected operating characteristics and safety considerations as required by 10 CFR §72.122(i). Staff found that the design and procedures provide acceptable capability to test and monitor components important to safety, in compliance with 10 CFR §72.128(a)(1).

The applicant ALARA considerations are reviewed in Chapter 11 of this Safety Evaluation Report (SER). Based on this review, staff found that the design and operations consider ALARA, as required by 10 CFR §72.104(b) and §72.104(c). Direct radiation monitoring is also considered in the design, in compliance with the requirements of 10 CFR §72.126(b) and §72.126(c).

3.1.2 Spent Nuclear Fuel Handling Systems

The description of the spent nuclear fuel handling systems was reviewed for conformance with the following regulations:

- 10 CFR §72.24(b) requires a description and discussion of the ISFSI structures with special attention to design and operating characteristics, and principal safety considerations.

- 10 CFR §72.104(b) and (c) requires the operational restrictions and limits to meet as low as reasonably achievable objectives for radioactive materials in effluent and direct radiation levels associated with ISFSI operations.

- 10 CFR §72.128(a) requires that the spent fuel waste storage and handling systems must be designed with: a capability to test and monitor components important to safety, suitable shielding under normal and accident conditions, confinement structures and systems, and heat removal capability.

Handling of the HI-STORM 100 System, including the MPC, is described in detail in the HI-STORM 100 System Final Safety Analysis Report (FSAR) (Holtec International, 2000), which the staff has previously reviewed and found acceptable (U.S. Nuclear Regulatory Commission, 2002a,b). Handling operations at the Diablo Canyon ISFSI Facility will be consistent with the handling operations described in the HI-STORM 100 System FSAR.

3.1.3 Other Operating Systems

The description of the other operating systems were reviewed for conformance with the following regulations:

- 10 CFR §72.104(b) and (c) requires the operational restrictions and limits to meet as low as reasonably achievable objectives for radioactive materials in effluent and direct radiation levels associated with ISFSI operations.

- 10 CFR §72.122(k)(2) requires that emergency utility services be designed to permit testing of each system for the transfer between the normal and emergency power supply and to permit the operation of associated safety systems.

- 10 CFR §72.122(k)(3) requires that proposed design of the ISFSI include provisions so that emergency power is provided to maintain safe storage condition and permit continued functioning of all systems essential to safe storage.

- 10 CFR §72.126(b) requires that radiological alarm systems be provided in accessible work area as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given set point.

- 10 CFR §72.126(c) requires means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions and means to measure direct radiation levels in and around the area.

In the SAR, the applicant discusses the structures, systems, and components (SSC) (i.e., security systems, cask transporter, fire protection systems, and radiation monitoring systems) classified as not important to safety, but having security or operational importance. The SAR states that the design of the SSC classified as not important to safety comply with applicable codes and standards. Further, the SAR states that the SSC classified as not important to safety will be compatible with SSC classified as important to safety and be designed to a level of quality to ensure that they will mitigate the effects of off-normal or accident-level events, as required.

The staff reviewed the description of the other operating systems described in the SAR. The applicant ALARA considerations are reviewed in Chapter 11 of this SER. Based on this review, the staff found that the design and operations consider ALARA as required by 10 CFR §72.104(b). Radiological alarm systems and direct radiation monitoring are considered in the design, in compliance with the requirements of 10 CFR §72.126(b) and §72.126(c).

The proposed design of the ISFSI does not require utility systems during spent nuclear fuel storage. Therefore, the emergency utility services required by 10 CFR §72.122(k)(2) are not applicable. The proposed design of the ISFSI does not include systems and subsystems that require continuous electric power to permit continued functioning. Because the design of the ISFSI does not require emergency power, 10 CFR §72.122(k)(3) is also not applicable.

3.1.4 Operation Support Systems

The descriptions of the operation support systems were reviewed for conformance with the following regulations:

- 10 CFR §72.122(i) requires that instrumentation systems for dry storage casks must be provided in accordance with cask design requirements to monitor conditions that are classified as important to safety over anticipated ranges for normal conditions and off-normal conditions.

- 10 CFR §72.122(k)(1) requires that each utility service system be designed to meet emergency conditions and include redundant systems to maintain the ability to perform safety functions assuming a single failure.

- 10 CFR §72.122(k)(3) requires that the proposed design of the ISFSI include provisions so that emergency power is provided to maintain safe storage conditions and permit continued functioning of all systems essential to safe storage.

The operation of the ISFSI is passive and self-contained. These storage casks do not require any instrumentation or control systems to ensure safe operation when they are placed into storage.

During operation of the ISFSI, however, the storage casks will be inspected periodically. These inspections will provide a means to assess the performance of the storage casks. The proposed design of the ISFSI does not require utility systems during spent nuclear fuel storage. As stated previously, the proposed design of the ISFSI does not include systems and subsystems that require continuous electric power to permit continued functioning, and the design of the ISFSI does not require emergency power.

The staff reviewed the proposed operation support systems described in the SAR. In addition, the staff evaluated the appropriate sections in the SAR that identify the SSC important to safety. The staff agrees that instrumentation systems to be used to periodically monitor the Diablo Canyon ISFSI Facility are appropriately classified as not important to safety; therefore, 10 CFR §72.122(i) is not applicable. The staff found that the proposed self-contained, passive storage facility requires no permanently installed auxiliary systems. All auxiliary systems required to support loading and off-loading of the system, periodic monitoring, and maintenance are designed to be portable systems. The systems are not important to safety, and therefore 10 CFR §72.122(k)(1) is not applicable. Additionally, as stated previously, the requirements of 10 CFR §72.122(k)(3) are not applicable because the design of the Diablo Canyon ISFSI Facility does not require emergency power for systems essential to safe storage, and there are no systems essential to safe storage requiring electrical power.

3.1.5 Control Room and Control Area

The descriptions of the control room and control area were reviewed for conformance with the following regulation:

- 10 CFR §72.122(j) requires that, if appropriate, a control room or control area must be designed to permit occupancy and actions to be taken to monitor the ISFSI under normal conditions and provide safe control under off-normal and accident conditions.

The storage casks are passive storage systems. A control room and control area are not necessary to maintain the conditions required for safe operation of the ISFSI, store spent nuclear fuel safely, prevent damage to the spent nuclear fuel during handling and storage, or provide reasonable assurance that the spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

The staff reviewed the control room and control areas described in the SAR. In addition, the staff has evaluated sections pertaining to monitoring instruments, limits, and controls of the proposed cask systems from the SAR. The staff found that the control room and control area are not important to safety. The ISFSI is a self-contained, passive storage facility that requires no permanent control room or control area to ensure safe operation; therefore, the requirements of 10 CFR §72.122(j) are not applicable.

3.1.6 Analytical Sampling

The description of the operating system was reviewed for conformance with the following regulations:

- 10 CFR §72.24(b) requires a description and discussion of the ISFSI structures with special attention to design and operating characteristics, and principal safety considerations.

As discussed in the SAR, no analytical sampling is required. The HI-STORM 100 System design will preclude release of effluents for normal, off-normal, and accident conditions during storage.

3.1.7 Pool and Pool Facility Systems

The pool and pool facility systems were reviewed for conformance with the following regulations:

- 10 CFR §72.24(b) requires a description and discussion of the ISFSI structures with special attention to design and operating characteristics, and principal safety considerations.

The Diablo Canyon ISFSI utilizes the dry-cask storage technology, which houses spent nuclear fuel inside sealed, inerted canisters rather than in a spent nuclear fuel pool. Therefore, neither the use of a pool nor any system supporting a pool is incorporated into the Diablo Canyon ISFSI as covered by the 10 CFR Part 72 SAR. Note that the spent nuclear fuel is transferred from the DCPP pool facility into the storage canisters within the confines of the DCPP. Activities associated with this operation are controlled under the 10 CFR Part 50 license.

3.2 Evaluation Findings

The staff found that the description of the proposed operating procedures and systems are adequate. Diablo Canyon ISFSI operations meet the regulatory requirements and can be conducted without endangering the health and safety of the public.

3.3 References

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System).* Vols I and II. HI–2002444. Docket No. 72-1014. Marlton, NJ: Holtec International. 2000.

Pacific Gas and Electric Company. *Diablo Canyon ISFSI Safety Analysis Report*. Amendment 1. Docket No. 72-26. Avila Beach, CA: Pacific Gas and Electric Company. October 2002.

U.S. Nuclear Regulatory Commission. *10 CFR Part 72 Certificate of Compliance No. 1014, Amendment 1, for the HI-STORM 100 Cask System*. Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. July 15, 2002a.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report*. Amendment 1. Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2002b.

4 STRUCTURES, SYSTEMS, AND COMPONENTS AND DESIGN CRITERIA EVALUATION

4.1 Conduct of Review

Safety Analysis Report (SAR) Section 3.1.1 identifies spent fuel, consisting of both Westinghouse LOPAR and VANTAGE 5 assemblies, as the material to be stored. SAR Section 4.5 categorizes all structures, systems, and components (SSCs) as either important to safety or not important to safety. Those SSCs important to safety are designed for safe confinement and storage of the spent fuel without the release of radioactive material. SAR Chapter 3 also identifies the principal design criteria for the Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI). These design criteria are derived from the requirements of 10 CFR Part 72 and applicable industry codes and standards. 10 CFR Part 72 also identifies the general design criteria for SSCs classified as important to safety with respect to withstanding the effects of environmental conditions and natural phenomena. The worst-case loads for normal, off-normal, and accident conditions are identified. SAR Tables 3.4-1 through 3.4-5 provide a summary of the key design criteria for the Diablo Canyon ISFSI. The design criteria are compared to the actual design in subsequent chapters of this Safety Evaluation Report (SER).

The storage system to be used at the Diablo Canyon ISFSI is the HI-STORM 100 System as described in the HI-STORM 100 System Final Safety Analysis Report (FSAR) (Holtec International, 2002). The HI-STORM 100 System has been approved by the U.S. Nuclear Regulatory Commission (NRC) for general use under Certificate of Compliance (CoC) No. 1014 (U.S. Nuclear Regulatory Commission, 2002a). Where applicable, the staff relied on the review carried out during the certification process of the cask system, as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b).

4.1.1 Materials to be Stored

As identified in SAR Section 3.1.1, "Materials to be Stored," the materials to be stored at the Diablo Canyon ISFSI are Westinghouse LOPAR and VANTAGE 5 spent fuel assemblies, damaged fuel, and debris that are approved for storage in the HI-STORM 100 System. The approved contents are specified in Appendix B of CoC No. 1014. SAR Section 3.1.1 provides a brief discussion of the materials to be stored at the Diablo Canyon ISFSI. SAR Tables 3.1-1 and 3.1-2 are summaries of the fuel physical and thermal and radiological characteristics. This discussion is consistent with the physical, thermal, and radiological characteristics of the spent fuel in the HI-STORM 100 System FSAR (Holtec International, 2002).

4.1.2 Classification of Structures, Systems, and Components

This section contains a review of SAR Section 4.5, "Classification of Structures, Systems, and Components." The staff reviewed the discussion on classifications of SSCs with respect to the following regulatory requirements:

- 10 CFR §72.120(a) requires that an application to store spent fuel waste in an ISFSI must include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing,

maintenance and performance requirements for structures, systems, and components important to safety as defined in 10 CFR §72.3.

- 10 CFR §72.144(a) requires that the licensee establish, at the earliest practicable time consistent with the schedule for accomplishing the activities, a quality assurance program that complies with the requirements of this subpart. The licensee shall document the quality assurance program by written procedures or instructions and shall carry out the program in accordance with these procedures throughout the period during which the ISFSI is licensed. The licensee shall identify the structures, systems, and components to be covered by the quality assurance program, the major organizations participating in the program, and the designated functions of these organizations.

In 10 CFR §72.3, SSCs are defined as items whose functions are to (1) maintain the conditions required to store spent fuel safely; (2) prevent damage to the spent fuel container during handling and storage; and (3) provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public. The SAR lists the SSCs based on this definition as required by 10 CFR §72.24(a) and §72.144(a).

SAR Section 4.5, "Classification of Structures, Systems, and Components," identifies safety protection systems and provides a brief description of the important characteristics of each system. The classification consists of two levels—important to safety and not important to safety. The important to safety classification contains three categories based on the potential impact to safe operation:

(1) **Classification Category A—Critical to Safe Operation**, whose failure or malfunction could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control.

(2) **Classification Category B—Major Impact on Safety**, whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of a Category B item, in conjunction with failure of an additional item, could result in an unsafe condition.

(3) **Classification Category C—Minor Impact on Safety**, whose failure or malfunction would not be likely to create a situation adversely affecting public health and safety.

4.1.2.1 Classification of Structures, Systems, and Components—Items Important to Safety

Those SSCs considered important to safety are identified in SAR Section 4.5, "Classification of Structures, Systems, and Components."

Category A Structures, Systems, and Components

The SSCs classified as Category A are given in SER Table 4-1. The failure of a single Category A item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control. In some cases, the items are classified as Category A because they are considered part of the single-failure-proof load path of systems used to handle and transport the spent fuel. Each of these items is designed to protect the spent fuel during specific phases of handling and storage. The following Category A components have been properly classified.

As identified in SAR Section 4.5.1, "Spent Fuel Storage Cask Components," the multipurpose canister (MPC) serves as the primary confinement structure. Its failure could lead to the release of radioactive material. Sufficient description of the MPC is provided in SAR Section 4.2.3 "Storage Cask Description."

As identified in SAR Section 4.5.1, "Spent Fuel Storage Cask Components," the fuel basket and damaged fuel container maintain fuel in a safe geometry. Failure could lead to criticality and reduced ability to retrieve the fuel. Sufficient descriptions of the fuel basket and damaged fuel container are provided in SAR Section 4.2.3, "Storage Cask Description."

As identified in SAR Section 4.5.1, "Spent Fuel Storage Cask Components," the HI-TRAC 125 Transfer Cask protects the MPC during handling for 10 CFR Part 50 and Part 72 operations and is part of the single-failure-proof load path. Sufficient description of the transfer cask is provided in SAR Section 4.2.3.2.4, "HI-TRAC 125 Transfer Cask."

The transfer cask lift links, MPC downloader slings, MPC lift cleats, HI-STORM 100 System lifting brackets, and HI-STORM 100 System lift links are part of the single-failure-proof load path. Sufficient descriptions of the associated lifting devices are provided in SAR Sections 4.3.2.5 through 4.3.2.9.

As identified in SAR Section 4.5.1, "Spent Fuel Storage Cask Components," the HI-STORM mating device bolts and shielding frame provide structural support and shielding at the interface between the top of the open overpack and the bottom of the transfer cask during MPC transfer operations. Sufficient description of these components is provided in SAR Section 3.3.4.2.5, "HI-STORM Mating Device."

The first 10 items in Table 4-1, those that correspond to the HI-STORM 100 System, are identified in the previous five paragraphs. Details associated with these 10 items are cask-specific, and additional descriptions are presented in the HI-STORM 100 System FSAR (Holtec International, 2002).

As identified in SAR Section 4.5.4, "Cask Transport System," the cask transporter load-bearing components prevent damage to the spent fuel and spent fuel storage cask system components during transport, lifting, and MPC transfer operations in all normal, off-normal, and accident conditions. The cask transporter will also be designed to preclude tip-over under site-specific seismic, tornado winds, and tornado missile loads. Components of the transporter are part of the single-failure-proof load path. Sufficient description of the cast transporter is provided in SAR Section 4.3.2.1, "Cask Transporter."

Table 4-1. Category A quality assurance classification of SSCs (Based on SAR Table 4.5-1)

Structures, Systems, and Components	Logic
Multi-Purpose Canister (MPC) (HI-STORM)	Serves as the primary confinement structure for the spent nuclear fuel assemblies and is designed to remain intact under all accident conditions analyzed. It provides confinement, criticality control, heat transfer capability, and radiation shielding.
Fuel Basket (HI-STORM)	Ensures the correct geometry of the stored fuel assemblies and provides the fixed neutron absorber to prevent criticality.
Damaged Fuel Container (HI-STORM)	Maintains damaged fuel or fuel debris in a safe geometry and enables retrieval.
HI-TRAC 125 Transfer Cask (HI-STORM)	Designed to support the canister during transfer lift operations and provide radiation shielding and canister heat rejection.
Transfer Cask Lift Links (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask and preclude the accidental drop of a canister.
MPC Downloader Slings (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the MPC and preclude the accidental drop of a MPC.
MPC Lift Cleats (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the MPC and preclude the accidental drop of a MPC.
HI-STORM Lifting Brackets (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask and preclude the accidental drop of a canister.
HI-STORM Lift Links (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask and preclude the accidental drop of a canister.
HI-STORM Mating Device Bolts and Shielding Frame (HI-STORM)	Provides structural support and shielding at the interface between the top of the open overpack and the bottom of the transfer cask during MPC transfer operations at Cask Transfer Facility (CTF).
Cask Transporter	The load-bearing components prevent damage to the spent nuclear fuel and spent nuclear fuel storage cask system components during transport, lifting, and MPC transfer operations in all normal, off-normal, and accident conditions.
Lateral Restraints (Transporter at CTF)	Used to restrain the transporter from motion during design basis seismic events.
Transfer Cask Impact Limiters ***(10 CFR Part 50 only)***	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask and preclude the accidental drop of a canister inside the Diablo Canyon Power Plant Fuel Handling Building/Auxiliary Building only.

The lateral restraints are used to restrain the HI-TRAC 125 Transfer Cask and transporter from motion at the Cask Transfer Facility (CTF) during design-basis seismic events. The lateral restraints are briefly described in SAR Section 4.2.1.2 "CTF Support Structure." Sufficient description of the lateral restraints is provided.

The transfer cask impact limiters are used only during the 10 CFR Part 50 operations inside the Diablo Canyon Power Plant (DCPP) Fuel Handling Building and Auxiliary Building (FHB/AB). They are not covered in this SER.

Based on the previous discussion, the staff concludes that these Category A important to safety items are correctly classified.

Category B Structures, Systems, and Components

The SSCs classified as Category B are summarized in SER Table 4-2. The failure of a Category B item, in conjunction with failure of an additional item, could result in an unsafe condition. The following Category B components have been properly classified.

As identified in SAR Section 4.5.1.1, "Multi-Purpose Canister and Fuel Basket," the upper and lower fuel spacer columns and end plates maintain fuel in the correct geometry. Sufficient descriptions of the upper and lower fuel spacer columns and end plates are provided in SAR Section 4.2.3 "Storage Cask Description."

As identified in SAR Section 4.5.1.3, "Overpack," the HI-STORM 100SA Overpack protects the MPC during storage. Sufficient description of the storage cask is provided in SAR Section 4.2.3, "Storage Cask Description."

Details associated with these two items are cask-specific, and additional descriptions are presented in the HI-STORM 100 System FSAR (Holtec International, 2002).

As identified in SAR Section 4.5.2, "Cask Storage Pads," the cask storage pads provide the necessary embedment for the anchorage of the overpack. The overpack anchorage hardware is designed to prevent sliding and tip-over during the design-basis seismic event. Sufficient descriptions of the cask storage pad and overpack anchorage system are provided in SAR Sections 3.3.2, "ISFSI Cask Storage Pads;" and 4.2.1.1, "Cask Storage Pads."

The CTF is designed to protect the canisters from adverse natural phenomena during cask loading, unloading and canister transfer operations. The CTF jacks are part of the single-failure-proof load path. Sufficient description of the CTF is provided in SAR Sections 4.2.1.2, "CTF Support Structure;" and 4.4.5, "Cask Transfer Facility."

As identified in SAR Table 4.3-1, the transporter connector pins are used to connect the transfer cask lift links or the overpack lifting brackets to the cask transporter lift links. The applicable design codes are American National Standards Institute (ANSI) N14.6 (American National Standards Institute/American Nuclear Society, 1993) in accordance with the guidance of NUREG–0612, Section 5.1.6 (U.S. Nuclear Regulatory Commission, 1980). Sufficient description of the transporter connection pins is provided. The transfer cask horizontal lift rig and transfer cask lift slings are designated as special lifting devices in the load path of the transfer cask during lifting and movement between the FHB/AB and the CTF. Sufficient

Table 4-2. Category B quality assurance classification of SSCs with a potentially major impact on safety (Based on SAR Table 4.5-1)

Structures, Systems, and Components	Logic
Upper and Lower Fuel Spacer Columns and End Plates (HI-STORM)	Ensures the correct geometry of the stored fuel assemblies.
HI-STORM 100 SA Overpack (HI-STORM)	Serves as the primary component for protecting the canister during storage from environmental conditions and provides radiation shielding and canister heat rejection.
Independent Spent Fuel Storage Installation (ISFSI) Storage Pads	Designed to ensure a stable and level support surface for the storage cask in all normal, off-normal, and accident conditions. It provides the necessary embedment for the anchorage of the overpack.
Overpack Anchorage Hardware	Designed to prevent sliding and tip-over during a design-basis seismic event.
Cask Transfer Facility (CTF) (except jacks)	Prevents damage to the spent nuclear fuel and spent nuclear fuel storage cask system components during lifting and multipurpose canister (MPC) transfer operations in all normal, off-normal, and accident conditions.
CTF Jacks	Designed to prevent uncontrolled lowering of the load during lifting and MPC transfer operations.
Transporter Connection Pins	Designed to lock the transfer cask in place during transport.
Transfer Cask Horizontal Lift Rig (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask for storage operations and preclude the accidental drop of a canister.
Transfer Cask Lift Slings (HI-STORM)	Designed as part of the single-failure-proof load path used to lift, handle, and move the cask for storage operations and preclude the accidental drop of a canister.
Helium Fill Gas ***(10 CFR Part 50 only)***	Provides an inert environment for storage of the spent nuclear fuel.

description of the associated lifting devices is provided in SAR Sections 4.3.2.2, "Transfer Cask Horizontal Lift Ring;" and 4.3.2.3, "Transfer Cask Lift Slings."

The helium fill gas is placed in the MPC during the 10 CFR Part 50 operations inside the DCPP FHB/AB and is not covered in this SER.

Failure of one or more of these Category B items, combined with the subsequent failure of the MPC, is necessary to lead to a condition adversely affecting public health and safety. Based on the previous discussion, the staff concludes that these Category B items are correctly classified.

Category C Structures, Systems, and Components

The SSCs summarized in Table 4-3 of this SER have been properly classified as Category C. The HI-STORM 100 System mating devices are conceptually shown in SAR Figure 4.2-11. Except as noted previously, failure of this component would not likely result in an unsafe condition. Based on the previous discussion, the staff concludes that this Category C item is correctly classified.

4.1.2.2 Classification of Structures, Systems, and Components—Items Not Important to Safety

Based on SAR Table 4.5-1, the classification of SSCs not important to safety do not involve a safety-related function and are not subject to NRC-imposed regulatory requirements.

A number of systems are security related including: security system, fencing, lighting, and communications systems. Each system is used to support the activities of the security personnel who monitor the controlled area of the Diablo Canyon ISFSI. If systems fail, the security personnel can still perform their required functions. Therefore, the security systems are correctly classified as not important to safety.

Because the HI-STORM 100SA Overpack is a passive system, normal electrical power can also be classified as not important to safety. No electrical power is required for the storage system to perform its design functions.

The automated welding system, MPC helium backfill system, MPC force helium dehydration system, and MPC vacuum drying system are used only during the 10 CFR Part 50 operations and are not covered in this SER.

Table 4-3. Category C quality assurance classification of SSCs (Based on SAR Table 4.5 1)

Structures, Systems, and Components	Logic
HI-STORM Cask Mating Devices (except bolts and shielding frame)	Used to lift, handle, and move the cask for storage operations and preclude the accidental drop of a canister.

As identified in SAR Section 4.3.2.4, "Cask Transport Frame," the cask transporter frame is used for rotating the transfer cask between the horizontal and vertical orientations. It is attached to, but does not support, the transfer cask during transport from the FHB/AB to the CTF. Therefore, the cask transport frame is correctly classified as not important to safety.

The cask transporter is designed such that any malfunction in the drive and control systems will cause it to stop in a safe condition. A malfunction of the drive system will initiate the fail-safe braking system. The control system is a dead-man system, in which the operator must actively drive the system. Therefore, it can be considered fail safe. The CTF drive and control systems are correctly classified as not important to safety.

4.1.2.3 Classification of Structures, Systems, and Components—Conclusion

The staff evaluated the classification of SSCs important to safety by reviewing SAR Chapter 4, "ISFSI Design"; documents cited in the SAR; and other relevant literature. Details of the quality assurance program evaluation are contained in SER Chapter 12. The staff determined that the classification of the SSCs important to safety and their associated categories are consistent with the regulatory requirements of 10 CFR §72.144(a) and associated technical information content of the application, as specified in 10 CFR §72.24(n).

4.1.3 Design Criteria for Structures, Systems, and Components Important to Safety

The principal design criteria identified for SSCs important to safety at the Diablo Canyon ISFSI are described in SAR Chapter 3, "Principal Design Criteria." This section contains a review of Section 3.2, "Design Criteria for Environmental Conditions and Natural Phenomena;" Section 3.3, "Design Criteria for Safety Protection Systems;" and Section 3.4, "Summary of Design Criteria." Diablo Canyon site-specific design criteria are derived from information in other sections of this SAR and from the DCPP FSAR update. The design criteria for the Holtec storage system are derived for the HI-STORM 100 System CoC (U.S. Nuclear Regulatory Commission, 2002a), the HI-STORM 100 System FSAR (Holtec International, 2002). Details of the design criteria evaluation are provided in SER Sections 4.1.3.1 to 4.1.3.7.

4.1.3.1 General

The staff reviewed the discussion of the general design criteria for SSCs with respect to the following regulatory requirements:

- 10 CFR §72.120(a) requires that, pursuant to the provisions of 10 CFR §72.24, an application to store spent nuclear fuel in an ISFSI must include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance, and performance requirements for SSCs important to safety as defined in 10 CFR §72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI.

- 10 CFR §72.122(h) specifies the criteria for confinement barriers and systems, including: §72.122(h)(1), which requires that the spent nuclear fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage; §72.122(h)(3), which requires that ventilation systems and off-gas systems be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions; §72.122(h)(4), which requires that storage confinement systems must have the capability for monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions; and §72.122(h)(5), which requires that the high-level waste be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of Part 20 limits. The package must be designed to confine the high level radioactive waste for the duration of the license.

- 10 CFR §72.144(c) requires that the licensee shall base the requirements and procedures of its quality assurance program(s) on the following considerations concerning the complexity and proposed use of the SSCs: (1) the impact of malfunction or failure of the item on safety; (2) the design and fabrication complexity or uniqueness of the item; (3) the need for special controls and surveillance over processes and equipment; (4) the degree to which functional compliance can be demonstrated by inspection or test; and (5) quality history and degree of standardization of the item.

A summary of the Diablo Canyon ISFSI general design criteria is provided in SER Table 4-4. The design of the proposed Diablo Canyon ISFSI is based on the use of the HI-STORM 100 System, which has been approved by the NRC for use according to the general license provisions of 10 CFR Part 72.

The design lifes of SSCs important to safety are based on their ability to withstand the applied loads. The applied loads are defined in terms of an annual probability of exceeding the design load. Analysis procedures are used to demonstrate the ability of the SSCs to withstand the applied loads with additional factors applied to the loads and material allowables by the referenced codes and standards. The storage capacity and number of casks to be stored at the Diablo Canyon ISFSI have been identified in SAR Section 3.1, "Purposes of Installation."

The staff reviewed the general design criteria identified, and summarized them in SER Table 4-5. The staff found that the criteria are consistent with the HI-STORM 100 System FSAR (Holtec International, 2002). Definitions of the normal, off-normal, and accident loads are given in SAR Sections 3.2, "Design Criteria for Environmental Conditions and Natural Phenomena;" and 3.3, "Design Criteria for Safety Protection Systems." The quality standards for design basis of SSCs important to safety are provided in SAR Chapters 3, "Principal Design Criteria" and 11, "Quality Assurance." These design criteria satisfy, in part, the requirements of 10 CFR §72.120(a), §72.122(h), and §72.236(b) in that design criteria are identified and SSCs important to safety will be designed to quality standards commensurate with the important to safety functions to be performed to satisfy the requirements of 10 CFR §72.144(c).

Table 4-4. Summary of Diablo Canyon ISFSI design criteria—general and spent nuclear fuel specification (Based on SAR Tables 3.4-2 through 3.4-5)

Design Parameters	Design Conditions	Reference
ISFSI Design Life	20 years	SAR Section 1.1
Storage Pad Design Life	40 years	SAR Table 3.4-3
Cask Transfer Facility Design Life	40 years	SAR Table 3.4-5
HI-STORM 100 System Design Life	40 years	SAR Section 3.3.1.3.1
Cask Transporter Design Life	20 years	SAR Table 3.4-4
Storage Capacity	4,400 spent nuclear fuel assemblies	SAR Section 3.1
Number of Casks	140 casks including two spares	SAR Section 3.1
Type of Fuel	Nonconsolidated Pressurized Water Reactor Westinghouse 17 x 17 LOPAR and VANTAGE 5	SAR Sections 3.1.1 and 10.2.1.1 and Tables 10.2-1 through 10.2-5
Fuel Characteristics	Physical, thermal, and radiological characteristics	SAR Sections 3.1.1 and 10.2.1.1 and Tables 3.1-1 and 3.1-2 and 10.2-1 through 10.2-5
Fuel Classification	Intact, damaged, debris	SAR Sections 3.1.1 and 10.2.1.1 and Tables 10.2-1 through 10.2-10 and Diablo Canyon ISFSI Tech Specs
Nonfuel Hardware	Borosilicate absorber rods Wet annular burnable absorber rods Thimble plug devices Rod cluster control assemblies	SAR Section 3.1.1.3 and Tables 3.1-1 and 10.2-10

Table 4-5. Design criteria for Diablo Canyon major ISFSI structures, systems and components (Based on SAR Table 3.4-1)

Design Criterion	Diablo Canyon ISFSI Design Value	Applicable Criteria and Codes
Seismic	Design Earthquake [5% Zero Period Acceleration (ZPA) = 0.201g horizontal and 0.134g vertical, periods up to 1.0 second] Double Design Earthquake (5% ZPA = 0.402g horizontal and 0.268g vertical, periods up to 1.0 second) Hosgri Earthquake (5% ZPA = 0.75g horizontal and 0.50g vertical, periods up to 0.8 second) LTSP (5% ZPA = 0.83g horizontal and 0.70g vertical, periods up to 2.0 seconds) ILP (5% ZPA = 0.83g horizontal and 0.70g vertical) (envelopes other design spectra and includes extended low frequency components) (SAR Section 3.2.3)	10 CFR §72.102
Wind	35.77 m/s [80 mph] with gust factor of 1.1 (SAR Section 3.2.1)	ASCE–7
Tornado	89.41 m/s [200 mph], maximum speed 70.19 m/s [157 mph], rotational speed 19.22 m/s [43 mph], translational speed 5.93 kPa [0.86 psi], pressure drop 2.48 kPa [0.36 psi]/sec rate of drop (SAR Section 3.2.1, Table 3.2-1)	Regulatory Guide 1.76
Tornado Missiles	DCPP—Generic 1,814 kg [1.79 ton] automobile, 14.89 m/s [33.3 mph] 34.5 kg [76.1 lb], 25.4-cm [3-in] diameter × 3.04-m [10-ft] schedule 40 pipe, 29.82 m/s [66.7 mph] 49.0 kg [108.03 lb] 10.16 cm [4 in] × 30.48 cm [12 in] × 3.04 m [10 ft] wood plank, 57.91 m/s [190 ft/sec] Diablo Canyon ISFSI—Site Specific 130 kg [286.60 lb], 15.24-cm [6-in] diameter schedule 40 pipe, 3.13 m/s [7 mph] 510 kg [0.50 ton] wooden utility pole, 15.65 m/s [35 mph] 340 kg [0.33 ton], 30.48-cm [12-in] diameter schedule 40 pipe, 2.24 m/s [5 mph] 3.9 kg [8.60 lb] 5.08 cm [2 in] × 5.08 cm [2 in] × 0.32 cm [1/8 in] × 1.52 m [5 ft] Steel Angle, 70.19 m/s [157 mph] 344.7 kg [0.34 ton] 500-kV insulator string, 70.19 m/s [157 mph] 6.8 kg [14.99 lb] 500-kV insulator segments, 70.19 m/s [157 mph] 4 kg [8.82 lb], 2.54-cm [1-in] diameter steel rod, 2.24 m/s [5 mph] (SAR Section 3.2.1, Table 3.2-2)	NUREG–0800, Section 3.5.1.4

Table 4-5. Design criteria for Diablo Canyon major ISFSI Structures, Systems and Components (Based on SAR Table 3.4-1) (continued)

Design Criterion	Diablo Canyon ISFSI Design Value	Applicable Criteria and Codes
Flood	Design-basis flooding event are not considered credible. (SAR Section 3.2.2)	NUREG–0800, Section 3.4.1
Snow and Ice	Design basis snow and ice loadings are not considered credible. (SAR Section 3.2.4)	ASCE–7
Fire	Fuel tank on transporter, load-handling equipment, or other vehicle Local stationary fuel tanks Local combustible materials Nearby grass/brush fire (SAR Sections 2.2.2.2, 3.3.1.6, and 8.2.5)	ANSI/AHS 57.9
Explosion	Fuel tank on transporter, load-handling equipment, or other vehicle 26.5-L [7-gal] propane bottle Standard acetylene bottle 0.95-m^3 [250-gal] propane tank, 7.57-m^3 [2,000-gal] No. 2 diesel fuel oil tank, and 11.36-m^3 [3,000-gal] gasoline tank Unit 2 transformers, approximately 49.21 m^3 [13,000 gal] of mineral oil Standard compressed gas bottle Hydrogen gas facility Acetylene bottle storage (SAR Sections 2.2.2.3, 3.3.1.6, and 8.2.6)	Reg. Guide 1.91
Severe Electrical Lightning	500-kVA Line Drop (SAR Section 3.2-6)	10 CFR §72.122(g)
Ambient Conditions	Annual Average = 12.78 °C [55 °F] Low Temperature = below freezing for a few hours Maximum Recorded = 36.11 °C [97 °F] Extreme Temperature Range = -4.4 °C [24 °F] to 40 °C [104 °F] Insolation = 791 Whr/m^2 [766 g-cal/cm^2] maximum for a 24-hour period (SAR Sections 2.3.2, 3.2.7, 8.2.6, and 8.2.10)	National Oceanic and Atmospheric Administration data

The ISFSI is located near the DCPP and must be designed and operated to ensure the cumulative effects of their combined operations will not constitute an unreasonable risk to the health and safety of the public as defined in 10 CFR §72.122(e). As identified in the applicant response to Request for Additional Information (RAI)[1] 2-8 through 2-11, onsite hazards and cumulative effects have been identified (SAR Section 2.2.2, "Onsite Potential Hazards") and accident analysis performed (SAR Section 8.2, "Accidents"). Therefore, there will be no undue risk to the public health and safety from combined ISFSI and DCPP operations.

4.1.3.2 Structural

The staff reviewed the discussion on structural design criteria of SSCs in the SAR with respect to the following regulatory requirements:

- 10 CFR §72.102(f) requires that the design earthquake (DE) for use in the design of structures be determined as follows: (1) For sites that have been evaluated in accordance with the criteria of Appendix A of 10 CFR Part 100, the DE must be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant. (2) Regardless of the results of investigations anywhere in the continental United States, the DE must have a value for the horizontal ground motion of no less than 0.10g with the appropriate response spectrum.

- 10 CFR §72.120(a) requires that, pursuant to the provisions of 10 CFR §72.24, an application to store spent nuclear fuel in an ISFSI must include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance, and performance requirements for structures, systems, and components important to safety as defined in 10 CFR §72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI.

- 10 CFR §72.122(b)(1) requires structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents.

- 10 CFR §72.122(b)(2)(i) requires structures, systems, and components important to safety to be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform their intended design functions. The design bases for these structures, systems, and components must reflect: (A) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have

[1]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Spent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

accumulated, and (B) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (2)(ii) The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects, as a result of building structural failure, on the spent nuclear fuel or high-level waste or on SSC important to safety.

- 10 CFR §72.122(b)(4) specifies that if the ISFSI is located over an aquifer that is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

- 10 CFR §72.122(c) requires that structures, systems, and components important to safety be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on SSC important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire-suppression system.

SAR Sections 3.2, "Criteria for Environmental Conditions and Natural Phenomena;" and 3.3, "Design Criteria for Safety Protection Systems," address the structural and mechanical design criteria. The design criteria for the site include dead loads, live loads, seismicity, wind, tornado, tornado missiles, flood, snow and ice, explosion overpressure, fire, and site ambient temperature, humidity, solar radiation, and lightning. Information on the derivation of site-specific design criteria for the meteorology, hydrology, and seismology is contained in SAR Chapter 2, "Site Characteristics."

The structural design loads for SSCs important to safety are provided in the following section of the SER. As identified, the SSCs important to safety are designed to withstand the effects of environmental conditions and natural phenomena for normal, off-normal, and accident conditions. Important to safety design criteria for the HI-STORM 100 System are described in FSAR (Holtec International, 2002). SAR Table 4-6 identified the HI-STORM 100 System design criteria in relations to the Diablo Canyon ISFSI design criteria. Review in this SER is limited to identification of enveloping design criteria. Adequacy of the HI-STORM 100 System design criteria is discussed in the NRC HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b).

Site-specific design criteria not enveloped by the HI-STORM 100 System FSAR (Holtec International, 2002) criteria are identified in SAR Sections 3.2, "Criteria for Environmental Conditions and Natural Phenomena;" and 3.3, "Design Criteria for Safety Protection Systems." These criteria include the specific site data, storage system affected, and the corresponding section in the SAR where it is addressed. Structural design criteria and radiological protection and confinement criteria are identified. The structural criteria are discussed in Chapter 4 of this SER. Radiological protection and confinement criteria are considered in Chapters 7, "Shielding Evaluation;" 8, "Criticality Evaluation;" 9, "Confinement Evaluation;" and 11, "Radiation Protection Evaluation" of this SER.

Seismicity

The staff reviewed the data presented in the SAR associated with seismic design criteria at the Diablo Canyon ISFSI in accordance with 10 CFR §72.102(f). SAR Section 3.2.3, "Seismic Design," gives the seismic design criteria, based on probabilistic site-specific seismology studies summarized in SAR Section 2.6, "Geology and Seismology." There are four site-specific design response spectra: design earthquake (DE), double design earthquake (DDE), Hosgri earthquake (HE), and the Long-Term Seismic Program (LTSP) earthquake. The DE and DDE correspond to the operational and SSE for the DCPP. The HE represents the maximum credible event on the Hosgri fault. This earthquake ground motion is the design-basis for the Facility, based on its close proximity to the site. Based on work during the LTSP, another seismic design basis was defined for the DCPP for verification of the adequacy of seismic margins of certain plant SSCs. The response spectra for these design-basis events are given in SAR Figures 2.6-43 and 2.6-44. These response spectra are considered acceptable for the ISFSI because they have been accepted by the staff for the DCPP. Because of ISFSI pad sliding, pad cutslope stability, transport route stability and transporter stability, that may be affected by a longer-period ground motion, an additional design-basis event was developed that extended out to periods of 10.0 seconds. This design-basis event includes fault normal and parallel directing response and fault fling. The response spectra for these design-basis events are given in SAR Figures 2.6-45 and 2.6-46. Spectra for the vertical and horizontal directions are identified for all design-basis ground motions. The site-specific seismic design criteria of the Diablo Canyon ISFSI are bounded by the HI-STORM 100 System seismic design criteria for an anchored cask. The staff assessment of the adequacy of the site-specific seismic design criteria is contained in Section 2.1.6 of this SER. The applicant analysis of the HI-STORM 100 System under the site-specific design-basis seismic event is evaluated in Chapters 5 and 15 of this SER. The staff reviewed the seismic design criteria for the Diablo Canyon ISFSI and found that they are properly identified as required by 10 CFR §72.120(a) and §72.122(b).

Wind

Figure 6-1 in ASCE 7-98 (American Society of Civil Engineers, 2000) identifies a design-basis 3-second gust wind speed of 38 m/s [85 mph] for the region. Information provided in SAR Section 3.2.1, "Tornado and Wind Loadings," identifies a maximum recorded wind gust speed at the DCPP site of 37.55 m/s [84 mph]. The design-basis wind speed of 35.76 m/s [80 mph] with a wind gust factor of 1.1 envelopes these values {35.76 m/s [80 mph] x 1.1 gust factor = 39.34 m/s [88 mph] > 38 m/s [85 mph]}. The staff reviewed the design-basis wind {35.76 m/s [80 mph]} for the Diablo Canyon ISFSI and found that it is consistent with that identified in ASCE 7-98 (American Society of Civil Engineers, 2000) for this location. The requirements of 10 CFR §72.120(a) and §72.122(b) are satisfied in that the effects of site conditions and environmental conditions are considered in the Diablo Canyon ISFSI design.

Tornado

The design-basis tornado wind loads are the DCPP site licensing-basis information provided in Section 3.3.2.1.1 of the DCPP FSAR update. The parameters for the tornado identified in the DCPP FSAR have been reviewed with respect to the 10 CFR Part 50 license and are consistent with those given in Regulatory Guide 1.76 (U.S. Atomic Energy Commission, 1974). The requirements of 10 CFR §72.120(a) and §72.122(b) are satisfied in that the effects of site conditions and environmental conditions are considered in the Diablo Canyon ISFSI design.

Tornado Missiles

The tornado missiles, identified in SAR Table 3.2-2, are a compilation of those items specified as Spectrum II missiles in NUREG–0800, Section 3.5.1.4 (U.S. Nuclear Regulatory Commission, 1981), the DCPP FSAR update, and three 500-kV tower missiles specific to the Diablo Canyon ISFSI site. These items are considered to be representative of potential missiles present at the site. As identified in the HI-STORM 100 System FSAR, the cask-specific tornado missiles correspond to Spectrum I missiles in NUREG–0800. Use of either Spectrum I or II missile is considered acceptable by the NRC. The staff reviewed the design-basis tornado conditions for the Diablo Canyon ISFSI and found that the conditions are consistent with the design criteria, as specified by NUREG–0800, Section 3.5.1.4, to withstand tornadoes, in accordance with the requirements of 10 CFR §72.120(a) and §72.122(b).

Flood

Based on the location of the Diablo Canyon ISFSI pad and the site surface hydrology, it is concluded in SAR Section 2.4, "Surface Hyrdology," that there is no potential for flooding in the vicinity of the ISFSI. The same conclusion is applicable to the HI-STORM 100SA Overpack that are stored on the pad and the CTF located in close vicinity to the storage pads. In addition, the CTF is protected by a sump that can remove any significant accumulation of water in the vault. Therefore, the forces resulting from flood waters and flood protection measures do not need to be considered in design of SSCs important to safety. The staff, therefore, concludes that the Diablo Canyon ISFSI design is consistent with the design criteria of NUREG–0800 and American Society of Civil Engineers (ASCE) 7-98 (American Society of Civil Engineers, 2000) to withstand floods as required by 10 CFR §72.120(a) and §72.122(b).

Snow and Ice

Figure 7-1 of ASCE 7-98 (American Society of Civil Engineers, 2000) indicates no snow up to an elevation of 457.20 m [1,500 ft]. SAR Section 3.2.4, "Snow and Ice Loadings," states that there is essentially no design ground snow load at Diablo Canyon ISFSI. As identified in the HI-STORM 100 System FSAR (Holtec International, 2002), the overpack is designed for a 4.79-kPa [100-lb/ft^2] snow and ice load. This load bounds the Diablo Canyon ISFSI site design criteria. The staff reviewed the snow and ice loading criteria and determined that they are appropriate and in accordance with the requirements of 10 CFR §72.120(a) and §72.122(b).

Fire

The applicant has identified a total of five fire events in SAR Section 8.2.5, "Fire." A range of onsite fire scenarios has been evaluated. The applicant has demonstrated that the 189.21-L [50-gal] transporter fuel tank fire is the bounding condition. Operational restrictions are in place to ensure that these levels are not exceeded. The staff reviewed the fire considerations in the SAR and found that they are consistent with equipment used at the Facility and operational restraints as required by 10 CFR §72.122(c). Appropriate design criteria are specified to ensure that SSCs important to safety will be designed and located so that they can perform their safety functions effectively during credible fire exposure conditions.

Explosion

The applicant has identified a total of eight explosion event categories (Events 1–8), SAR Section 8.2.6, "Explosion," that identify the explosive hazard and its relationship to the Facility. Because of the combinations of the amount of explosive hazard and the standoff distance, it was not possible to establish an explosive overpressure design criterion for the Diablo Canyon ISFSI based on the assumption that all credible events will result in a pressure at the SSCs important to safety of less than 6.89 kPa [1 psi]. The 6.89-kPa [1-psi] limit is based on Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) that states:

> "A method for establishing the distances referred to above can be based on a level of peak positive incident overpressure (designated as P_{so} in Ref.1) below which no significant damage would be expected. It is the judgement of the Staff that, for the SSC, this level can be chosen as 1 psi (approximately 7 kPa)."

Rather than use the 6.89-kPa [1-psi] design limit, the applicant took several approaches in an attempt to define the design basis explosion overpressure design criterion. Above 6.89 kPa [1 psi], the structural integrity of SSCs must be addressed in addition to the potential for explosion-induced projectiles (U.S. Nuclear Regulatory Commission, 1978). During a blast event, the normal reflected pressure is twice the peak incident pressure P_{so}. Therefore, a normal reflected pressure should be considered in the analysis of the structure when considering the blast pressure loading.

For four of the cases (Events 1 to 4), the equivalent weight of trinitrotoluene (TNT) and scaled ground distance were calculated, and the resultant incident overpressure calculated, as shown in SAR Table 8.2-11.

The applicant demonstrated that the potential risk from an explosion of the mineral oil in the DCPP Unit 2 main bank transformers (Event 5) is insignificant, as shown in SAR Section 8.2.6, "Explosion."

The explosive decompression of the compressed gas cylinders (Event 6) is assumed to generate a missile. The mass and velocity of these missiles are identified in SAR Section 8.2.6.2.2, "Missiles Due to Explosive Decompression of a Decompressed Gas Cylinder."

The applicant demonstrated that the potential risk from an explosion of the bulk hydrogen facility (Event 7) is insignificant, as shown in SAR Section 8.2.6.2.3, "Potential Explosion Event of the Bulk Hydrogen Facility."

The applicant took an alternate approach for the detonation of acetylene bottles (Event 8). As identified in SAR Section 8.2.6.2.1, "Explosive Overpressure Due to Detonation Events," the transfer cask has been evaluated for the effects of a handling accident (a 45g deceleration during a drop event). The applicant calculates a uniform pressure of 2.65 mPa [384 psig] over the projected area of the transfer cask, which is equivalent to the force produced during the deceleration. The applicant states that approximately 16,000 acetylene bottles would be required to detonate at this location to develop an overpressure at the passing transfer cask greater than 2.65 mPa [384 psig]. This number far exceeds the available bottle storage space.

Note that the applicant indicates that Events 4 to 8 could occur on the transport route and, therefore, are applicable only to the transfer cask, which does not have an explicit explosive design criteria (Table 2.0.3 of the HI-STORM 100 System FSAR). These loads are also applicable to the cask transporter and auxiliary lifting components as shown in Table 3.4-4 of the SAR to demonstrate that their safety function is not impaired by these events.

The staff reviewed the explosion considerations in the SAR and found that the applicant has identified eight events. At various locations within the SAR, the applicant has characterized the credible events with respect to the resulting peak pressure. The procedures used are consistent with standard design criteria, NFPA 30 (National Fire Protection Association, 1996) and Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) as required by 10 CFR §72.122(c).

Lightning

During thunderstorms, a lightning strike is possible. The overhead transmission line prevents a direct lightning strike on any overpack or the CTF. The HI-STORM 100SA Overpack is also designed for lightning protection. Any lightning strike on the cask will discharge through its steel shell to the ground, having no adverse impact on the cask or fuel. The staff reviewed the lightning design criterion as identified in the SAR with reference to the ISFSI specifications (Pacific Gas and Electric Company, 2000) and determined that it is acceptable for the design of SSCs important to safety as required by 10 CFR §72.122(b).

Ambient Conditions

The loads from environmental conditions and natural phenomena specified in the SAR are consistent with standard engineering practice, as identified in ASCE 7-98 (American Society of Civil Engineers, 2000), which identifies the minimum design loads for buildings and other structures.

HI-STORM 100 System Design Criteria

The applicant has identified design criteria for each of the major systems at the ISFSI as provided in SAR Table 3.4-2 through 3.4-5 and Table 4-6 of this SER. The design criteria for the pressure vessel portions of the HI-STORM 100 System conform to standard engineering practice, as identified in the ASME International Boiler and Pressure Vessel Code (ASME International, 1998). The ASME Boiler and Pressure Vessel Code establishes rules of safety governing the design, fabrication, and inspection during construction of boilers and pressure vessels. This code contains mandatory requirements, specific prohibitions, and nonmandatory guidance for selection of materials, design, fabrication, examination, inspection, testing, certification, and pressure relief.

For concrete components of the HI-STORM 100 System, as identified in SAR Table 3.4-2, the design criteria are based on the American Concrete Institute ACI 318-95 (American Concrete Institute, 1995) and ACI 349-85 (American Concrete Institute, 1985). ACI 349-85 specifies the proper design and construction of concrete structures that form part of a nuclear power plant

Table 4-6. Design criteria for Diablo Canyon HI-STORM 100 System (Based on SAR Table 3.4-2)

Design Criterion	HI-STORM 100 System Design Value	Applicable Criteria and Codes
Structural Design		
HI-STORM 100 System Design Codes	Canister: Internals: Overpack: Transfer Cask: } Holtec FSAR, as amended by LAR 1014-1 Tables 2.2.2, 2.2.7, 2.2.14, and 2.2.15.	ASME III-95, with 1996 and 1997 Addenda, Subsection NB ASME III, NG ASME III, NF, ACI-318 (95), ACI–349 (85) ASME III, NF, ANSI N14.6 (93) ASCE 7-88
Environmental Conditions and Natural Phenomena	See SAR Table 3.4-1 (SAR Sections 3.2 and 3.3)	
Weights	Maximum loaded transfer cask handling weight = 113,400 kg [250,000 lb] Maximum loaded overpack weight = 163,300 kg [360,000 lb] Transporter weight = 77,100 kg [170,000 lb]	
MPC Internal Pressure	Normal/off-normal = 0.69 MPa [100 psig] Accident = 1.38 MPa [200 psig] (Holtec FSAR, Table 2.0.1 as amended by LAR 1014-1)	
Cask Loads and Load Combinations	Varies (Holtec FSAR Sections 2.2.1 through 2.2.3 and Tables 2.2.13 and 2.2.14 as amended by LAR 1014-1)	
Thermal Design		
Maximum Cask Heat Duty	Varies Maximum pressurized water reactor basket heat duty = 28.74 kW (Holtec FSAR, Section 4.4.2, as amended by LAR 1014-1, Section 4.4.2 and Table 4.4.28)	

Table 4-6. Design criteria for Diablo Canyon HI-STORM 100 System (Based on SAR Table 3.4-2) (continued)

Design Criterion	HI-STORM 100 System Design Value	Applicable Criteria and Codes
Peak Fuel Cladding Temperature Limits	Long-term (normal) limits vary based on fuel-cooling time Short-term (accident) = 570 °C [1,058 °F] (Holtec FSAR, Rev. 1, Tables 4.3.7 and 4.A.3 and HI-STORM 100 System FSAR Table 4.3.1)	
Other SSCs Temperature Limits	Varies by material (Holtec FSAR, Tables 2.0.1 through 2.0.3, as amended by Holtec LAR 1014-1)	
MPC Backfill Gas	99.995% pure helium (SAR Section 10.2.2.4 and Holtec CoC, Appendix A, Table 3-1, as amended by LAR 1014-1)	
Maximum Air Inlet to Outlet Temperature Rise	52.2 °C [126 °F] (Holtec CoC, Appendix A, LCO 3.1.2, as amended by LAR 1014-1)	
Radiation Protection and Shielding Design		
Storage Cask Dose Rate Objectives	0.6 mSv/hr [60 mrem/hr] on sides, top, and adjacent to air ducts (SAR Section 3.3.1.5.2 and Holtec FSAR Section 2.3.5.2 as amended by LAR 1014-1)	
Occupational Exposure Dose Limits	Total effective dose equivalent 50 mSv/yr [5 rem/yr] lens dose equivalent 15 mSv/yr [15 rem/yr] shallow-dose equivalent and extremities 500 mSv/yr [50 rem/yr]	10 CFR §20.1201
Restricted Area Boundary Dose Rate Limit	0.02 mSv/hr [2 mrem/hr]	10 CFR §20.1301
Normal Operation Dose Limits to Public	0.25 mSv/yr [25 mrem/yr] whole body 0.75 mSv/yr [75 mrem/yr] thyroid 0.25 mSv/yr [25 mrem/yr] other critical organ	10 CFR §72.104

Table 4-6. Design criteria for Diablo Canyon HI-STORM 100 System (Based on SAR Table 3.4-2) (continued)

Design Criterion	HI-STORM 100 System Design Value	Applicable Criteria and Codes
Accident Dose Limits to Public	50 mSv [5 rem] TEDE 150 mSv [15 rem] lens dose equivalent 300 mSv [50 rem] shallow dose equivalent to skin or extremity	10 CFR §72.106
Overpack Unreinforced Concrete	Various (Holtec FSAR, Appendix 1.D as amended by LAR 1014-1)	
Criticality Design		
Maximum Initial Fuel Enrichment	≤5% (SAR Sections 3.3.1.4.1 and 3.1.1.1, Tables 10.2-1 through 10.2-5, Diablo Canyon ISFSI Technical Specification and Holtec CoC, Tables 2.1-1 and 2.1-2 as amended by LAR 1014-1	
Control Method (Design Features)	See SAR Table 3.4-2 (Diablo Canyon ISFSI Technical Specification)	
Control Method (Operational)	See SAR Table 3.4-2 (Diablo Canyon ISFSI Technical Specification)	
Maximum k_{eff}	<0.95 (SAR Section 3.3.1.3 and Holtec FSAR Table 2.0.1, as amended by LAR 1014-1)	
Confinement Design		
Confinement Method	MPC with redundant welds (Holtec FSAR, Section 2.3.2.1, and Chapter 7, as amended by LAR 1014-1)	
Confinement Barrier Design	MPC (Holtec FSAR, Sections 2.2.6 and 2.2.15, as amended by LAR 1014-1 and Diablo Canyon ISFSI Technical Specification)	ASME III, Section NB
Maximum Confinement Boundary Leak Rate	5.0×10^{-6} atm-cm^3/sec (SAR Section 10.2.2.5)	

and that have nuclear safety-related functions, but does not cover concrete reactor vessels and concrete containment structures.

The structures covered by the ACI code include concrete structures inside and outside the containment system.

Thermal, shielding and confinement, and criticality design criteria are discussed in Sections 4.1.3.3 through 4.1.3.5 of this SER.

ISFSI Storage Pad Design Criteria

The applicant identified the design criteria for the storage pads, as indicated in SAR Table 3.4-3 and Table 4-7 of this SER. For concrete components of the storage pads, the design criteria are based on the ACI 349-97 (American Concrete Institute, 1998) and Draft Appendix B. ACI 349-97 specifies the proper design and construction of concrete structures that form part of a nuclear power plant and that have nuclear safety-related functions, but does not cover concrete reactor vessels and concrete containment structures. The structures covered by the ACI code include concrete structures inside and outside the containment system. The Draft Appendix B covers anchorage systems in concrete. Note that the anchorage system used at the Diablo Canyon ISFSI does not fall within the guidelines of this appendix, because the size and length of the anchors used at the ISFSI exceed those with supporting data in the Draft Appendix B. The applicant used the Draft Appendix B to identify that the anchorage design should have a failure mode that is ductile (i.e., failure of the anchorage steel).

Cask Transporter Design Criteria

The applicant identified the design criterion for the cask transporter, SAR Table 3.4-4 and Table 4-8 of this SER. The item is to be purchased as a commercial-grade product and qualified by testing prior to use. In addition, NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980) is identified for the design criteria of the cask transporter, for compliance with a single-failure-proof lift system. NUREG–0612 identifies controls for handling heavy loads at nuclear power plants.

Cask Transfer Facility Design Criteria

The applicant identified the design criteria for the CTF, SAR Table 3.4-5 and Table 4-9 of this SER. The design criteria for the load support portions of the CTF conform to standard engineering practice, as identified in the ASME Boiler and Pressure Vessel Code (ASME International, 1998). The ASME Boiler and Pressure Vessel Code establishes rules of safety governing the design, fabrication, and inspection during construction of boilers and pressure vessels. This code contains mandatory requirements, specific prohibitions, and nonmandatory guidance for selection of materials, design, fabrication, examination, inspection, testing, certification, and pressure relief. In addition, NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980) is identified for the design criteria of the CTF load support SSCs, for compliance with a single-failure-proof lift system. NUREG–0612 identifies controls for handling heavy loads at nuclear power plants.

Table 4-7. Design criteria for Diablo Canyon storage pad (Based on SAR Table 3.4-3)

Design Criterion	Diablo Canyon ISFSI Design Values	Applicable Criteria and Codes
Design Codes	ACI-349 (97) and Draft Appendix B (10/01/01) (SAR Sections 3.3.2.3 and 4.2.1.1.2)	NUREG–1536
Design Life	40 years (SAR Section 3.3.2.3)	
Maximum Single Loaded Cask Weight	163,300 kg [360,000 lb] (Holtec FSAR Table 2.0.1)	
Transporter with Loaded HI-STORM 100 System	240,000 kg [530,000 lb] (Holtec FSAR Table 2.0.1 and assumed value)	
Maximum Number of Casks on a Single Pad	20 (SAR Sections 1.3 and 4.1)	
Maximum Number of Pads at the ISFSI	7 (SAR Sections 1.3 and 4.1)	
Operating Temperature Range	-17 to 38 °C [0 to 100 °F] (DCPP FSAR update, Section 2)	
Concrete Pad Strength	34.4 MPa [5,000 psi] at 90 days	D-1090 (8/00) ACI-349 (97) and Draft Appendix B (10/01/01)
Pad Loads and Load Combinations	Various	NUREG-1536, Table 3-1
Cask Anchor Stud Loads and Load Combinations	Various (SAR Section 3.3.2.3.2)	ASME, Section III, Subsection NF and Appendix F
Environmental Conditions and Natural Phenomena	see SAR Table 3.4-1 (SAR Section 3.2 and 3.3)	

Table 4-8. Design criteria for Diablo Canyon cask transporter (Based on SAR Table 3.4-4)

Design Criterion	Diablo Canyon ISFSI Design Value	Applicable Criteria and Codes
Transporter Design Codes	Purchase commercial grade and qualify by testing prior to use. (SAR Sections 3.3. and 4.3.2.1, and Diablo Canyon ISFSI Technical Specifications)	NUREG–0612
Design Life	20 years (SAR Section 3.3.3.2.1)	
Maximum Payload	163,300 kg [360,000 lb] (Holtec FSAR Table 2.0.1)	
Transporter Weight	77,100 kg [170,000 lb] (assumed value)	
Maximum Loaded Travel Speed	0.64 kmh [0.4 mph] (assumed value)	
Minimum Uphill Grade Capability	5% (carrying a loaded overpack) 10% (carrying a loaded transfer cask) (assumed value)	
Maximum On-Board Fuel Quantity	189 L [50 gal] (SAR Section 2.2.2.3 and Diablo Canyon ISFSI Technical Specifications)	
Maximum Hydraulic Fluid Volume	Unlimited (must the nonflammable) (SAR Section 3.3.3.2.2 and Diablo Canyon ISFSI Technical Specifications)	
Operating Temperature Range	-17 to 38 °C [0 to 100 °F] (DCPP FSAR update, Section 2)	
Redundancy and Safety Factors for Load Path Structures and Special Lifting Devices	Holtec FSAR Section 2.3.3.1	In accordance with applicable guidelines of NUREG–0612
Hoist Load Factor	15%	CMAA 70 (94)
Position Control Maintained with Loss of Motive Power	Stops in position	In accordance with applicable guidelines of NUREG-0612
Environmental Conditions and Natural Phenomena	see SAR Table 3.4-1 (SAR Section 3.2 and 3.3)	

Table 4-9. Design criteria for Diablo Canyon cask transfer facility (Based on SAR Table 3.4-5)

Design Criterion	Diablo Canyon ISFSI Design Value	Applicable Criteria and Codes
Design Codes	ASME III-95, with 1996 and 1997 Addenda, Subsection NF NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980) NUREG–1536 ACI–349 (97) and Draft Appendix B (10/01/00) (SAR Sections 3.3.4 and 4.4.5.2)	
Design Life	40 years (HOLTEC FSAR Section 2.3)	
Design Payload for Lift System	163,300 kg [360,000 lb] (HOLTEC FSAR Table 2.0.1)	
Loads and Load Combinations	Various (SAR Section 3.3.4.2.7)	ASME, Section III, Subsection NF and Appendix F
Hoist Load Factor	15%	CMAA 70 (94)
Operating Temperature Range	-17 to 38 °C [0 to 100 °F] (DCPP FSAR update, Section 2)	
Redundancy and Safety Factors for Load Path Members Special Lifting Devices	HOLTEC FSAR Section 2.3.3.1	In accordance with the applicable guidelines of NUREG–0612
Load Travel on Loss of Power or Jack Malfunction	Stops in position	In accordance with the applicable guidelines of NUREG–0612
Environmental Conditions and Natural Phenomena	SAR Table 3.4-1	

For concrete components of the CTF, the design criteria are based on ACI 349-97 (American Concrete Institute, 1998). ACI 349-97 specifies the proper design and construction of concrete structures that form part of a nuclear power plant and that have nuclear safety-related functions, but does not cover concrete reactor vessels and concrete containment structures. The structures covered by the ACI code include concrete structures inside and outside the containment system.

Load Combinations

The load combinations identified in SAR Section 3.2.5, "Combined Load Criteria," are used in the analysis of SSCs important to safety. The HI-STORM 100 System is designed for normal, off-normal, and accident conditions under the appropriate load combinations as identified in HI-STORM 100 System FSAR Sections 2.2.1 through 2.2.7 (Holtec International, 2002). Load combinations for the concrete storage pad and HI-STORM 100SA Overpack anchor studs are given in HI-STORM 100 System FSAR Sections 3.3.2.3.1 and 3.3.2.3.2. The cask transporter meets the applicable load criteria for the CTF as identified in HI-STORM 100 System FSAR Section 2.3.3.1. Load combinations for the CTF steel structure and equipment are discussed in HI-STORM 100 System FSAR Section 2.3.3.1. Load combinations for the CTF concrete are discussed in HI-STORM 100 System FSAR Section 3.3.4.2.7.1.

These load combinations are based on the requirements of ANSI/ANS 57.9 (American National Standards Institute/American Nuclear Society, 1992) and ACI–349–97 (American Concrete Institute, 1998).

The staff reviewed the Diablo Canyon ISFSI documentation and determined that the load combinations design criteria are appropriately considered for the design of SSCs important to safety as required by 10 CFR §72.122(b). Appropriate combinations of the effects of normal and accident conditions and the effects on natural phenomena are considered.

Structural Design Criteria Conclusion

The structural design criteria discussed previously represent the structural loads that may be present at the site. The Diablo Canyon ISFSI SSCs important to safety must be designed to withstand these structural loads, as applicable. The ability of the SSCs to perform their intended safety functions in accordance with the applicable structural design loads is evaluated in Chapters 5 and 15 of this SER.

The Diablo Canyon ISFSI site-specific structural design criteria are bounded by the applicable structural design criteria for the HI-STORM 100 System, except for the seismic design criteria. Thus, except for the seismic analysis, the structural analysis presented in the HI-STORM 100 System FSAR and the NRC structural evaluation as documented in the HI-STORM 100 System SER are valid for the Diablo Canyon ISFSI. Because the seismic design loads for the Diablo Canyon ISFSI are not enveloped by the seismic design loads for the HI-STORM 100 System, the applicant performed an analysis to demonstrate that the HI-STORM 100SA Overpack would perform acceptably under the site-specific design-basis seismic event. This analysis is also evaluated in Chapters 5 and 15 of this SER.

The staff reviewed the Diablo Canyon ISFSI documentation and determined that the principal design criteria, given in SAR Sections 3.2.2 through 3.4 considered for the design of SSCs are developed from appropriate site characteristics and are used in the determination of appropriate structural loads and load combination analyses. The values for these parameters form the basis for the structural design, mechanical design, and criticality assessment of the Diablo Canyon ISFSI.

4.1.3.3 Thermal

The staff reviewed the discussion on thermal design criteria of SSCs with respect to the following regulatory requirement:

- 10 CFR §72.120(a) requires that pursuant to the provisions of 10 CFR §72.24, an application to store spent nuclear fuel in an ISFSI must include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance, and performance requirements for SSC important to safety as defined in 10 CFR §72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI.

Thermal design criteria are based on environmental conditions and heat generated by the materials stored.

Ambient condition design criteria are based on site-specific meteorological conditions. The extreme minimum, annual average, and extreme maximum design temperatures identified for the site are -4, 13, and 40 °C [24, 55, and 104 °F]. The design temperatures are based on data from the region, which are consistent with the values measured to date according to the onsite meteorological measurement program. The staff reviewed the ambient condition loading design criteria and determined that they are acceptable because they are based on site-specific information, and the values are consistent with data from the National Oceanic and Atmospheric Administration for the region. Consequently, the ambient condition loading design criteria satisfy the requirements of 10 CFR §72.122(b), and the criteria are detailed in Chapter 6 of this SER.

The site-specific maximum total insolation for a 24-hour period was 742 Whr/m^2 (766 g cal/cm^2). As identified in SAR Section 2.3.2, "Local Meteorology," this value is based on a 13-year database for California Polytechnic State University in San Luis Obispo, California. The HI-STORM 100SA Overpack has been evaluated for the solar insolation values specified in 10 CFR §71.71(c)(1), which are 774 Whr/m^2 (800 g cal/cm^2) for flat surfaces and 387 Whr/m^2 (400 g cal/cm^2) for curved surfaces. The Standard Review Plan for dry-cask storage systems, NUREG–1536 (U.S. Nuclear Regulatory Commission, 1997) states, "The NRC staff accepts insolation presented in 10 CFR Part 71 for 10 CFR Part 72 applications. Because of the large thermal inertia of a storage cask, the values listed in 10 CFR §71.71 may be treated as the average insolation, calculated by averaging over a 24-hour day the reported 10 CFR Part 71 values for insolation over a 12-hour solar day, in a steady-state calculation." The staff concluded that both site-specific measurement and regional data are bounded by the design of the HI-STORM 100SA Overpack.

The storage systems are passive and incorporate passive heat removal. The thermal design criteria for the cask system were evaluated during licensing of the HI-STORM 100 System and documented in the NRC HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b).

Design temperatures for various HI-STORM 100 System materials are identified in the FSAR and are in compliance with acceptable codes. ACI 349-97 (American Concrete Institute, 1998) specifies the maximum temperature for normal operation and accident conditions. Allowable temperature for the fuel cladding is based on NUREG–1536. The ASME Boiler and Pressure Vessel Code, Section II, Part D, Table 1A specifies a design temperature for steel casks under all load conditions (ASME International, 1999). The performance requirements for all materials, as identified by the acceptable temperature, required for compliance with 10 CFR §72.120(a) are in conformance with accepted standards. Cask-specific material properties given in the SAR are derived from the HI-STORM 100 System FSAR Table 2.2.3.

4.1.3.4 Shielding and Confinement

The staff reviewed the discussion on shielding and confinement design criteria of SSCs with respect to the following regulatory requirements:

- 10 CFR §72.104(a) requires that, during normal operations and anticipated occurrences, the annual dose equivalent to any real individual 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.

- 10 CFR §72.106(a) requires that for each ISFSI, a controlled area be established.

- 10 CFR §72.122(h)(1) requires that the spent nuclear fuel cladding be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel while in storage will not pose operational safety problems with respect to its removal from storage.

- 10 CFR §72.126(a) requires that radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure must be designed, fabricated, located, shielded, controlled, and tested so as to control external and internal radiation exposures to personnel. The design must include means to (1) Prevent accumulation of radioactive material in those systems requiring access; (2) Decontaminate those systems to which access is required; (3) Control access to areas of potential contamination or high radiation within the ISFSI; (4) Measure and control contamination of areas requiring access; (5) Minimize the time required to perform work in the vicinity of radioactive components; for example, by providing sufficient space for ease of operation and designing equipment for ease of repair and replacement; and (6) Shield personnel from radiation exposure.

- 10 CFR §72.126(b) requires that radiological alarm systems be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given setpoint and of

concentrations of radioactive material in effluents above control limits. Radiation alarm systems must be designed with provisions for calibrating and testing their operability.

- 10 CFR §72.126(c) requires that effluent and direct radiation monitoring meet the following criteria: (1) as appropriate for the handling and storage system, effluent systems must be provided. Means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions must be provided for these systems. (2) Areas containing radioactive materials must be provided with systems for measuring the direct radiation levels in and around these areas.

- 10 CFR §72.126(d) requires that the ISFSI be designed to provide means to limit to as low as reasonably achievable the release of radioactive materials in effluents during normal operations and to control the release of radioactive materials under accident conditions. Analyses must show that releases to the general environment in normal operations and anticipated occurrences will be within the exposure limit given in 10 CFR §72.104. Analyses of design-basis accidents must be made to show that releases to the general environment will be within the exposure limits given in 10 CFR §72.106. Systems designed to monitor the release of radioactive materials must have means for calibrating and testing their operability.

- 10 CFR §72.128(a) requires that spent fuel storage, and other systems that might contain or handle radioactive materials associated with spent fuel, must be designed to ensure adequate safety under normal and accident conditions.

- 10 CFR §72.128(b) requires that radioactive waste treatment facilities be provided. Provisions must be made for the packing of site-generated, low-level wastes in a form suitable for storage onsite awaiting transfer to disposal sites.

Criteria used in the design of cask radiological protection features and confinement design of the cask systems are provided in the SAR and the HI-STORM 100 System FSAR and are summarized in this SER, Table 4-6. The basic concept for the Diablo Canyon ISFSI shielding and confinement system is protection by multiple barriers and systems, as required by 10 CFR §72.126(a), §72.126(b), and §72.126(c). The use of the HI-STORM 100 System, which is a sealed canister-based system, satisfies the requirements of 10 CFR §72.122(h)(1). Operating procedures, shielding design, and access controls provide the necessary radiological protection to ensure radiological exposures to facility personnel and the public are ALARA, as required by 10 CFR §72.126(d). The bounding dose rate design criteria are consistent with the requirements in 10 CFR §72.104(a) and §72.106(a).

The staff reviewed the design criteria for spent nuclear fuel storage and handling and determined that the criteria are appropriately identified as required by 10 CFR §72.128(a) and §72.128(b). In this SER, the staff shielding evaluation is presented in Chapter 7; the confinement evaluation in Chapter 9; and the radiation protection evaluation in Chapter 11. Evaluation findings given in this chapter are drawn from Chapters 7, 9, and 11 of this SER.

4.1.3.5 Criticality

The staff reviewed the discussion on criticality design criteria of SSCs with respect to the following regulatory requirements:

- 10 CFR §72.124(a) requires that spent nuclear fuel handling, packaging, transfer, and storage systems be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer, and storage conditions and in the nature of the immediate environment under accident conditions.

- 10 CFR §72.124(b) requires that, when practicable, the design of an ISFSI be based on favorable geometry, permanently fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design shall provide for positive means to verify their continued efficacy.

- 10 CFR §72.124(c) requires that a criticality monitoring system shall be maintained in each area where special nuclear material is handled, used, or stored that will energize clearly audible alarm signals if accidental criticality occurs. Monitoring of dry-storage areas where special nuclear material is packaged in its stored configuration under a license issued under 10 CFR Part 72 is not required.

Criteria used in criticality design of the cask systems are provided in the Diablo Canyon ISFSI SAR and the HI-STORM 100 System FSAR and are summarized in this SER, Table 4-6. The staff criticality evaluation is discussed in Chapter 8 of this SER. The design criteria for criticality are identified in the SAR as required by 10 CFR §72.124(a), §72.124(b), and §72.124(c).

4.1.3.6 Decommissioning

The staff decommissioning evaluation of SAR Section 4.6, "Decommissioning Plan," is presented in Chapter 13 of this SER.

4.1.3.7 Retrieval

The staff reviewed the discussion on retrieval design criteria of SSCs with respect to the following regulatory requirements:

- 10 CFR §72.122(l) requires that storage systems must be designed to allow ready retrieval of spent nuclear fuel for further processing or disposal.

- 10 CFR §72.128(a) requires that spent nuclear fuel storage, and other systems that might contain or handle radioactive materials associated with spent fuel must be designed to ensure adequate safety under normal and accident conditions.

The spent nuclear fuel will be stored in and handled with the HI-STORM 100 System, which has been approved for use under the general license provisions of 10 CFR Part 72. As discussed in the HI-STORM 100 System FSAR and the staff related SER, the HI-STORM 100 System is designed to ensure adequate safety and to protect fuel integrity and retrievability under the design-basis loads specified in the HI-STORM 100 System FSAR. The design-basis loads considered in the HI-STORM 100 System FSAR bound the structural and thermal loads found at the Diablo Canyon ISFSI except for the seismic load (see SER, Sections 4.1.3.2 and 4.1.3.3). For the seismic event, the applicant provided an analysis that demonstrated the HI-STORM 100SA Overpack would neither tip-over nor slide during a site-specific seismic event. Further, the loads on the canister would remain bounded by the canister loads considered in the HI-STORM 100 System FSAR. The seismic analysis is evaluated in Chapters 5 and 15 of this SER.

Based on the previous discussion, there is reasonable assurance that the HI-STORM 100 System will provide adequate safety and maintain fuel retrievability during the Diablo Canyon ISFSI site-specific conditions. Therefore, the staff finds that the requirements of 10 CFR §72.122(l) and §72.128(a) are satisfied.

4.1.4 Design Criteria for Other Structures, Systems, and Components

No specific requirements are identified in 10 CFR Part 72 for other SSCs not important to safety. Therefore, no evaluation findings are made in this section; only the information provided in the SAR is discussed. The design criteria for SSCs classified as not important to safety, but which have security or operational importance, are addressed in SAR Section 4.5.6, "Design Criteria for SSCs Not Important to Safety." The SAR specifies that these SSCs will be designed to comply with their applicable codes and standards to maintain the capability to mitigate the effects of off-normal or accident events.

4.2 Evaluation Findings

Based on the review of the information presented in the SAR, the following evaluation findings are made regarding the proposed Diablo Canyon ISFSI:

- The staff find that the materials to be stored at the Diablo Canyon ISFSI are appropriately identified as those that are approved for storage in the HI-STORM 100 System.

- The staff finds that the SSCs important to safety have been properly classified and their associated categories are consistent with the regulatory requirements of 10 CFR §72.144(a) and associated technical information content of the application, in accordance with 10 CFR §72.24(n). This list of SSCs is based on the definition in 10 CFR §72.3 of SSCs important to safety. The SAR appropriately specifies the design criteria for the SSCs important to safety in accordance with 10 CFR §72.120(a). The design criteria are to be included in the quality assurance procedures, as required in 10 CFR §72.144(a).

- The staff finds that the structural design criteria, given in SAR Section 3.2, "Structural and Mechanical Safety Criteria," considered for the SSCs important to safety, are developed from site characteristics and are used in the determination

of structural loads and load combination analyses. The values for these parameters form the basis for the structural design, mechanical design, and criticality assessment of the Diablo Canyon ISFSI. These design criteria satisfy the requirements of 10 CFR §72.120(a) and §72.122(h). Additionally, the SSCs important to safety will be designed to quality standards commensurate with important-to-safety functions performed to satisfy the requirements of 10 CFR §72.144(c).

- The staff finds that the seismic design criteria are appropriately identified in accordance with 10 CFR §72.120(a), §72.120(b), §72.120(c), §72.120(d), §72.120(e), §72.120(f) and §72.122(b). The seismic design criteria are in accordance with the site-specific seismic hazards analysis given in SAR Chapter 2, "Site Characteristics."

- The staff finds that the explosion considerations in the SAR are consistent with standard design criteria specified by Regulatory Guide 1.91, as required by 10 CFR §72.122(c). Design peak incident pressures have been defined.

- The staff finds that the load combinations design criteria are adequately considered for the design of SSCs, as required by 10 CFR §72.122(b). Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena have been considered.

- The staff finds that the bounding dose rate design criteria given in the SAR are consistent with the requirements of 10 CFR §72.104(a). The design criteria for spent nuclear fuel storage and handling have been properly specified, as required by 10 CFR §72.128. In this SER, the staff shielding evaluation is presented in Chapter 7, the confinement evaluation in Chapter 9, and the radiation protection evaluation in Chapter 11.

- The staff finds that design criteria for criticality are identified in the SAR, as required by 10 CFR §72.124(a), §72.124(b), and §72.124(c). A criticality evaluation is presented in Chapter 8 of this SER.

- The staff decommissioning findings are discussed in Chapter 13 of this SER.

- The staff finds that the Diablo Canyon ISFSI design, which includes use of the HI-STORM 100 System, allows for retrieval of the spent nuclear fuel in accordance with 10 CFR §72.122(l). Storage systems are designed to ensure adequate safety during normal and accident conditions in accordance with 10 CFR §72.128(a).

4.3 References

American Concrete Institute. *Code Requirements for Nuclear Safety Related Concrete Structures.* ACI 369-85. Detroit, MI: American Concrete Institute. 1985.

American Concrete Institute. *Building Code Requirements for Structural Concrete.* ACI 318-95. Detroit, MI: American Concrete Institute. 1995.

American Concrete Institute. *Code Requirements for Nuclear Safety Related Concrete Structures.* ACI 349-97. Detroit, MI: American Concrete Institute. 1998.

American National Standards Institute/American Nuclear Society. *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).* ANSI/ANS 57.9. La Grange Park, IL: American Nuclear Society. 1992.

American National Standards Institute/American Nuclear Society. *Radioactive Materials Special Lifting Devices for Shipping Containers Weighing 10,000* (4,500 kg). ANSI/ANS 14.6. La Grange, IL: American National Standards Institute. 1993.

American Society of Civil Engineers. *Minimum Design Loads for Buildings and Other Structures.* ASCE 7-98. New York City, NY: American Society of Civil Engineers. 2000.

ASME International. *ASME Boiler and Pressure Vessel Code, Section III, Division 1.* New York City, NY: ASME International. 1998.

ASME International. *ASME Boiler and Pressure Vessel Code, Section II, Materials, Part D–Properties. 1998 Edition with 1999 Addenda.* New York City, NY: ASME International. 1999.

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System Revision 1 (HI-STORM 100 Cask System).* Volumes I and II. HI–2002444. Docket 72-1014. Marlton, NJ: Holtec International. 2002.

National Fire Protection Association. *Flammable and Combustible Liquids Code.* NFPA 30. Quincy, MA: National Fire Protection Association. 1996.

Pacific Gas and Electric Company. *Dry Cask Storage System.* PG&E Specification 10012–N–NPG. Avila Beach, CA: Pacific Gas and Electric Company. September 2000.

U.S. Atomic Energy Commission. *Design Basis Tornado for Nuclear Power Plants.* Regulatory Guide 1.76. Washington, DC: U.S. Atomic Energy Commission. 1974.

U.S. Nuclear Regulatory Commission. *Evaluation of Explosives Postulated to Occur on Transportation Routes Near Nuclear Power Plants.* Regulatory Guide 1.91. Revision 1. Washington, DC: U.S. Nuclear Regulatory Commission. 1978.

U.S. Nuclear Regulatory Commission. *Control of Heavy Loads at Nuclear Power Plants.* NUREG–0612. Washington, DC: U.S. Nuclear Regulatory Commission. 1980.

U.S. Nuclear Regulatory Commission. Missiles Generated by Natural Phenomena. Revision 2, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.* LWR Edition. NUREG–0800 Section 3.5.1.4: (formerly issued as NUREG–76/087). Washington, DC: U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. July 1981.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Dry Cask Storage Systems.* NUREG–1536. Washington, DC: U.S. Nuclear Regulatory Commission. 1997.

U.S. Nuclear Regulatory Commission. 10 CFR Part 72 *Certificate of Compliance No. 1014, Amendment 1, for the HI-STORM 100 Cask System.* Docket No. 72-1014. July 15, 2002a.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report, Amendment 1.* Docket No. 72-1014. July 15, 2002b.

5 INSTALLATION AND STRUCTURAL EVALUATION

5.1 Conduct of Review

This chapter of the Safety Evaluation Report (SER) contains reviews of the information presented in Safety Analysis Report (SAR) Chapter 4, "ISFSI Design." The objective of the installation design review is to ensure compliance with the site features and to support other evaluation areas. The objective of the structural evaluation review is to ensure structural integrity of structures, systems, and components (SSCs) with emphasis on those that are important to safety. The review also considers selected sections and documents referenced in SAR Chapters 1, 2, 3, and 8. These chapters discuss general information, site characteristics, principal design criteria, and accident analysis.

Spent nuclear fuel dry storage facilities are designed for safe confinement and storage of the spent nuclear fuel. The design of the proposed Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) is based on the use of the HI-STORM 100 System, which has been reviewed by the U.S. Nuclear Regulatory Commission (NRC) and approved for general use under Certificate of Compliance (CoC) No. 1014 Amendment 1 (U.S. Nuclear Regulatory Commission, 2002a). The Diablo Canyon ISFSI relies on the HI-STORM 100 System, as described in the HI-STORM 100 System Final Safety Analysis Report (FSAR) Revision 1 (Holtec International, 2002), for confinement and radiological safety. Where applicable, staff relied on the review carried out during the certification process of the cask system, as documented in the HI-STORM 100 System SER Amendment 1 (U.S. Nuclear Regulatory Commission, 2002b). The major categories of safety protection systems discussed in the following sections include confinement SSCs, reinforced concrete structures, other SSCs important to safety, and other SSCs not important to safety.

Staff reviewed the Diablo Canyon ISFSI installation and structural evaluation with respect to the following regulatory requirements:

- 10 CFR §72.24(a) requires a description and safety assessment of the site on which the ISFSI is to be located, with appropriate attention to the design bases for external events. Such assessment must contain an analysis and evaluation of the major structures, systems, and components of the ISFSI that bear on the suitability of the site when the ISFSI is operated at its design capacity. If the proposed ISFSI is to be located on the site of a nuclear power plant or other licensed facility, the potential interactions between the ISFSI and such other facility—including shared common utilities and services—must be evaluated.

- 10 CFR §72.24(b) requires a description and discussion of the ISFSI structures with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

- 10 CFR §72.24(c) requires that the design of the ISFSI be described in sufficient detail to support the findings in §72.40, including (1) the design criteria for the ISFSI pursuant to subpart F of this part, with identification and justification for any additions to or departures from the general design criteria; (2) requires that the design of the ISFSI be described in sufficient detail to support the findings in Section 72.40, including the design bases and the relation of the design bases to

the design criteria. (3) requires that the design of the ISFSI be provided in sufficient detail to support the findings in Section 72.40, including information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all structures, systems, and components important to safety, in sufficient detail to support a finding that the ISFSI will satisfy the design bases with an adequate margin for safety. (4) requires that the design of the ISFSI be described in sufficient detail to support the findings in Section 72.40, including applicable codes and standards.

- 10 CFR §72.24(d) requires an analysis and evaluation be provided of the design and performance of structures, systems, and components important to safety, with the objective of assessing the impact on public health and safety resulting from operation of the ISFSI and including determination of the (1) margins of safety during normal operations and expected operational occurrences during the life of the ISFSI and (2) the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents, including natural and manmade phenomena and events.

- 10 CFR §72.24(i) requires the identification of any structures, systems, or components important to safety whose functional adequacy or reliability have not been demonstrated by prior use for that purpose or cannot be demonstrated by reference to performance data in related applications or to widely accepted engineering principles, along with a schedule showing how safety questions will be resolved prior to the initial receipt of spent fuel or high-level radioactive waste for storage at the ISFSI.

- 10 CFR §72.106(a) requires that a controlled area must be established.

- 10 CFR §72.120(a) requires that, pursuant to the provisions of §72.24, an application to store spent fuel in an ISFSI must include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance and performance requirements for structures, systems, and components important to safety as defined in §72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI.
- 10 CFR §72.122(a) requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed.

- 10 CFR 72.122(b) requires that (1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. (2)(i) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases

for these structures, systems, and components must reflect (A) appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (B) appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (2)(ii) The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent nuclear fuel or high-level waste or onto structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety. (4) If the ISFSI is located over an aquifer that is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

- 10 CFR §72.122(c) requires that structures, systems, and components important to safety be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability too minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

- 10 CFR §72.122(f) requires that systems and components that are important to safety be designed to permit inspection, maintenance, and testing.

- 10 CFR §72.122(g) requires that structures, systems, and components important to safety be designed for emergencies. The design must provide for accessibility to the equipment of onsite and available offsite emergency facilities and services such as hospitals, fire and police departments, ambulance service, and other emergency agencies.

- 10 CFR §72.122(h)(1) requires that the spent fuel cladding be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

- 10 CFR §72.122(h)(4) requires that the storage confinement systems have the capability for monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. For dry spent fuel storage, periodic monitoring is sufficient provided

that periodic monitoring is consistent with the dry spent fuel storage cask design requirements. The monitoring period must be based upon the spent fuel storage cask design requirements.

- 10 CFR §72.122(l) requires that storage systems be designed to allow ready retrieval of spent fuel for further processing or disposal.

- 10 CFR §72.128(a) requires that spent fuel storage and other systems that might contain or handle radioactive materials associated with spent fuel, be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with: (1) a capability to test and monitor components important to safety; (2) suitable shielding for radioactive protection under normal and accident conditions; (3) confinement structures and systems; (4) a heat-removal capability having testability and reliability consistent with its importance to safety; and (5) means to minimize the quantity of radioactive wastes generated.

- 10 CFR §72.236(b) requires that design bases and design criteria be provided for structures, systems, and components important to safety.

- 10 CFR §72.236(c) requires that the spent fuel storage cask be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

- 10 CFR §72.236(e) requires that the spent fuel storage cask be designed to provide redundant sealing of confinement systems.

- 10 CFR §72.236(f) requires that the spent fuel storage cask be designed to provide adequate heat removal capacity without active cooling systems.

- 10 CFR §72.236(g) requires that the spent fuel storage cask be designed to store the spent fuel safely for a minimum of 20 years and permit maintenance as required.

- 10 CFR §72.236(l) requires that the spent fuel storage cask and its systems important to safety be evaluated, by appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

5.1.1 Confinement Structures, Systems, and Components

The discussion about confinement SSCs is presented in SAR Section 4.2.3, "Storage Cask Description;" and in Chapter 4 of the HI-STORM 100 System FSAR (Holtec International, 2002). The staff reviewed the discussion about confinement SSCs with respect to the applicable regulatory requirements as discussed next.

5.1.1.1 Description of Confinement Structures

The Diablo Canyon ISFSI confinement structure is a spent nuclear fuel canister, specifically the Holtec multi-purpose canister (MPC) of the HI-STORM 100 System. Detailed descriptions of the MPC are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff previously reviewed and found this description acceptable, as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). SAR Section 4.2.3, "Storage Cask Description," referenced the HI-STORM 100 System FSAR and provided a summary description of the confinement structure. The staff found the summary to be consistent with the information in the HI-STORM 100 System FSAR. The confinement structure was sufficiently described in accordance with 10 CFR §72.24 and §72.236.

5.1.1.2 Design Criteria for Confinement Structures

The design criteria for the MPC are presented in the HI-STORM 100 System FSAR (Holtec International, 2002) and evaluated in the related SER (U.S. Nuclear Regulatory Commission, 2002b). A summary of the design criteria is contained in the SAR Table 3.4-2. Design criteria for the MPC have been shown in Chapter 4 of this SER to be representative of the site.

The MPC confinement boundary is designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Articles NG–3200, and NG–3300 (ASME International, 1995a). Fabrication of the MPC is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB–4000; and Subsection NG, Article NG–4000 (ASME International, 1995a). MPC inspection is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Articles NB–5000 and NG–5000 (ASME International, 1995a), and Section V (ASME International, 1995b).

MPC confinement boundary welding will be performed using welders and weld procedures that have been qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section IX (ASME International, 1995c); and Section III, Subsections NB and NG (ASME International, 1995a). Nondestructive examination of the MPC welds are specified in engineered drawings in the HI-STORM 100 System FSAR, Section 1.5 (Holtec International, 2002). MPC fabrication welds will be inspected using visual testing and radiographic testing or ultrasonic testing and penetrant testing in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB; Article NB–5300; and Subsection NG, Article NG–5300 (ASME International, 1995a).

Exceptions to the ASME and Pressure Vessel Code are provided in the CoC 1014-1, Appendix B, Table 3-1 (U.S. Nuclear Regulatory Commission, 2002a) and the Diablo Canyon ISFSI SAR Table 3.4-6 (Pacific Gas and Electric Company, 2002a).

The staff conclude that the design criteria of the confinement structures meet the requirements of the ASME Code, as applicable. The conclusions drawn in this section of the confinement structures design criteria are based on the evaluation findings made in Section 4.1.3 of this SER. The confinement structure design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c), §72.120(a), and §72.236(b).

5.1.1.3 Material Properties for Confinement Structures

Materials Selection

The applicant provided a description of the HI-STORM 100 System including materials of construction, fabrication details, and testing in SAR Sections 4.2, "Storage System;" and 4.7, "Operating Environment Evaluation;" and Appendix A, "Materials" (Pacific Gas and Electric Company, 2002a). Engineering drawings and additional details of the storage system are included by reference to the Holtec HI-STORM 100 System FSAR Sections 3.3 and 3.4 (Holtec International, 2002). Technical specifications for the HI-STORM 100 System are included by reference to 10 CFR Part 72 CoC No. 1014-1 (U.S. Nuclear Regulatory Commission, 2002a) for the HI-STORM 100 System. The HI-STORM 100 System has been evaluated by the staff and approved for use for dry storage of spent nuclear fuel (U.S. Nuclear Regulatory Commission, 2002b). The staff reviewed the information contained in these documents to determine compliance of the proposed Diablo Canyon ISFSI with the requirements of 10 CFR §72.24(c)(3), §72.24(c)(4), §72.122(a), §72.122(b), §72.122(c), §72.122(h), §72.122(i), and §72.236(g).

The structural components of the MPC are constructed from Types 304, 304LN, 316, or 316LN austenitic stainless steel (Holtec International, 2002). Stainless steels were selected based on mechanical properties and corrosion resistance. Shielding is provided by additional material thickness of the MPC shell, baseplate, lid, and closure ring (Holtec International, 2002). Applicable codes for the material procurement, design fabrication, and inspection of the MPC are provided in HI-STORM 100 System FSAR Table 2.2.7 (Holtec International, 2002) and in the Proposed Technical Specifications for Diablo Canyon ISFSI (Pacific Gas and Electric Company, 2002b). Material procurement is in accordance with ASME Boiler and Pressure Vessel Code, Section II (ASME International, 1995d,e,f) and Section III, Subsection NG, Article NG–2000 (ASME International, 1995a). Materials for the MPC baseplate, lid, closure ring, port cover plates, and shell are examined in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB–2500 (ASME International, 1995a). No ductile-to-brittle transition temperature exists for the austenitic stainless steel structural materials, so the MPC is not susceptible to brittle fracture. The staff concludes that the selection of these materials is acceptable for the MPC.

Welds

The MPC welds are characterized in engineered drawing 3923 in the HI-STORM 100 System FSAR Section 1.5 (Holtec International, 2002). The drawing includes standard welding symbols and notations in accordance with American Welding Society (AWS) Standard A2.4 (American Welding Society, 1998). The stainless steel materials for the MPC are readily weldable using commonly available welding techniques. MPC closure welds are inspected using visual testing and ultrasonic testing or multilayer penetrant testing. If multilayer penetrant testing is used, the examination will include penetrant tests for the root and final passes and for each approximately 0.95 cm [3/8 in] of weld depth consistent with the minimum flaw size for the MPC identified in Holtec position paper DS–213 (Holtec International, 1999). Additional details of the nondestructive examination of the lid-to-shell weld, structural and pressure tests, and hydrostatic testing of the MPC are provided in HI-STORM 100 System FSAR Section 9.1 and Table 9.1.1 (Holtec International, 2002). The staff conclude that the welded joints of the MPC meet the requirements of the ASME and AWS codes, as applicable.

Mechanical Properties

Mechanical properties of the structural materials for the MPC, are provided in the Holtec HI-STORM 100 System FSAR Section 3.3 and Tables 3.3.1 and 1.A.1 (Holtec International, 2002) for stainless steels. The mechanical properties of the stainless steel structural materials, such as design stress intensity (S_m), tensile strength (S_u), yield strength (S_y), coefficient of thermal expansion (α), and coefficient of thermal conductivity (k), vary with stainless steel composition. Qualification of the MPC structure is accomplished using the least favorable mechanical and thermal properties of the entire group for all mechanical, structural, neutronic, radiological, and thermal conditions. Mechanical properties of the stainless steel structural materials are provided in the HI-STORM 100 System FSAR, Tables 3.3.1 and 1.A.1 (Holtec International, 2002). The values in these tables were obtained from ASME Code Section II Part D (ASME International, 1995f). The staff independently verified the temperature-dependent values for the stress allowables, ultimate strength, yield strength, modulus of elasticity, and coefficient of thermal expansion. The staff conclude that these material properties are acceptable and appropriate for the expected load conditions during the license period.

Coatings

No coatings are used on the MPC.

Chemical and Galvanic Reactions

Evaluation of possible chemical, galvanic, and other reactions among the materials in the range of possible exposure environments is included in SAR Section 4.7, "Operating Environment Evaluation," and Table 4.7-1 (Pacific Gas and Electric Company, 2002a). The evaluation includes stainless steels used in the MPC. The staff concur that no adverse reactions are anticipated for stainless steels used in the MPC.

Based on the previous discussion of the materials selection welds, mechanical properties, coating, and chemical and galvanic reaction the staff concludes that the material selection for the confinement structures meets the requirements of the ASME codes, as applicable. The material properties for confinement structure have been identified in accordance with 10 CFR §72.24(c)(3), §72.120(a), §72.122, and §72.236(b).

5.1.1.4 Structural Analysis for Confinement Structures

The staff reviewed the discussion of the MPC design relative to the storage requirements of the Diablo Canyon ISFSI provided in SAR Section 4.2.3, "Storage Cask Description." The Diablo Canyon ISFSI SAR provides a summary of the analysis performed in the HI-STORM 100 System FSAR (Holtec International, 2002). The Diablo Canyon ISFSI SAR Section 4.2.3.3.2, "Structural Design," states that the MPC has the structural capability to withstand the loads created by all design basis normal, off-normal, and accident conditions and for the design basis natural phenomena. The following loads and combined loading conditions were considered.

- Dead and Live Loads (SAR Section 4.2.3.3.2.1)
- Internal and External Pressure Loads (SAR Section 4.2.3.3.2.2)
- Thermal Expansion (SAR Section 4.2.3.3.2.3)

- Handling Loads (SAR Section 4.2.3.3.2.4)
- Overpack/Transfer Cask Tip-Over and Drop (SAR Section 4.2.3.3.2.5)
- Tornado Winds and Missiles (SAR Section 4.2.3.3.2.6)
- Flood (SAR Section 4.2.3.3.2.7)
- Earthquake (SAR Section 4.2.3.3.2.8)
- Explosion Overpressure (SAR Section 4.2.3.3.2.9)
- Fire (SAR Section 4.2.3.3.2.10)
- Lightning (SAR Section 4.2.3.3.2.11)
- 500-kV Line Drop (SAR Section 4.2.3.3.2.12)

The detailed structural analysis of confinement structures is presented in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff previously reviewed this structural analysis and found it acceptable, as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). As documented in that SER, the structural analysis shows the structural integrity of the HI-STORM 100 System is maintained under all credible loads. Based on the results presented in the HI-STORM 100 System FSAR, the stresses in the confinement structures under the most critical load combinations are less than the allowable stresses of ASME Boiler and Pressure Vessel Code Section III (ASME International, 1995a) for the confinement structures materials.

The loading conditions at the ISFSI are enveloped by the loading conditions considered in the HI-STORM 100 System FSAR. The applicant did not perform any drop or tip-over analysis. Outside the Fuel Handling Building and Auxiliary Building (FHB/AB), tip-over of the HI-STORM 100SA Overpack is considered a noncredible accident. When not on the ISFSI pad, the system will either be in the Cask Transfer Facility (CTF) or attached to the Cask Transporter, both of which are designed, fabricated, inspected, operated, maintained, and tested in accordance with NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). A hypothetical tip-over of a HI-STORM 100SA Overpack because of handling, a seismic event, or tornado winds with concurrent impact of the tornado-driven design missile (an automobile) at the top of the storage cask is noncredible. When on the storage pad, the casks are anchored to the pad. As demonstrated in the HI-STORM 100 System FSAR, an anchored cask will not tip-over, therefore, tip-over of the cask is noncredible.

Therefore, the staff conclusions in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b) with respect to the structural integrity of the MPC are valid for the Diablo Canyon ISFSI and, therefore, meets the Diablo Canyon ISFSI design criteria given in SAR Section 3.3, "Design Criteria for Safety Protection Systems." The confinement structure analysis has demonstrated compliance with 10 CFR §72.24(a), §72.24(d), §72.122(b), §72.122(c), §72.122(h), §72.128(a), and §72.236(l).

A site-specific analysis was performed to assess the vulnerability of the MPC at the CTF to a transmission tower collapse (Holtec International, 2001a). The transmission tower collapse looked at the vulnerability of the MPC lid during transfer at the CTF. The analysis results indicate there would be localized yielding of the MPC in the immediate vicinity of the impact site. Based on the results of the analysis the resultant material stress, away from the vicinity of the impact, does not exceed allowable values. Because the yielding is localized, there is no loss of shielding or confinement from the sealed MPC and no loss of retrievability of a spent nuclear fuel assembly from the MPC.

5.1.2 Pool and Pool Confinement Facilities

This provision is not applicable to 10 CFR Part 72 dry storage facilities.

5.1.3 Reinforced Concrete Structures

This section contains a review of SAR Section 4.2.1.1, "Cask Storage Pad." The staff reviewed the discussion about reinforced concrete structures important to safety with respect to the applicable regulatory requirements, as discussed below.

5.1.3.1 Description of Reinforced Concrete Structures

There is one reinforced concrete structure in the Diablo Canyon ISFSI that has been classified as important to safety, the ISFSI storage pad with overpack anchorage hardware (QA Category B), with the design and analysis information given in SAR Section 4.2.1.1, "Cask Storage Pads." The ISFSI storage pads are independent structural units constructed of reinforced concrete, designed in accordance with ACI 349-97 (American Concrete Institute, 1998) and Draft Appendix B, for the anchorage hardware. Figure 4.2-1 of the SAR shows a schematic of the storage pads. Each pad is 20.7 × 32 × 2.3 m [68 × 105 × 7.5 ft] and is capable of supporting 20 loaded HI-STORM 100SA Overpacks. The pad thickness may vary from 2.3 to 2.4 m [7.5 to 8 ft], based on the interface to the foundation rock and the 1 percent slope on the top surface for drainage. The size of the pad is based on a 5.2-m [17-ft] center-to-center spacing of the storage casks. It provides a level and stable surface for placement and storage of the storage casks. The ISFSI Storage Pad design is based on the maximum loaded weight of a storage cask of 163,300 kg [360,000 lb], the weight of the HI-STORM 100SA Overpack loaded with an MPC canister. The storage pad also provides the necessary embedment for the anchorage hardware for the HI-STORM 100SA Overpacks.

The overpack anchorage hardware is shown in the SAR Figures 4.2-1 and 4.2-2. The ISFSI storage pad is designed with a 50.8-mm-[2-in-] thick steel plate ring at the surface of the concrete that mates with the bottom of the cask. The base plates are designed to provide sufficient bearing area on the concrete to transfer loads. Each cask is attached to the pads using 16 studs that are threaded into a coupling steel block located on the underside of the embedment plate. Each stud is preloaded to approximately 71,200 kg [157,000 lb] by threading the stud into the coupling block, applying tension to the stud, and installing the nut on the stud to maintain the preload. The preloaded anchor studs are used to ensure that interface contact is maintained between the ISFSI pad embedment upper surface and the lower surface of the HI-STORM 100SA Overpack baseplate. Shear loads for each cask will be carried through the embedment plate/coupling block into the concrete. Long steel rods are embedded in the concrete to transfer the load from the cask to the concrete.

The SAR provides a design description of the ISFSI storage pads with overpack anchorage hardware in sufficient detail to support a detailed review and evaluation in accordance with 10 CFR §72.24(a) and §72.24(b).

Description of the ISFSI storage pads and associated operations procedures include consideration of inspection, maintenance, and testing as required in 10 CFR §72.122(f). The design of the reinforced concrete pads, a slab with an upper surface level with the surrounding grade, provides access to all locations and allows access to the storage casks in the event of

emergencies. There are no barriers built into the ISFSI storage pads that would prevent access to any location on the pads adjacent to the storage casks. This design allows emergency response capability, as required in 10 CFR §72.122(g). This reinforced concrete slab embedded in a rock foundation also incorporates the capability for retrieving the spent nuclear fuel canisters. The cask transporter can drive onto the pad to access any storage cask and transport it back to the CTF. Because of the rock foundation, settlement of the pad is considered insignificant, as shown in this SER Section 5.1.3.4; stress in the rock under static and dynamic loads is significantly below allowables. Consequently, the storage cask can be retrieved from the storage pads in accordance with 10 CFR §72.122(l).

5.1.3.2 Design Criteria for Reinforced Concrete Structures

The design bases for the reinforced concrete ISFSI storage pads with overpack anchorage hardware are given in the SAR Section 3.3.2, "ISFSI Concrete Storage Pad;" and SAR Section 4.2.1.1, "Cask Storage Pads." Table 3.4-3 of the SAR identifies details of the Diablo Canyon ISFSI compliance with the general design criteria of 10 CFR Part 72, Subpart F. This conclusion is also supported by the structural analysis described in Section 5.1.3.4 of this SER. Design criteria for the storage pad have been shown in Chapter 4 of this SER to be representative of the site.

ISFSI storage pads are designed in accordance with ultimate strength design methods specified in ACI 349-97 (American Concrete Institute, 1998). The ACI 349-97 Code specifies the minimum requirements for the design and construction of nuclear safety-related concrete structures and structural elements for nuclear power generating stations. Draft Appendix B of ACI 349-97, identifies requirements for steel embedments. Appendix D of ACI 318-02 (American Concrete Institute, 2002) is the most recent revision to the design requirements for anchors in concrete to transmit structural loads. The design requirements identified are commonly accepted by the construction industry but have not been explicitly adopted by the NRC. The procedures are acceptable but the load cases must be supplemented by the load cases of American National Standards Institute (ANSI) 57.9 (American National Standards Institute/American Nuclear Society, 1992). A feature of the design requirements is a design that results in a ductile failure of the metal components of the anchorage system prior to brittle failure of the concrete. The applicant uses the guidance of Draft Appendix B to define the requirement for a ductile failure of the steel portion of the anchorage hardware prior to brittle failure of the reinforced concrete.

The design criteria for the ISFSI storage pads with overpack anchorage hardware establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for reinforced concrete storage pads. Additionally, the design criteria of the storage pads address site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events. The conclusions drawn in this section on the storage pad design criteria are based on the evaluation findings made in Section 4.1.3 of this SER. The reinforced concrete structures design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c) and §72.120(a).

5.1.3.3 Material Properties for Reinforced Concrete Structures

The staff reviewed the materials of construction of the ISFSI storage pads, as identified in SAR Section 4.2.1.1. Material selection is based on concrete with a compressive strength of

34.5 MPa [5,000 psi] at 90 days following Chapter 5 of ACI 349-97 (American Concrete Institute, 1998). The reinforcing steel is specified to have a minimum yield strength of 414 MPa [60,000 psi] following Chapter 3 of ACI 349-97 (American Concrete Institute, 1998) and American Society for Testing and Materials (1990). Materials of construction of the overpack anchorage hardware, as identified in SAR Section 4.2.1.1, "Cask Storage Pads," include SA193–B7 studs, SA516 Grade 79 receptacles, and A–36 plates and bars. As identified in SAR Section 3.3.2.3, "Design Criteria," the exposed portions of this hardware will be coated to protect them from the environment.

Based on the review of information presented by Pacific Gas and Electric Company (PG&E), the staff conclude that materials to be used to construct the ISFSI storage pads with anchorage hardware have been adequately identified in accordance with 10 CFR §72.24(c)(3). The applicant has identified the appropriate codes and standards in accordance with 10 CFR §72.24(c)(4).

5.1.3.4 Structural Analysis for Reinforced Concrete Structures

The Diablo Canyon ISFSI reinforced concrete structures, as described in the SAR, are designed to meet the requirements of ACI 349-97 (American Concrete Institute, 1998). The staff accepts the strength design method, as presented in the ACI 349-97, for concrete structures important to safety. The reinforced concrete storage pads important to safety were analyzed for normal, off-normal, and accident-loading conditions. These analyses were carried out to ensure the storage pads would be able to perform their intended safety functions under the extreme environmental and natural phenomena, as specified in 10 CFR §72.122(b)(1) and §72.122(b)(2) and ANSI/ American Nuclear Society (ANS) 57.9 (American National Standards Institute/American Nuclear Society, 1992). The ultimate strength method of analysis is used with the appropriate load factors for the following loads:

- Dead loads (D)
- Live loads (L)
- Soil pressure loads (H)
- Temperature gradients (T)
- Wind loads (W)
- Earthquake loads (E)
- Accident (A) loads including explosion over pressure, drop/tip-over, accidental pressurization, fire, and aircraft impact
- Design-basis tornado wind loads and tornado-generated missile loads (W_t)
- Probable maximum flood loads (F)
- Lightning

The staff reviewed the SAR and found that the structural analysis procedures have been identified and are in conformance with standard engineering practice, as described in ACI 349-97 (American Concrete Institute, 1998). The relationship between the design criteria, identified in Chapter 3 of the SAR, and the analysis procedures was established in accordance with the requirements of 10 CFR §72.24(c)(2). The applicable codes and standards used in the analysis of the reinforced concrete structures also have been identified in the SAR, in accordance with the requirements of 10 CFR §72.24(c)(4).

Analysis of the stability of the subsurface materials under the storage pads and the potential for failure along the clay bed and tilting of the storage pads was reviewed in this SER Section 2.1.6.4, "Stability of Subsurface Materials." The slope stability of the material above the storage pads and the potential for it to impact and/or cover the storage casks on the storage pads was reviewed in the SER Section 2.1.6.5, "Slope Stability."

The SAR Section 8.2.1.2.3.1, "Cask and Anchorage Seismic Analysis," summarizes seismic analysis of the cask and anchorage system performed by Holtec. Although the Diablo Canyon site-specific seismic zero period accelerations (ZPAs) for all events are lower than those identified in Appendix B of the Holtec CoC (U.S. Nuclear Regulatory Commission, 2002a), Holtec International performed an analysis of the anchored HI-STORM 100SA Overpack at the Diablo Canyon ISFSI (Holtec International, 2001c). The primary reason for this analysis was a difference in the number of anchor rods identified for the Diablo Canyon ISFSI with respect to the design basis given for the HI-STORM 100 System (Holtec International, 2002). The Diablo Canyon ISFSI anchorage system uses sixteen 63.5-mm- [2.5-in] diameter rods, and the HI-STORM 100 System generic design calls for twenty-eight 50.8-mm [2-in] diameter rods, Section 3.4 of CoC No. 1014 (U.S. Nuclear Regulatory Commission, 2002a). The objectives of the analysis were to

(1) Demonstrate that the seismic events do not induce acceleration levels that exceed the design basis. Based on comparison of the maximum net shear at the base of the embedment plate to the bounding cask weight, the effective "g" is 1.43 g. The results indicate that the casks do not develop body decelerations that exceed the cask design basis of 45 g.

(2) Demonstrate that the seismic events do not induce stress in the preloaded anchor studs, cask flange, and shell that exceed the design basis ASME Code limits. This included a demonstration of all structural safety factors greater than 1 for ASME load Level A (preload) and Level D (Preload + Seismic Load). For the Level A load, a 72,600-kg [160,000-lb] load is assumed to act on the components that bound the actual 71,200-kg-[157,000 lb-] preload. For the Level D load, the maximum load per stud was 97,250 kg [214,400 lb], which is below the ultimate capacity of 97,800 kg [215,600 lb]. These loads were then applied to a local model of the cask flange and shell. Table 2 of Holtec International (2001c) identifies the factors of safety for these load conditions. The minimum factor of safety is 1.089, associated with a weld shear stress. In addition, the alternating stress intensity under the seismic loading must be sufficiently low that a factor of safety against fatigue failure is demonstrated. The estimated fatigue cycles required to fail the stud and sector lug are significantly higher than the number of cycles during a single earthquake event.

(3) Establish interface loads transferred to the ISFSI pad embedment. The peak interface loads at the lower surface of the embed plate are summarized in Table 3 of Holtec International (2001c). The resultant interface loads are identical to those identified in Table 2 of the storage pad seismic analysis (ENERCON Services Inc., 2001b).

The staff reviewed the design documentation to verify the assumptions, analysis procedures, modeling procedures, and summary of results. The dynamic analysis employed a

three-dimensional model of the MPC, overpack, and anchoring system using VisualNastran 2001 (MSC Software Corporation, 2001). As identified in Table 1 of Holtec International (2001c), the input data for the model are consistent with the HI-STORM 100 System FSAR. Relative motion and impact between the MPC and overpack are properly modeled. The interface between the embedment and the ISFSI concrete is properly modeled by discrete springs to simulate the anchor rods and the compression only concrete. Of the four seismic events for the Diablo Canyon site, only the Long-Term Seismic Program (LTSP) and Hosgri Earthquake (HE) events were used for the dynamic simulation because they impart the highest acceleration and, therefore, highest loading to the anchorage components. The appropriate materials properties were identified and used. The staff concur with the results presented in SAR Section 8.2.1.2.3.1, "Cask and Anchorage Seismic Analysis."

SAR Section 8.2.1.2.3.2, "Storage Pad Seismic Analyses," identifies the analysis performed to ensure that the reinforced concrete pads and the anchored casks remain functional during all seismic conditions. Two analyses are covered in this section, a static analysis (ENERCON Services, Inc., 2001b) and a nonlinear pad sliding analysis (Pacific Gas and Electric Company, 2001d).

The static analysis (ENERCON Services Inc., 2001b) determined the storage pad size and thickness required to resist the loads resulting from seismic accelerations applied to the pad and resultant loads from the cask dynamic analysis (Holtec International, 2001c). The limiting parameter considered in this assessment is that the pad displacement under the pad dead weight and seismic loads be held to an acceptable value (displacement < length/320), ACI 349-97 (American Concrete Institute, 1998). The HE and LTSP spectra were used because these spectra produce the largest ZPAs. Load combinations for the sequencing of cask placement were considered. The ANSYS nonlinear static finite element model consisted of the pad, a portion of the underlying rock foundation, and the casks on top the pad. The model was constructed of three-dimensional solid elements. All material properties were linear, and the concrete values are consistent with the facility design. The rock foundation Young's modulus was varied from 1,380 to 33,800 MPa [0.2×10^6 to 4.9×10^6 psi] (representing soft, hard, and very hard rock). The extent of the rock foundation was sufficient that the boundaries did not influence the response at the storage pad location. The casks are modeled up to the plane of their center of gravity. The Young's modulus for the cask was adjusted to force the fundamental frequency of the cask to match that obtained by Holtec International (2001c). Compression only gap elements were used to model the interface between the slab (target surface) and rock foundation (contact surface). The element stiffness and convergence parameters were computed from the geometry and material properties of the adjoining elements. Shear loads were transmitted from the pad into the rock foundations through constraint equations. The bounding load cases were derived from the results of the Holtec analysis (Holtec International, 2001c) and applied to the center of gravity of the storage casks. In addition to the cask loads, an inertia force was applied to the pad with reference to the ZPA of the seismic event. The Newmark 100-40-40 method was used to combine the three specified directions of the seismic load. A total of 19 discrete load steps were applied to the model.

The results of each load step were checked for equilibrium and found to be acceptable. The results were post-processed to obtain the maximum displacement of the pad, maximum displacement of the centerline of the casks on the perimeter of the array, vertical displacement of the pad for the soft rock model, max/min stresses in the X and Y horizontal directions, maximum principal stress (largest tensile stress), minimum principal stresses (largest

compressive stress). The results are summarized in Tables 2 to 5 of the ENERCON calculation package (ENERCON Services Inc., 2001b). The pad and cask vertical displacement are small and within acceptable limits. The maximum tensile stress in the concrete is 2.48 MPa [359 psi] and is less than the tensile stress, 3.65 MPa [530 psi], that will cause cracking in the 34.5-MPa [5,000-psi] concrete. The maximum compressive stress is 5.56 MPa [806 psi], significantly less than the 34.5-MPa [5,000-psi] design value.

The analysis demonstrated that the greatest demand on the slab was for the HE seismic event. It was also determined that the mass of the pad and its inertial loads were as important to the overall response of the pad as the applied loads from the casks. Sections throughout the pad were isolated for the HE seismic event, and the internal forces acting upon them were computed. The results are summarized in Tables 6 to 10 of the ENERCON calculation package (ENERCON Services Inc., 2001b). The resulting internal forces for design purposes are given in Table 11 of the ENERCON calculation package (ENERCON Services Inc., 2001b). A factor of 1.15 will be applied to these loads to account for potential variations due to Poisson's ratio of the rock foundation.

The results were scanned to identify the stress in the rock below the pad. The applied gravity pressure was 0.11 MPa [2.36 ksf], which is less than the allowable bearing pressure of 1.91 MPa [40 ksf]. The maximum seismic pressure calculated was 0.33 MPa [6.95 ksf] for the HE load case and 0.31 MPa [6.42 ksf] for the LTSP load case, each is less than the allowable bearing pressure of 2.49 MPa [52 ksf]. A review of the stability against bearing capacity failure under static and dynamic loading is contained in this SER Section 2.1.6.4, "Stability of Subsurface Materials."

The reduced weight of concrete (90 percent of the design value) to account for construction variability resulted in displacements only 11 percent greater, still within the acceptable range, and lower stresses. The analyses for placement sequence configurations and cask extraction were shown to be bounded by the fully loaded analyses.

The results of the analysis were used in Calculation No. PGE–009–CALC–007 (ENERCON Services Inc., 2003a) to evaluate the concrete per the design codes and to determine the size of the steel reinforcement in compliance with the requirements of ACI 349-97 (American Concrete Institute, 1998). The analysis (PGE–009–CALC–007) has been completed and is in compliance with the requirements of ACI 349-97 (American Concrete Institute, 1998).

A nonlinear analysis was performed to determine the extent of sliding at the pad/rock interface (Pacific Gas and Electric Company, 2001d). The ISFSI long period (ILP) time histories were used because the pad sliding may be sensitive to long-period ground motion. This analysis used a lumped mass model of 20 casks on a pad using SAP2000. The pad was modeled using a small lumped mass. The interface between the pad and the rock foundation was a biaxial friction element that has coupled fiction properties for the two shear deformation, post-slip stiffness in the shear directions, and gap behavior in the axial direction. The review of stability against sliding under dynamic loading is contained in this SER Section 2.1.6.4, "Stability of Subsurface Materials."

The anchorage system components for the HI-STORM 100SA Overpack provide a level surface for the casks to set on, 16 receptacles for the anchorage studs, and strength to transmit cask loads caused by external events to the concrete pads (ENERCON Services Inc., 2003b). The

loads applied correspond to those calculated by Holtec (Holtec International, 2001c). The anchorage system was designed to meet the ductile anchorage provision of the Proposed Draft, Appendix B for ACI 349-97 (American Concrete Institute, 1998). The anchor bars are 63.5-mm-[2.5-in-] diameter A-36 steel bars to carry the applied load and have the appropriate stiffness, approximately 28,600 kg/mm [1.9×10^6 lb/in]. The A-36 steel was chosen because it has a well-defined yield plateau, which gives a ductile design strength of 1,625 MPa [235.63 kips]. The minimum yield strength of the anchor bars is more than 250 percent of the computed demand load, 428 MPa [62.13 kips]. In addition, the anchorage system was designed so the anchor plates, attached to the bottom of the anchor bars, are of a sufficient size, 19.5 × 19.5 cm [7.5 × 7.5 in], to transfer the load by bearing on the concrete. The length of the coupler, 29.8 cm [11.72 in], is controlled by the engagement length for the threads for both the anchor studs and the round bar. The diameter is larger than the corresponding heavy hex bolts used for the round bars and has sufficient capacity to carry the applied loads. The geometry of the embedment plate is controlled by the geometry of the storage cask {inner diameter = 3.30 ± 0.01 m [130 ± 0.25 in], outer diameter = 3.78 ± 0.01 m [149 ± 0.25 in]} and the slope of the storage pad {50.8 mm [2 in] thick}.

To satisfy the requirements of Appendix B of ACI 349-97 (American Concrete Institute, 1998), the concrete breakout strength of anchor in tension and shear and pullout strength of anchor in tension must exceed the anchor bar ductile design strength of 1,625 MPa [235.63 kips]. The applicant has demonstrated the sizing and placement of reinforcing steel within the reinforced concrete storage pad are sufficient to ensure a ductile failure (ENERCON Services Inc., 2003b) The applicant has provided sufficient reinforcing steel to ensure that the failure cone for concrete pullout intersects sufficient rebar to prevent this brittle failure (ENERCON Services Inc., 2003b).

In response to requests for additional information (RAI), the applicant provided a calculation package that determines the shrinkage and thermal stress in the massive concrete storage pad (ENERCON Services Inc., 2003c). The forces and moments, together with the seismic forces and moments (ENERCON Services Inc., 2001b), were used to demonstrate that the design is compliant with the ACI code and to size the pad reinforcement (ENERCON Services Inc., 2003a). The shrinkage and thermal load data were provided to ENERCON Services, Inc. by Pacific Gas and Electric Company (2001a). The temperature data are internal concrete pad temperatures as a function of time that occur as the heat generated by cement hydration is dissipated through the structure to the surrounding environment. The heating transient caused by hydration lasted for 8 days. The loads applied consisted of incremental temperatures at various locations within the pad at the time of concern (ENERCON Services Inc., 2003c, Table 2). The shrinkage strains are a result of the moisture loss and the moisture gradient established in the concrete. The shrinkage data, provided in terms of microstrain through the pad thickness, are converted into temperatures for use in the model. The shrinkage process is very slow, reaching a maximum shrinkage in 117 days. Therefore, the two effects can be considered separately.

The analysis was performed using a three-dimensional ANSYS FEA model. Because the load conditions are symmetrical, one quarter of the rock foundation was modeled. The concrete pad and rock were modeled using eight-noded brick elements. Concrete materials properties were varied in the model, with respect to time, to account for the increase in strength (ENERCON Services Inc., 2003c, Table 5). A 0.6-m [2-ft] gap is provided around the pad to allow for some consideration for construction access. Only the stiff rock foundation,

13,800 MPa [2.0 × 10^6 psi], is considered because this will result in the maximum restraint of the pad. Constraint equations and contact elements are used to model the contact between the bottom of the pad and the rock foundation. The appropriate boundary conditions were applied to account for using a one-quarter symmetry model.

A check of the temperatures at the specified locations for the various load steps showed they were consistent with the load data (Pacific Gas and Electric Company, 2001a). In addition, a check showed that the overall equilibrium of the storage pad and rock foundation was maintained. The vertical displacements (ENERCON Services,, Inc., 2003c, Table 7) in the pad bottom, 15.3 mm [0.602 in], are significantly greater than for the seismic loads, 2.57 mm [0.101 in], but are still less than the deflections of 65 to 100 mm [2.55 to 3.9 in] (length/320) caused by dead weight. The internal stresses in the pad (ENERCON Services Inc., 2003c, Table 8) are such that the tensile stress is greater than the allowable; therefore, significant reinforcing will be required to prevent cracking. The pad internal forces and moments (ENERCON Services Inc., 2003c, Table 9) were used to size reinforcement and assess the level of expected cracking. These values are significantly greater than those for the seismic analysis and will therefore control the design. The results of the analysis were used in Calculation No. PGE–009–CALC–007 (ENERCON Services Inc., 2003a) to evaluate the concrete according to the design codes and to determine the size of the steel reinforcement in compliance with the requirements of ACI 349-97 (American Concrete Institute, 1998).

These static and dynamic analyses confirm the structural adequacy of the reinforced concrete storage pad for supporting the storage casks when subjected to the design loading conditions. From the static and dynamic analyses, pad responses were obtained and then combined to give the maximum response values in accordance with the applicable load combinations. The combined response values were then used to check the structural adequacy of the concrete pad and the soil bearing and sliding stabilities. The structural analysis performed by PG&E demonstrated that the ISFSI storage pads are adequately designed to resist the loads based on the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events in accordance with the requirements of 10 CFR §72.122 (b)(1). The structural analysis carried out by the applicant demonstrates that the ISFSI Storage Pads are designed to withstand the effects of natural phenomena, such as earthquakes, without impairing the capacity to perform safety functions in accordance with the requirements of 10 CFR §72.122(b)(2).

For the slab-on-grade design of the storage pads, the tornado winds will not exert any additional load to the structure. Additionally, the ISFSI storage pad will not be subjected to flood load because the storage pads will be above the maximum probable flood level. Moreover, lightning strikes will not affect the safety function of the pad because it is grounded. Therefore, the ISFSI Storage Pads are designed to withstand the effects of natural phenomena, such as tornadoes, lightning, and floods without impairing the capacity to perform safety functions in accordance with the requirements of 10 CFR §72.122(b)(2).

The Diablo Canyon ISFSI concrete storage pads are massive reinforced concrete structures made of noncombustible materials and surrounded by an open gravel surface. The gravel surface will be kept free of growth so no combustibles will be present. The staff has reviewed the SAR and determined that the design of the ISFSI storage pads provides adequate fire protection. SAR Section 8.2.5.2, "Fire," indicates the ISFSI system will not be affected by the postulated combustion of local fuel tanks, combustible materials outside the ISFSI storage pad

perimeter or along the transport route, or as unsuppressed vegetation fire and, therefore, the requirements of 10 CFR §72.122(c) are satisfied.

5.1.4 Other Structures, Systems, and Components Important to Safety

This section contains a review of SAR Sections 4.2.1, "Structures;" 4.2.3, "Storage Cask Description;" 4.3, "Transport System;" 4.4, "Operating Systems;" and 8.2, "Accidents." The staff reviewed the discussion of other SSCs important to safety with respect to the applicable regulatory requirements.

5.1.4.1 Description of Other Structures, Systems, and Components Important to Safety

The following structures and components were identified in the SAR as other SSCs important to safety. The staff reviewed the description of SSCs important to safety with respect to the regulatory requirements of 10 CFR §72.24(b), §72.122(f), and §72.122(g).

Fuel Basket and Damaged Fuel Container [Quality Assurance (QA) Category A] and Upper and Lower Fuel Spacer Columns and End Plates (QA Category B)

The Diablo Canyon ISFSI MPC may include the Holtec fuel basket, SAR Section 4.2.3.2.1, "MPC;" damaged fuel container (DFC), SAR Section 4.2.3.2.2, "DFC;" and components of the HI-STORM 100 System. The fuel basket provides support for the fuel assemblies as well as the geometry and fixed neutron absorbers for criticality control. The lower fuel spacer columns and end plate are part of the structure of the fuel basket. The DFC provides a basket to contain damaged fuel that can be placed in one of the positions in the fuel basket. Detailed descriptions of the fuel basket and DFC are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff previously reviewed and found these descriptions acceptable, as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002a). The fuel basket and DFC structure were sufficiently described in accordance with 10 CFR §72.24(b).

HI-TRAC 125 Transfer Cask (QA Category A)

The HI-TRAC 125 Transfer Cask, as identified in SAR Section 4.2.3.2.4, "HI-TRAC 125 Transfer Cask" is a heavy-walled cylindrical vessel constructed of carbon steel with water for neutron and lead for gamma shielding. The transfer cask provides an internal cylindrical cavity of sufficient size for housing a HI-STORM 100 System MPC. The transfer cask is designed for transient use, to contain the MPC and to be submerged in the spent fuel pool (SFP) to support fuel loading. SAR Figures 4.2.8 to 4.2-10 show the major components of the transfer cask. Table 4.2-3 of the SAR identifies the physical characteristics of the HI-TRAC 125 Transfer Cask. Detailed design descriptions of the transfer cask are given in the HI-STORM 100 System FSAR (Holtec International, 2002).

The description of the transfer cask includes consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). Components requiring inspection and maintenance are identified, and operational procedures are summarized. Preoperational, startup, and operational tests will be performed to verify the

functional operations of SSCs important to safety. This design also allows for emergency load carrying capability. Design of the transfer cask allows for control of loads in the event of emergencies. The HI-TRAC 125 Transfer Cask has been sufficiently described in accordance with 10 CFR §72.24(b), §72.122(f), and §72.122(g).

HI-TRAC Lift Links, MPC Downloader Slings, MPC Lift Cleats, HI-STORM 100 System Lifting Brackets (QA Category A), Transporter Connection Pins, Transfer Cask Horizontal Lift Rig, Transfer Cask Lift Slings (QA Category B)

As identified in the SAR Sections 4.3.2.2, "Transfer Cask Horizontal Lift Ring;" 4.3.2.3, "Transfer Cask Lift Slings;" and 4.3.2.5 through 4.3.2.9, "HI-TRAC Lift Links;" "MPC Downloader Slings;" "MPC Lift Cleats;" "HI-STORM Lifting Brackets;" and "HI-STORM Lift Links," the HI-STORM canister transfer equipment includes various lifting devices. The HI-TRAC lift links are attached to the HI-TRAC 125 Transfer Cask lifting trunnions and are used to raise and lower the transfer cask in a single-failure-proof mode. The MPC downloader slings are attached to the MPC and are used to raise and lower the MPC between the HI-TRAC 125 Transfer Cask and the HI-STORM 100SA Overpack in a single-failure-proof mode. The function of the MPC lift cleats is to provide a temporary means to lift the MPC. The function of the HI-STORM 100SA Overpack lifting brackets and lift links is to provide a means of lifting the storage cask. The transporter connection pins, as described in the SAR, Table 4.3-1, connect the transfer cask lift links or the overpack lifting brackets to the cask transporter lift links. The transfer cask horizontal lift rig transmits the load of the lifted transfer cask from the transfer cask lift slings to the cask transporter lift points. The transfer cask lift slings are used to support the weight of the loaded transfer cask and the cask transport frame during horizontal lifting by the cask transporter. Detailed design descriptions of the associated lifting devices are given in the HI-STORM 100 System FSAR (Holtec International, 2002).

Components requiring inspection and maintenance are identified, and operational procedures are summarized. Preoperational, startup, and operational tests will be performed to verify the functional operations of SSCs important to safety. This design also allows for emergency load carrying capability. Design of the associated lifting devices allows for control of loads in the event of emergencies. The associated lifting devices have been sufficiently described in accordance with 10 CFR §72.24(b), §72.122(f), and §72.122(g).

HI-STORM Mating Device Bolts and Shielding Frame (QA Category A) and HI-STORM Cask Mating Devices (QA Category C)

As identified in SAR Section 4.2.3.2.4, "HI-TRAC 125 Transfer Cask," the cask mating device replaces the transfer lid on the HI-TRAC 125 Transfer Cask. The cask mating device bolts and shielding frame transfer the load of the MPC to the transfer cask and are part of the single-failure-proof load path. The remainder of the cask mating device is not part of the single-failure-proof load path and are considered QA Category C. A drawing of the mating device is provided in SAR Figure 4.2-11. Detailed design descriptions of the mating device are provided in the HI-STORM 100 System FSAR (Holtec International, 2002).

The description of the entire cask mating device includes consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). Components requiring inspection and maintenance are identified, and

operational procedures are summarized. Preoperational, startup, and operational tests will be performed to verify the functional operations of SSCs important to safety. This design also allows for emergency load carrying capability. Design of the cask mating device allows for control of loads in the event of emergencies. The cask mating device has been sufficiently described in accordance with 10 CFR §72.24(b), §72.122(f), and §72.122(g).

Cask Transporter (QA Category A)

SAR Section 4.3.2.1.1, "Description," identifies the cask transporter as a self-propelled tracked vehicle used to move the transfer cask with loaded MPC from the FHB/AB to the CTF and the loaded storage cask between the CTF and the storage pad. The cask transporter, shown in SAR Figures 4.3-1 to 4.3-3 and Figures RAI 5-11-1 to 5-1-5, is a custom-designed, commercial-grade system that will be qualified upon receipt. The transporter was designed as a mobile single-failure-proof system in accordance with NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980) requirements. The transporter also was designed to preclude tip-over during a design-basis earthquake or impact by a design-basis tornado missile.

The description of the cast transporter includes consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). Components requiring inspection and maintenance are identified, and operational procedures are summarized. Preoperational, startup, and operational tests will be performed to verify the functional operations of SSCs important to safety. This design also allows for emergency load carrying capability. Design of the cask transporter allows for control of loads in the event of emergencies. The cask transporter has been sufficiently described in accordance with 10 CFR §72.24(b), §72.122(f), and §72.122(g).

Lateral Restraints (QA Category A)

As identified in SAR Section 4.2.1.2, "CTF Support Structure," four lateral ground restraints provide ground-level attachment points for restraint of the cask transporter during canister transfer operations. SAR Figure 4.2-4 shows the general layout of the lateral restraints. Details of the position of the lateral restraints and maximum load at each of the restraint locations were determined for design purposes (Pacific Gas and Electric Company, 2001b). The restraints ensure that the MPC will remain stable and will not topple in the event of a design-basis event during the transfer operation. Although no design details are provided for the lateral restraints or the attachment points, the design basis and codes and standards are identified (Pacific Gas and Electric Company, 2002d). The staff determined that the SAR adequately describes the configuration of the lateral restraints as required in 10 CFR §72.24(b).

HI-STORM 100SA Overpack (QA Category B)

As identified in the SAR Section 4.2.3, "Storage Cask Description," the storage cask is a steel and concrete cylindrical structure that serves as a missile barrier and radiation shield and provides flow paths for natural convective heat transfer and stability for the system. The storage cask is designed to meet ASME Boiler and Pressure Vessel Code, Section III, Subsection NF requirements (ASME International, 1995a). The unreinforced concrete infill material is not considered a structural material. Table 4.2-2 and Figure 4.2-7 of the SAR provide a summary of

the physical characteristics of the storage cask. A complete design description of the storage cask system is provided in the HI-STORM 100 System FSAR (Holtec International, 2002).

The staff reviewed SAR Section 4.2.3, "Storage Cask Description," with respect to the description of the storage cask. These descriptions include consideration of inspection, maintenance, and testing. Components requiring inspection and maintenance were identified and operational procedures summarized. Inspection was limited to checks of the air vents to ensure that they are not blocked and assessment of the condition of the anchorage hardware. This design also allows for emergency access. Spacing of the storage casks on the reinforced concrete pads allows for access to critical locations and regions in the event of emergencies.

Additionally, the staff review of SAR Section 4.2.3, "Storage Cask Description," determined that the design features of the storage cask related to shielding and heat removal capability were appropriately described. A comprehensive shielding evaluation is contained in Chapter 7 of this SER. The design of the storage cask places the spent nuclear fuel in a sealed canister to limit the amount of radioactive waste generated at an ISFSI. A comprehensive waste confinement and management evaluation is contained in Chapter 14 of this SER. The HI-STORM 100SA Overpack has been sufficiently described in accordance with 10 CFR §72.24(b), §72.122(f), and §72.122(g)

Cask Transfer Facility and Lift Jacks (QA Category B)

As identified in SAR Section 4.2.1.2, "CTF Support Structure," the CTF provides physical protection and shielding of the canisters during transfer from the transportation cask to the storage cask. SAR Figures 4.2-4 and 4.4-3 illustrate the layout of the CTF. The CTF is a cylindrical steel-lined structure embedded in rock and made of reinforced concrete slabs and walls. The reinforced concrete structure portion of the structure is designed in accordance with ACI 349-97 (American Concrete Institute, 1998). The ACI 349-97 code provides the minimum requirements for the design and construction of nuclear safety-related concrete structures and structural elements for nuclear power generating stations. The structural steel elements are designed in accordance with ASME Code, Section III Subsection NF (ASME International, 1995a). Components requiring inspection, testing, and maintenance are identified and adequately described in accordance with 10 CFR §72.122(f). Preoperational, startup, and operational tests will be performed to verify the functional operations of SSCs important to safety. The design in accordance with ACI 349-97 and ASME Section III Subsection NF addresses these topics. The design of the CTF allows access to all locations and regions in the event of emergencies in accordance with the requirements of 10 CFR §72.122(g).

The SAR provides a design description of the CTF in sufficient detail to support a detailed review and evaluation. Consequently, the requirements of 10 CFR §72.24(a) and §72.24(b) have been satisfied. The staff, therefore, conclude that the design of the CTF complies with 10 CFR §72.24(c)(4).

Transfer Cask Impact Limiters (QA Category A) and Helium Fill Gas (QA Category B)

These SSCs are only associated with the 10 CFR Part 50 operations and are not considered in this SER.

5.1.4.2 Design Criteria for Other Structures, Systems, and Components Important to Safety

The design bases for the other SSCs important to safety are given in the SAR. Table 4.2-5 of the SAR identifies details of the Diablo Canyon ISFSI compliance with the general design criteria of 10 CFR Part 72, Subpart F. As identified in the SAR, the other SSCs important to safety are designed in accordance with the design criteria contained in Chapter 3 of the SAR. This conclusion is supported by the structural analysis performed as described in Section 5.1.4.4 of this SER. Design criteria have been shown in Chapter 4 of this SER to be representative of the site. A complete discussion of the design criteria for the transfer cask and associated lifting devices is given in Section 4.1.3, "Design Criteria for Structures, Systems, and Components Important to Safety," of this SER. As identified in Chapter 4 of this SER, the site-specific criteria are enveloped by the design criteria identified in the HI-STORM 100 System FSAR (Holtec International, 2002). The design criteria establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for SSCs important to safety. The design criteria address the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events. The design criteria include the effects of natural phenomena and cover credible fires and explosions.

Fuel Basket, Damaged Fuel Container, Upper and Lower Fuel Spacer Columns, and End Plates

The design criteria for the MPC internals are presented in the HI-STORM 100 System FSAR (Holtec International, 2002) and evaluated in the related staff SER (U.S. Nuclear Regulatory Commission, 2002b). The design criteria are summarized in SAR Table 3.4-2. The MPC internals are designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME International, 1995a). Fabrication of the MPC internals is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Article NG–4000 (ASME International, 1995a). Inspection of MPC internals are in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NG–5000 (ASME International, 1995a), and Section V (ASME International, 1995b). Exceptions to the ASME and Pressure Vessel Code are provided in the CoC 1014-1 Appendix B Table 3-1 (U.S. Nuclear Regulatory Commission, 2002a) and the Diablo Canyon ISFSI SAR Table 3.4-6 (Pacific Gas and Electric Company, 2002a). The lifting bolts of the damaged fuel container are designed in accordance with requirements for a single-failure-proof system identified in ANSI N14.6 (American National Standard Institute/American Nuclear Society, 1993) per the applicable guidelines of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980, Section 5.1.6).

The staff conclude that the design criteria of the MPC internals meet the requirements of the ASME Code, as applicable. The conclusions drawn in this section for other structures, systems, and components important to safety design criteria are based on the evaluation findings made in Section 4.1.3 of this SER. Details are contained within the HI-STORM 100 System FSAR (Holtec International, 2002), which the staff previously reviewed and accepted. The MPC internals design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c), §72.120(a), and §72.236(b).

HI-TRAC 125 Transfer Cask

The structural steel elements of the HI-TRAC are designed in accordance with ASME Code, Section III, Subsection NF (ASME International, 1995a). The transfer cask is designed in

accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NF Article NF–3300 (ASME International, 1995a). Material procurement is in accordance with ASME Boiler and Pressure Vessel Code, Section II (ASME International, 1995d,f) and Section III, Subsection NF, Article NF–2000 (ASME International, 1995a). Metal components of the transfer cask are fabricated and inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Class 3 (ASME International, 1995b) or AWS D1.1 (American Welding Society, 2002).

Welding of the transfer cask structure will be performed using welders and weld procedures that have been qualified in accordance with the ASME and Pressure Vessel Code, Section IX (ASME International, 1995c) and Section III, Subsection NF (ASME International, 1995a). For nonNF welds, the AWS D1.1 code (American Welding Society, 2002) will be used for welders and weld procedures that have been qualified in accordance with the AWS requirements or in accordance with ASME Boiler and Pressure Vessel Code, Section IX (ASME International, 1995c). All welds require visual examination in accordance with the ASME Boiler and Pressure Vessel Code, Section V, Article 9 (ASME International, 1995b), with acceptance criteria specified in Section III, Subsection NF, Article NF–5360 (ASME International, 1995a). As specified in engineered drawings 1880 and 2145 in the HI-STORM 100 System FSAR Section 1.5 (Holtec International, 2002), the transfer cask structural welds of the outer shell, enclosure shell, and radial ribs will be inspected using liquid penetrant testing in accordance with the ASME Boiler and Pressure Vessel Code, Section V, Article 6 (ASME International, 1995b) with acceptance criteria specified in Section III, Subsection NF, Article NF–5350 (ASME International, 1995a), or magnetic particle testing in accordance with ASME Boiler and Pressure Vessel Code, Section V, Article 7 (ASME International, 1995b), with acceptance criteria specified in Section III, Subsection NF, Article NF–5340 (ASME International, 1995a). Transfer cask inspection is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Article NF–5360 (ASME International, 1995a) and Section V (ASME International, 1995b). Exceptions to the ASME Boiler and Pressure Vessel Code are provided in the CoC 1014-1, Appendix B, Table 3-1 (U.S. Nuclear Regulatory Commission, 2002a) and the Diablo Canyon ISFSI SAR Table 3.4-6 (Pacific Gas and Electric Company, 2002a).

The HI-TRAC 125 Transfer Cask is also designed as a special lifting device in accordance with ANSI N14.6–1993 (American National Standard Institute/American Nuclear Society, 1993) and NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). ANSI N14.6–1993 sets forth the requirements for design, fabrication, testing, maintenance, and QA programs for special lifting devices used to handle containers with radioactive materials.

Details of the design criteria are contained within the HI-STORM 100 System FSAR (Holtec International, 2002), which the staff previously reviewed and accepted. The HI-TRAC 125 Transfer Cask design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c), §72.120(a), and §72.236(b).

<u>Transfer Cask Lift Links, MPC Downloader Slings, MPC Lift Cleats, HI-STORM Lifting Brackets, HI-STORM Lift Links Transporter Connection Pins, Transfer Cask Horizontal Lift Rig, and Transfer Cask Lift Slings</u>

The lifts slings are designed in accordance with ASME B30.9 (ASME International, 2000) according to the guidance of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). All

the other components include consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). Specifics of the design bases for these components are given in SAR Section 4.2.3.3, "Design Bases and Safety Assurance." A complete discussion of the design criteria for the transfer cask and associated lifting devices is provided in Section 4.1.3, "Design Criteria for Structures, Systems, and Components Important to Safety" of this SER.

Details are contained within the HI-STORM 100 System FSAR, which the staff previously reviewed and accepted. The associated lift hardware design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c) and §72.120(a).

HI-STORM Mating Device, Mating Device Bolts, Mating Device Shielding Frame

Explicit design criteria for the HI-STORM 100 System mating device are not contained within the SAR. Details are contained within the HI-STORM 100 System FSAR, which the staff previously reviewed and accepted.

Cask Transporter

The transporter is custom designed in accordance with the single-failure-proof requirements of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). The functional specification for the cask transporter is identified in Holtec International (2001i). The description of the cast transporter include consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG–0612.

The cask transporter design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c) and §72.120(a).

Lateral Restraints

Explicit design criteria for the lateral restraints are based on the seismic induced loads as identified in RAI response 3-3 (Pacific Gas and Electric Company, 2002d). The final design of these elements has not been completed but will be designed to meet the stress limits of ASME Section III, Subsection NF (ASME International, 1995a). The attachment points will be designed in accordance with ACI 349-97 (American Concrete Institute, 1998).

HI-STORM 100SA Overpack

Design criteria for the cask systems are contained in the HI-STORM 100 System FSAR (Holtec International, 2002), which has been previously reviewed and accepted by the staff. The generic design base loadings specified in the HI-STORM 100 System FSAR for the anchored cask system envelop the Diablo Canyon ISFSI site parameters. A discussion of the design criteria for the storage cask is provided in Section 4.1.3, "Design Criteria for Structures, Systems, and Components Important to Safety" of this SER.

Applicable ASME Boiler and Pressure Vessel Code criteria for the overpack steel structure are provided in HI-STORM 100 System FSAR Table 2.2.7 (Holtec International, 2002). Material

procurement for the overpack steel structure is in accordance with ASME Boiler and Pressure Vessel Code, Section II (ASME International, 1995d,e,f) and Section III, Subsection NF, Article NF–2000 (ASME International, 1995a). The overpack steel structure is designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Article NF–3200 (ASME International, 1995a). Metal components of the overpack are fabricated and inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Class 3 (ASME International, 1995a) or AWS D1.1 (American Welding Society, 2002).

Welding of the overpack structure will be performed using welders and weld procedures that have been qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section IX (ASME International, 1995c) and Section III, Subsection NF (ASME International, 1995a). For nonNF welds, the AWS Code will be used for welders and weld procedures that have been qualified in accordance with the AWS D1.1 (American Welding Society, 2002) requirements or in accordance with ASME Boiler and Pressure Vessel Code, Section IX (ASME International, 1995a). All welds require visual examination in accordance with the ASME Boiler and Pressure Vessel Code, Section V, Article 9 (ASME International, 1995b), with acceptance criteria specified in Section III, Subsection NF, Article NF–5360 (ASME International, 1995a). As specified in engineered drawing 1495 in the HI-STORM 100 System FSAR Section 1.5 (Holtec International, 2002), the overpack inner shell seam weld will be inspected using penetrant testing in accordance with the ASME Boiler and Pressure Vessel Code, Section V, Article 6 (ASME International, 1995b) with acceptance criteria specified in Section III, Subsection NF, Article NF–5350 (ASME International, 1995a) or magnetic particle testing in accordance with ASME Boiler and Pressure Vessel Code, Section V, Article 7 (ASME International, 1995e), with acceptance criteria specified in Section III, Subsection NF, Article NF–5340 (ASME International, 1995a). Exceptions to the ASME Boiler and Pressure Vessel Code are provided in the CoC 1014-1, Appendix B Table 3-1 (U.S. Nuclear Regulatory Commission, 2002a) and the Diablo Canyon ISFSI SAR, Table 3.4-6 (Pacific Gas and Electric Company, 2002a). Overpack steel structure inspection is in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Articles NF–5350 and NF–5360 (ASME International, 1995a), and Section V (ASME International, 1995b). Material procurement for the overpack anchor studs is in accordance with ASME Boiler and Pressure Vessel Code, Section II (ASME International, 1995d,f) and Section III, Subsection NF, Article NF–2000 (ASME International, 1995a) and the anchor studs are designed in accordance with Subsection NF, Article NF–3300 (ASME International, 1995a).

The unreinforced concrete elements of the HI-STORM 100SA Overpack are designed in accordance with ACI 349-85 (American Concrete Institute, 1985).

The HI-STORM 100SA Overpack design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c), §72.120(a), and §72.236(b).

Cask Transfer Facility and Lift Jacks

The design criteria for the CTF establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for this structure important to safety. A summary of the design criteria for the CTF is provided in SAR Table 3.4-5. Additional information is provided in the site-specific design criteria document (Holtec International, 2001d). The structural steel elements of the CTF are designed in accordance with ASME Code, Section III,

Subsection NF (ASME International, 1995a). CTF concrete elements are designed in accordance with ultimate strength design methods specified in ACI 349-97 (American Concrete Institute, 1998). A complete discussion of the design criteria applicable to the CTF is provided in Section 4.1.3 of this SER.

The conclusions in this section regarding the CTF design criteria are based on the evaluation findings in Section 4.1.3 of this SER. The CTF design criteria and relevant codes and standards have been identified in accordance with 10 CFR §72.24(c) and §72.120(a).

5.1.4.3 Material Properties for Other Structures, Systems, and Components Important to Safety

The staff reviewed the material properties for other SSCs important to safety with respect to the applicable regulatory requirements.

Fuel Basket, Damaged Fuel Container, and Upper and Lower Fuel Spacer Columns and End Plates

The applicant provided a description of the HI-STORM 100 System including materials of construction, fabrication details, and testing in SAR Sections 4.2, "Storage System;" 4.7, "Operating Environment Evaluation;" and Appendix A, "Materials" (Pacific Gas and Electric Company, 2002a). The structural components of the fuel basket and DFC are identical to those used for the MPC. Engineering drawings and additional details of the storage system are included by reference to the Holtec HI-STORM 100 System FSAR Sections 3.3 and 3.4 (Holtec International, 2002). Technical specifications for the HI-STORM 100 System are included by reference to 10 CFR Part 72 CoC No. 1014 (U.S. Nuclear Regulatory Commission, 2002a) for the HI-STORM 100 System and License Amendment Request 1014-1 (U.S. Nuclear Regulatory Commission, 2002a). The HI-STORM 100 System has been evaluated by the staff and approved for use for dry storage of spent nuclear fuel (U.S. Nuclear Regulatory Commission, 2002b).

Mechanical properties of the structural materials for the fuel basket and DFC are provided in the Holtec HI-STORM 100 System FSAR Section 3.3 and Tables 3.3.1, 1.A.1, 3.3.2, 3.3.3, and 3.3.4 (Holtec International, 2002) for the major structural materials including stainless steels, carbon steel, low-alloy steels, nickel-alloy steel and nickel-base alloy. The values in these tables were obtained from ASME Code, Section II, Part D (ASME International, 1995f). The staff independently verified the temperature-dependent values for the stress allowables, ultimate strength, yield strength, modulus of elasticity, and coefficient of thermal expansion. The staff conclude that these material properties are acceptable and appropriate for the expected load conditions during the license period.

The materials of construction for the fuel basket and DFC are readily weldable using commonly available welding techniques. The fuel basket and DFC are constructed from stainless steel materials as specified in the HI-STORM 100 System FSAR Tables 3.3.1 and 1.A.1 (Holtec International, 2002). The fuel basket assemblies for the MPCs are shown in engineered drawings 3925 through 3928 in the HI-STORM 100 System FSAR Section 1.5 (Holtec International, 2002). The drawings include standard welding symbols and notations in accordance with AWS Standard A2.4 (American Welding Society, 1998). The MPC baskets and basket supports are fabricated and inspected in accordance with the ASME Boiler and

Pressure Vessel Code, Section III, Subsection NG, and Section V (ASME International, 1995a,b). MPC welding will be performed using welders and weld procedures that have been qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section IX (ASME International, 1995c) and Section III, Subsections NB, and NG (ASME International, 1995a).

Criticality control in the MPC is achieved using a fuel basket structure of edge-welded composite boxes and Boral neutron poison panels (Holtec International, 2002). Boral has a long, proven history in the nuclear industry and has been used in other spent nuclear fuel storage casks. The Boral sheets are enclosed within the welded stainless steel cladding to minimize degradation as a result of environmental exposure.

Chemical reaction between the Boral and the borated water in pressurized water reactor spent nuclear fuel pools may produce small amounts of hydrogen gas during loading and unloading operations (Pacific Gas and Electric Company, 2002a). The safety hazards associated with ignition of this hydrogen gas are mitigated by monitoring for combustible gas concentrations prior to, and during, MPC lid welding operations. The space below the MPC lid will be exhausted or purged with inert gas prior to and during MPC lid welding operations to provide additional assurance that explosive gas mixtures will not develop in this space as indicated in the SAR Section 5.1.1.2, "MPC Loading and Sealing Operations," and Table 4.7-1 (Pacific Gas and Electric Company, 2002a). In addition, the Boral will be passivated before installation in the fuel basket to minimize the amount of hydrogen released from the aluminum-water reaction to a noncombustible concentration during MPC lid welding or cutting operations. The staff conclude that the guidance in the generic procedures is adequate to prevent ignition of any hydrogen gas that may be generated during loading and MPC closure welding operations.

The staff conclude that the selection of these materials is acceptable for the DFC. The staff conclude that the material selection for the MPC internals meets the requirements of the ASME Codes, as applicable. The MPC internal materials have been identified in accordance with 10 CFR §72.24(c)(3), §72.120(a), §72.122, and §72.236(b).

HI-TRAC 125 Transfer Cask

The transfer cask is constructed from carbon steel, low-alloy steel, nickel-alloy steel, stainless steel, and nickel-base alloy as specified in the HI-STORM 100 System FSAR Table 2.2.6 (Holtec International, 2002). The inner shell, radial channels, enclosure shell panels, water jacket end plate, lower water jacket shell, top lid plates, pool lid plates, lid tongues, fill port caps, and top lid inner and outer rings are constructed from SA516 Grade 70 carbon steel. The top flange is constructed from SA350-LF3 low-alloy steel or SA203E nickel-alloy steel, and the bottom flange is constructed from SA350-LF3 low-alloy steel or SA516 Grade 70 carbon steel. The pool lid outer ring is constructed from either SA516 Grade 70 carbon steel, SA350-LF3 low-alloy steel, or SA203E nickel-alloy steel. Top lid studs are constructed from SA193-B7 low-alloy steel, and the top lid nuts are constructed from SA194-2H carbon steel. The pocket trunnion and the lifting trunnion block are constructed from SA350-LF3 low-alloy steel. The dowel pins are constructed from SA564-630 precipitation hardened stainless steel. Mechanical properties of the transfer cask structure are provided in the HI-STORM 100 System FSAR Tables 3.3.2, 3.3.3, and 3.3.4 (Holtec International, 2002). Applicable ASME Boiler and Pressure Vessel Code (ASME International 1995f) criteria for the

transfer cask structure are provided in HI-STORM 100 System FSAR Table 2.2.7 (Holtec International, 2002).

The transfer cask fabrication welds are characterized in engineered drawings 1880, 1928, 2145, and 2152 in the HI-STORM 100 System FSAR Section 1.5 (Holtec International, 2002). The drawings include standard welding symbols and notations in accordance with AWS Standard A2.4 (American Welding Society, 1998).

Structural steel components of the transfer cask are subject to brittle fracture at low temperatures. The lowest service temperature for the structural components of the transfer cask is specified as -18 °C [0 °F], which is above the ductile-to-brittle transition temperature for the structural steel components. A lowest service temperature of -18 °C [0 °F] is specified for all parts used to lift the transfer cask, which is above the ductile-to-brittle transition temperature for the pocket trunnions, lifting trunnions, and the lifting trunnion block.

The HI-TRAC 125 Transfer Cask uses lead encased between steel plates and water to provide gamma and neutron shielding in the radial direction. Layers of steel-lead-steel in the transfer cask lids provide shielding in the axial direction. The neutron shield material, Holtite-A, is used in the HI-TRAC 125 Transfer Cask lid. The HI-TRAC 125 Transfer Cask top lid also contain Holtite-A to provide gamma attenuation. Holtite-A is a high-hydrogen content, durable, fire-resistant material. A detailed description of Holtite-A is provided in the HI-STORM 100 System FSAR Appendix 1.B (Holtec International, 2002).

Possible chemical, galvanic, and other reactions among the materials in the range of possible exposure environments are evaluated in SAR Section 4.7, "Operating Environment Evaluation," and Table 4.7-1 (Pacific Gas and Electric Company, 2002a). Steels used in the transfer cask are coated to prevent corrosion during exposure to water during loading operations. Threaded portions of the transfer cask that are not coated will be plugged or covered to prevent corrosion while immersed during loading operations. No adverse reactions were identified for coatings used on the transfer cask, or the elastomer seal used on the transfer cask during loading operations.

Material properties for the HI-TRAC 125 Transfer Cask are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER. The transfer cask materials have been identified in accordance with 10 CFR §72.24(c)(3).

<u>Transfer Cask Lift Links, MPC Downloader Slings, MPC Lift Cleats, HI-STORM Lifting Brackets, HI-STORM Lift Links Transporter Connection Pins, Transfer Cask Horizontal Lift Rig, and Transfer Cask Lift Slings</u>

Materials for the associated lifting devices are not explicitly identified in the SAR. As identified, they will be designed and fabricated in accordance with the applicable codes and standards. These standards identify the acceptable material characteristics. Additional details of the material properties for the associated lifting devices are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER. The associated lifting devices materials have been identified in accordance with 10 CFR §72.24(c)(3).

HI-STORM Cask Mating Device and HI-STORM Mating Device Bolts and Shielding Frame

Materials for the cask mating device, including bolts and shielding frame, are not explicitly identified in the SAR. As identified, it will be designed and fabricated in accordance with the applicable codes and standards. These standards identify the acceptable material characteristics. Additional details of the material properties for the cask mating device, including bolts and shielding frame, are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER. The cask mating device, including bolts and shielding frame, materials have been identified in accordance with 10 CFR §72.24(c)(3).

Cask Transporter

Materials for the cask transporter are not explicitly identified in the SAR. This is to be a custom-designed system that will be designed and fabricated in accordance with the applicable codes and standards. These standards identify the acceptable material characteristics. The staff conclude that use of the applicable codes and standards for the materials of construction is in accordance with 10 CFR §72.24(c)(3).

Lateral Restraints

Materials for the lateral restraints are not explicitly identified in the SAR. This is to be a custom-designed system that will be designed and fabricated in accordance with the applicable codes and standards (ASME International, 1995a; American Concrete Institute, 1998). These standards identify the acceptable material characteristics. The staff conclude that, if the applicable codes and standards are followed, the proper materials for construction will be selected in accordance with 10 CFR §72.24(c)(3).

HI-STORM 100SA Overpack

The overpack is constructed using carbon steel, low-alloy steel, nickel-alloy steel, and stainless steel as specified in the HI-STORM 100 System FSAR Table 2.2.6 (Holtec International, 2002). The inner and outer cylindrical shells, base plate, and lid are constructed from SA516 Grade 70 carbon steel. Lid studs are constructed from SA564-630 precipitation hardened stainless steel, and the nuts are SA194-2H carbon steel. The bolt anchor blocks are constructed from SA350-LF3 low-alloy steel or SA203E nickel-alloy steel. Material properties for the overpack are provided in the HI-STORM 100 System FSAR, Table 2.2.6 (Holtec International, 2002). Mechanical properties of the overpack structural materials are provided in the HI-STORM 100 System FSAR Revision 1, Tables 3.3.2, 3.3.3, and 3.3.4 (Holtec International, 2002). Structural steel components of the disposal overpack are subject to brittle fracture at low temperatures. The lowest service temperature for the structural components of the overpack is specified as -40 °C [-40 °F], which is above the ductile-to-brittle transition temperature for the structural steel components. A lowest service temperature of -18 °C [0 °F] is specified for all parts used to lift or anchor the overpack. The 0.76-m [30-in] annular space between the inner and outer shells is filled with unreinforced concrete for radiation shielding. Concrete is also used as a shielding material for the bolted lid. Material procurement for the concrete used for shielding meets the requirements of ACI 349, and the design of the overpack concrete meets the requirements of ACI 349-97 (American Concrete Institute, 1998). The staff conclude that the materials used to construct the disposal overpack are suitable for structural support, shielding,

and protection of the MPC from environmental conditions. The staff conclude that the welded joints of the overpack meet the requirements of the ASME International (1995a) and American Welding Society (2002) Codes, as applicable.

All steel surfaces of the overpack are coated with either Thermaline 450, Carbozinc 11, or Carbozinc 11HS as indicated by the Holtec HI-STORM 100 System FSAR, Revision 1, Table 2.2.6 (Holtec International, 2002) and the response to RAI 18-8 (Pacific Gas and Electric Company, 2002c). The coatings are used to protect the steel overpack from oxidation and corrosion. External surface coatings will be maintained as indicated in Diablo Canyon ISFSI SAR Table 4.7-1. During review of the Diablo Canyon ISFSI SAR, staff identified the maximum normal operating temperature of the overpack may be 63 °C [145 °F], exceeding the maximum application temperature for Thermoline 450 {43 °C [110 °F]}. In response to RAI 18-8, the applicant indicated that the actual exterior surface temperatures are expected to be less than the calculated bounding value (Pacific Gas and Electric Company, 2002d). Repair of coatings on the exterior surfaces of the overpack will be made in accordance with the manufacturer's recommendations. Overpack surface temperatures will be checked with a pyrometer prior to coating repair. If necessary, the external surface will be cooled to below the maximum coating application temperature. The specified maximum surface temperature for application of Carbozinc 11 or Carbozinc 11HS is above the maximum normal operating temperature of the overpack (Pacific Gas and Electric Company, 2002d).

Exterior steel surfaces of the transfer cask are coated with Carboguard 890, and interior steel surfaces are coated with Thermaline 450 as indicated by the Holtec HI-STORM 100 System FSAR, Revision 1, Table 2.2.6 (Holtec International, 2002) and the response to RAI 18-8 (Pacific Gas and Electric Company, 2002d). The coatings are used to protect the steel surfaces from oxidation and corrosion. Thermaline 450 is used on the interior for its higher temperature ratings. Carboguard 890 is used on the exterior because of its decontamination characteristics. The coatings are not expected to be affected by short-term exposure to borated spent nuclear fuel pool water or the gamma radiation dose and neutron fluence. The transfer cask will be fully decontaminated, inspected, and recoated as necessary prior to its next use. Small nicks in the coatings are not expected to affect either the coatings or any of the exposed carbon steel.

The coatings are resistant to chemical attack in a variety of chemical environments, resistant to abrasion, and resistant to permeation. The coatings have been successfully used in other licensed nuclear power plants. The staff conclude that the applications of the epoxy paint to the exposed surfaces of the transfer cask and the epoxy paint and zinc coating applied to the carbon steel components of the overpack are acceptable.

The storage overpack uses concrete and steel for shielding in the radial direction and a thick circular concrete slab attached to the lid and a thick circular concrete pedestal to provide gamma and neutron attenuation in the axial direction. Additional steel plates and shell elements are used to provide additional gamma shielding in specific areas.

Material properties for the HI-STORM 100SA Overpack are provided in the HI-STORM 100 System FSAR. The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER. The overpack materials have been identified in accordance with 10 CFR §72.24(c)(3).

Cask Transfer Facility and Lift Jacks

Materials for the steel components and lift jacks used in the CTF and lift jacks are identified in SAR Figure 4.4-3. Additional details of the material properties are given in the CTF analysis document (Holtec International, 2001h). Structural steel elements will be constructed of SA516, Grade 70, SA36, and WELDOX 130 steel. Material properties for the lift jack screws are identified in Holtec, International (2001h). Concrete material selection is based on concrete with a compressive strength of 20.7 MPa [3,000 psi] at 28 days following Chapter 5 of ACI 349-97 (American Concrete Institute, 1998) as identified in the CTF analysis document (ENERCON Services, Inc,. 2001a). The reinforcing steel is specified to have a minimum yield strength of 414 MPa [60,000 psi] following Chapter 3 of ACI 349-97 (American Concrete Institute, 1998) and American Society for Testing and Materials (1990).

Based on the review of information presented by PG&E, the staff conclude that materials to be used to construct the CTF and lift jacks have been adequately identified in accordance with 10 CFR §72.24 (c)(3). The applicant identified the appropriate codes and standards in accordance with 10 CFR §72.24(c)(4).

5.1.4.4 Structural Analysis for Other Structures, Systems, and Components Important to Safety

The staff reviewed the SAR and found that the structural analysis procedures have been identified and are in conformance with standard engineering practice. Other SSCs important to safety were designed and analyzed to resist the loads and loading combinations specified in the design criteria. The analyses of other SSCs important to safety included loading conditions of dead and live loads, thermal loads, earthquake, tornado, wind, or tornado missiles, and fire, as applicable. The staff reviewed the structural analysis for other SSCs important to safety with respect to the regulatory requirements of 10 CFR §72.24 and §72.122.

Fuel Basket, Damaged Fuel Container, and Upper and Lower Fuel Spacer Columns and End Plates

Structural analyses of the fuel basket, DFC and upper and lower fuel spacer columns and end plates for the HI-STORM 100 System are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). No additional review was performed for this SER.

HI-TRAC 125 Transfer Cask

Structural analysis of the HI-TRAC 125 Transfer Cask for the HI-STORM 100 System is provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in Section 3.5.5 of the NRC HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). The discussion below is based, in part, on the results presented in the HI-STORM 100 System FSAR and SER and summarized in the Diablo Canyon ISFSI SAR.

The HI-TRAC 125 Transfer Cask, with the transfer lid attached, is designed to meet ASME Section III Subsection NF (ASME International, 1995a) stress limits for all loading conditions.

The structural analysis for the HI-TRAC 125 Transfer Cask in the HI-STORM 100 System FSAR demonstrates that the transfer cask is designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions.

Fire loading conditions of the HI-TRAC 125 Transfer Cask are addressed in Section 11.2.4 of the HI-STORM 100 System FSAR and in Section 8.2.5, "Fire," of the Diablo Canyon ISFSI SAR. As shown in Section 11.2.4 of the HI-STORM 100 System FSAR, fires near a loaded transfer cask would not produce any structural degradation, and only a small amount of neutron shielding material is lost or damaged. The FSAR indicates that fuel cladding, MPC, and transfer cask temperatures would remain below the design temperature limits.

A site-specific analysis was performed to assess the vulnerability of the transfer cask to site-specific tornado missiles (Holtec International, 2001e). The analysis demonstrated that the HI-TRAC 125 Transfer Cask provides effective missile barriers for the MPC. No missile strike compromises the integrity of the confinement boundary. In addition, global stress intensities arising from the missile satisfy ASME Code Level D limits for an ASME Section III, Subsection NF structure (ASME International, 1995a).

Site-specific evaluations of the risk that explosions could damage the HI-TRAC 125 Transfer Cask were performed (Holtec International, 2001b; Pacific Gas and Electric Company, 2002c). All evaluations concluded that these hazards were not credible.

The structural analysis performed by PG&E demonstrates that the HI-TRAC 125 Transfer Cask is adequately designed to resist the loads based on the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events in accordance with the requirements of 10 CFR §72.122(b)(1). Structural analysis carried out by PG&E demonstrates that the HI-TRAC 125 Transfer Cask is designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, lightning, and floods, without impairing the capacity to perform safety functions in accordance with the requirements of 10 CFR §72.122(b)(2).

<u>Transfer Cask Lift Links, MPC Downloader Slings, MPC Lift Cleats, HI-STORM Lifting Brackets, HI-STORM Lift Links Transporter Connection Pins, Transfer Cask Horizontal Lift Rig, and Transfer Cask Lift Slings</u>

Structural analysis of the associated lifting hardware is provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). No additional review was performed for this SER.

The lift links, slings, and rigs are designed as nonredundant lifting devices with a factor of safety of 10 or greater for material ultimate strength and 6 or greater for yield strength. A dynamic load increase factor of 10 percent has been applied to the lifting loads. Therefore, these elements meet the NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980) stress limits for nonredundant special lifting devices.

The lift cleats and brackets and connector pins are designed with a minimum factor of safety of 3 for material yield strength and 5 for material ultimate strength, as well as a dynamic load

increase factor of 10 percent. Multiple elements are used, and each can totally support the weight of the canister, thereby making them single-failure proof in accordance with NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980).

HI-STORM Cask Mating Device and HI-STORM Mating Device Bolts and Shielding Frame

Structural analyses of the cask mating device, including bolts and shielding frame, for the HI-STORM 100 System are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). No additional review was performed for this SER.

Cask Transporter

The cask transporter is custom designed for the site-specific criteria (Holtec International, 2001i) in accordance with NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980). Structural analysis to be completed by the applicant in accordance with the requirements on NUREG–0612 will demonstrate that the cask transporter is designed to resist the loads based on the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events, in accordance with the requirements of 10 CFR §72.122(b)(1). The structural analysis will also demonstrate that the cask transporter is designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions in accordance with the requirements of 10 CFR §72.122(b)(2).

The applicant provided site-specific analyses of the stability of the cask transporter (Holtec International, 2001f; ENOVA Engineering Services, 2002). In each of these analyses, the cask transporter is assumed to be rigid within the range of seismic excitation. Nonlinear analysis was performed to determine the extent of motion of the cask transporter in the direction of travel and perpendicular to the direction of travel. The Holtec analysis identified the response because of the site-specific design events using VisualNastran (MSC Software Corporation, 2001). To account for potential amplification of the seismic event because of soils, the ENOVA analysis considered a seismic event twice the design-basis event. Both sets of analysis demonstrated that the cask transporter would not tip-over or slide off the road. Both the analyses assume that the road surface remains stable. The staff reviewed this assumption in SER Section 15.1.2.6, based on the joint probability of the annual exposure probability for transport casks and the annual exceedance probability for the earthquake ground motion. Based on the review, the staff concluded that earthquake induced damage to the casks, while in transit from the power plant to the CTF, is not a credible hazard to the proposed facility.

Lateral Restraints

The applicant identified the basic configuration of the CTF seismic restraints (Pacific Gas and Electric Company, 2001b). The analysis identified the loads in the restraints as well as the resulting loads at the attachment points to the cask transporter and the rock foundation. These loads included the load of the transporter, 77,110 kg [170 kips]. The weight of either the MPC, 39,920 kg [88 kips] or the HI-STORM 100 System overpack 115,210 kg [254 kips] are not included. For a suspended load the contribution of the lifted load to the horizontal and longitudinal forces are small (5 to 10 percent) as identified in ASME NOG–1

(ASME International 1995g). The amount of time that the MPC is supported by the cask transporter during the process of lowering the MPC into the storage cask is less than the transport time from the FHB/AB to the CTF. In addition, the amount of time it takes the cask transporter to raise the loaded storage cask the final 1.02 m [40 in] is less than the transport time from the FHB/AB to the CTF. The combined probability of these two events when the cask transporter is supporting additional weight and a seismic event is comparable to the review given in SER Section 15.1.2.6 for cask transporter stability. Therefore, the staff concludes that the joint probability of the annual exposure probability for these two cases where the cask transporter supports additional weight and the annual exceedance probability for the earthquake ground motion is not a credible hazard to the proposed facility. The applicant has committed to conduct structural analysis and final design of the restraints and their attachments following standard methods and professional practice. Structural analysis to be completed by the applicant should demonstrate that the restraints are designed in accordance with the applicable codes (ASME International, 1995a; American Concrete Institute, 1998) to resist the loads based on the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events, in accordance with the requirements of 10 CFR §72.122(b)(1). The structural analysis should also demonstrate that the restraints are designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions in accordance with the requirements of 10 CFR §72.122(b)(2).

HI-STORM 100SA Overpack

Structural analyses of the HI-STORM 100SA Overpack are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). A limited review was performed for this SER.

The Diablo Canyon ISFSI SAR provides a summary, in SAR Section 4.2.3, "Storage Cask Description," of the analyses performed in the HI-STORM 100 System FSAR. The loading conditions at the Diablo Canyon ISFSI are enveloped by the loading conditions considered in the HI-STORM 100 System FSAR (Holtec International, 2002). The following loads and combined loading conditions were considered:

- Dead and Live Loads (SAR Section 4.2.3.3.2.1)
- Internal and External Pressure Loads (SAR Section 4.2.3.3.2.2)
- Thermal Expansion (SAR Section 4.2.3.3.2.3)
- Handling Loads (SAR Section 4.2.3.3.2.4)
- Overpack/Transfer Cask Tip-Over and Drop (SAR Section 4.2.3.3.2.5)
- Tornado Winds and Missiles (SAR Section 4.2.3.3.2.6)
- Flood (SAR Section 4.2.3.3.2.7)
- Earthquake (SAR Section 4.2.3.3.2.8)
- Explosion Overpressure (SAR Section 4.2.3.3.2.9)
- Fire (SAR Section 4.2.3.3.2.10)
- Lightning (SAR Section 4.2.3.3.2.11)
- 500-kV Line Drop (SAR Section 4.2.3.3.2.12)

The detailed structural analysis of the HI-STORM 100SA Overpack is presented in the HI-STORM 100 System FSAR (Holtec International, 2002). The design of storage casks to

mitigate environmental effects is identified, and SAR Section 8.2, "Accidents," demonstrates the capability of SSCs important to safety to withstand postulated accidents and environmental conditions. The staff previously reviewed this structural analysis and found it acceptable, as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). As documented in that SER, the structural analysis shows that the structural integrity of the HI-STORM 100SA Overpack is maintained under all credible loads. Based on the results presented in the HI-STORM 100 System FSAR, the stresses in the overpack structures for the most critical load combinations are less than the allowable stresses of ASME Boiler and Pressure Vessel Code Section III (ASME International, 1995a) for the structure materials.

The applicant did not perform any drop or tip-over analysis. Outside the FHB/AB, tip-over of the HI-STORM 100SA Overpack is considered a noncredible accident. When on the ISFSI pad, the system will be anchored. As demonstrated in the HI-STORM 100 System FSAR, an anchored cask will not tip-over, therefore tip-over of the cask is noncredible. During transport, the overpack will be carried by the cask transporter designed to satisfy the single-failure-proof load requirement of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980).

Site-specific analysis was performed to assess the vulnerability of the HI-STORM 100SA Overpack on the storage pad to a transmission tower collapse (Holtec International, 2001a). The transmission tower collapse analysis evaluated the vulnerability of the overpack to impact by components of the transmission tower. The analysis results indicate there would be localized yielding of the overpack in the immediate vicinity of the impact site. Based on the results of the analysis, the material stress away from the vicinity of the impact does not exceed allowable values. Because the yielding is localized, there is no loss of shielding or confinement. In addition, there is no loss of retrievability of a spent nuclear fuel assembly.

Site-specific analysis was performed to assess the vulnerability of the overpack to site-specific tornado missiles (Holtec International, 2001e). The analysis demonstrated that the HI-STORM 100SA Overpack is an effective missile barrier. No missile strike compromises the integrity of the confinement boundary.

Site-specific evaluation of the risks that explosion could damage the HI-STORM 100SA Overpack were performed (Holtec International, 2001b; Pacific Gas and Electric Company, 2002c). All hazards evaluated resulted in a conclusion that these hazards were not credible.

Lightning is addressed in SAR Section 4.2.3.3.2.11, "Lightning." Site-specific evaluation of the effects of lightning and a 500-kV line break are provided (Holtec International, 2001g). The HI-STORM 100SA Overpack is a large steel/concrete cask that will discharge lightning current through the steel shell of the overpack to the ground. The conductive carbon steel overpack outer shell will provide a direct path to the ground. Since the lightning current will discharge through the overpack, the MPC will be unaffected. The heat buildup in the material will be small and there may be some local spalling of materials. Therefore, the HI-STORM 100SA Overpack design meets the Diablo Canyon ISFSI design criteria in Section 3.2.6 of the Diablo Canyon ISFSI SAR for lightning protection.

Based on a review of the PG&E site-specific loads as discussed above, the staff conclude that the Diablo Canyon ISFSI design criteria meet the loading conditions identified in the HI-STORM 100SA Overpack design. A discussion of the cask design relative to the storage requirements of the Diablo Canyon ISFSI is provided in SAR Chapter 4, "ISFSI Design." The Diablo Canyon

ISFSI SAR provides a summary of the analyses performed in the HI-STORM 100 System FSAR (Holtec International, 2002). The loading conditions at the Diablo Canyon ISFSI are enveloped by the loading conditions considered in the HI-STORM 100 System FSAR (Holtec International, 2002). Therefore, the staff conclusions in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b), with respect to the structural integrity of the HI-STORM 100SA Overpack, are valid for the Diablo Canyon ISFSI and, therefore, the system meets the Diablo Canyon ISFSI design criteria given in SAR Section 3.3, "Design Criteria for Safety Protection Systems." The structure analysis has demonstrated compliance with 10 CFR §72.24(a), §72.24(d), §72.122(b), §72.122(c), §72.122(h), §72.128(a), and §72.236(l).

Cask Transfer Facility and Lift Jacks

The staff reviewed Section 4.2.1.2 of the SAR and found that structural analysis of the CTF to mitigate environmental effects has been conducted by the applicant. The structural analysis under accident loads is given in SAR Section 8.2. The design criteria for the CTF were identified (Holtec International, 2001d). Additional information used as input into the analysis includes the bearing capacity of the rock surrounding the CTF (Pacific Gas and Electric Company, 2001c). The acceptability of these values is assessed in this SER Section 2.1.6.4.

The structural steel elements of the CTF are designed in accordance with ASME Code Section III Subsection NF (ASME International, 1995a). The analysis of the steel structures to demonstrate compliance with the material allowables was performed using a three-dimensional ANSYS finite element model of the systems (Holtec International, 2001h). This analysis addressed the following major structural elements: main shell, lifting jacks, jack support platform, CTF base support block, and lifting platform. Analyses were performed for lifting and lowering operations, MPC transfer operations, and seismic effects. The analysis evaluated the loads by considering force and moment equilibrium using the bounding values for weights. The appropriate spectral values are used to account for possible amplification of the horizontal accelerations of the stacked components. The stresses in the CTF structural components were evaluated under the combined action of the dead loads and the design basis seismic loads and then compared to the Level B allowables (ASME International, 1995a). The stresses in the CTF structural components were also evaluated under the combined action of the three Level D design-basis seismic loads and then compared to the Level D allowables. It was demonstrated that the factors of safety for all components and all load conditions are greater than 1.0. The adequacy of the structures has been demonstrated by the analysis results given in the SAR, as designed to satisfy the requirements of ASME Section III Subsection NF (ASME International, 1995a).

Loads from this analysis are also used in calculating of the necessary thickness and reinforcement for the CTF concrete (ENERCON Services Inc., 2001a). The analysis determined the required size and general reinforcing requirements to resist the loads applied to the concrete structure. Additional specific reinforcing requirements will be developed during the course of preparing the construction drawings to address issues related to discontinuities, embedments, and cutouts in the concrete wall and extensions. The concrete structure is designed to withstand loads from both the CTF and the transporter. The only configuration considered is with the HI-STORM 100SA Overpack located at the bottom of the CTF. Using the controlling load combinations, finite element analyses carried out by Holtec identified the design loads. Steel reinforcement size and placement for the pad and wall were established based on these demands. The design of the concrete structure and its reinforcement are based on the requirements in ACI 349-97 (American Concrete Institute, 1998). The procedures for selection

of the reinforcement and checks for axial, shear, moment, and torsional resistance of the elements are in conformance with standard engineering practice, as described in ACI 349-97. Results of the analysis indicate that the available design strength exceeds that required for the factored design loads.

The structural analysis performed by the applicant demonstrates that the structural elements of the CTF are designed to resist the seismic loads based on the site characteristics and environmental conditions in accordance with the requirements of 10 CFR §72.122(b)(1) and §72.122(b)(2). The PG&E analysis of the stability of the subsurface materials under the CTF and the potential for failure along the clay bed and movement of the CTF is reviewed in this SER Section 2.1.6.4, "Stability of Subsurface Materials." The slope stability of the material above the CTF and the potential for it to encroach on the facility are reviewed in this SER Section 2.1.6.5, "Slope Stability."

5.1.5 Other Structures, Systems, and Components Not Important to Safety

This section describes the design, design criteria, and design analysis for other SSCs not important to safety. There are no specific requirements identified in 10 CFR Part 72 for other SSCs not important to safety. Section 5.4.5, "Other SSCs," of NUREG–1567 (U.S. Nuclear Regulatory Commission, 1998) identifies the regulatory requirements that are applicable to other SSC subject to NRC approval.

5.1.5.1 Description of Other Structures, Systems, and Components Not Important to Safety

As identified in SAR Section 4.5.6, the following SSCs are considered.

- Security Systems (Section 4.2.2)
- Fencing (Section 4.2.2)
- Lighting (Section 4.2.2)
- Electrical Power (Section 4.4.4)
- Communications Systems (Section 4.2.2)
- Cask Transport Frame (Section 4.3.2.4)
- CTF Drive and Control Systems (Sections 4.2.1.2 and 4.4.5)

Descriptions of the other SSCs are given in the SAR sections identified necessary to satisfy the requirement of 10 CFR §72.24(a) and §72.24(b). They are limited to a general description of the various systems. The majority of these systems will be based on commercially available systems that are designed, fabricated, constructed, tested, and maintained in accordance with approved engineering practices.

The HI-STORM 100 System is a completely passive system, and no electrical power is required to ensure safe, long-term storage of the spent nuclear fuel. The cask transport frame is not within the direct load path for handling the HI-TRAC 125 Transfer Cask and, therefore, is not considered important to safety. The CTF drive and control systems will be controlled manually during operations at the CTF. The CTF structural system would fail in place, and active control is not necessary to maintain public health and safety.

The following items are only used during the 10 CFR Part 50 operations and, therefore, are not considered in this SER.

- Automated Welding System
- MPC Helium Backfill System
- MPC Force Helium Dehydration System
- MPC Vacuum Drying System

5.1.5.2 Design Criteria for Other Structures, Systems, and Components Not Important to Safety

The design criteria for the various other SSCs not important to safety that have been identified in the SAR. Table 4.1-1 of the SAR identifies details of the Diablo Canyon ISFSI compliance with the general design criteria of 10 CFR Part 72 Subpart F. The design criteria identified for other SSCs are based on commonly used codes and standards. The design of the other SSCs not important to safety permits inspection, maintenance, and testing. The inspection, maintenance, and testing requirements are based on the appropriate codes and standards. This design also allows for emergency capability. The layout of the Diablo Canyon ISFSI allows areas to be reached in the event of an accident.

5.1.5.3 Material Properties for Other Structures, Systems, and Components Not Important to Safety

No specific material properties are identified in the SAR for the other SSCs not important to safety. Material properties, however, must satisfy the code or standards used for the SSCs as required and, therefore, satisfy the requirement of 10 CFR §72.24(c)(3).

5.1.5.4 Structural Analysis for Other Structures, Systems, and Components Not Important to Safety

Other SSCs not important to safety will be designed based on standard engineering practice that are in accordance with the applicable codes and standards. In most cases, other SSCs that are not important to safety are commercially available, and their design to standard industrial requirements is acceptable. This demonstrates compliance with the requirement of 10 CFR §72.24(d) and §72.24(i) and the applicable section of 10 CFR §72.122.

5.2 Evaluation Findings

Based on the review of the SAR, the staff made the following determinations.

- Information regarding HI-STORM 100 System specific structures, systems, and components important to safety is in the HI-STORM 100 System FSAR. The NRC approval of the HI-STORM 100 System is documented in CoC 1014 and the related HI-STORM 100 System SER.

- The Diablo Canyon SAR, including the materials incorporated by reference adequately describes the materials used for the structures, systems, and components important to safety and the suitability of those materials for their intended functions in sufficient detail to evaluate their effectiveness.

- There will not be a pool or pool confinement system at the proposed Diablo Canyon ISFSI.

- The SAR adequately describes all structures, systems, and components that are important to safety, providing drawings and text in sufficient detail to allow evaluation of their structural effectiveness to meet the requirements of 10 CFR §72.24(b) and §72.24(c). The structural analysis procedures used by PG&E have been identified. The relationship between the design basis and the design criteria was identified. The materials of construction are identified. The applicable codes and standards used in the analysis of the reinforced concrete structures were established.

- The structures, systems, and components important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety functions to be performed. The structures, systems, and components important to safety are classified based on their primary function and importance to overall safety. Therefore, the requirements of 10 CFR §72.122(a) are satisfied.

- The applicant has met the requirements of 10 CFR §72.122(a). The material properties of structures, systems, and components important to safety conform to quality standards commensurate with their safety function.

- The applicant has met the requirements of 10 CFR §72.104(a), §72.106(b), §72.124, and §72.128(a)(2). Materials used for criticality control and shielding are adequately designed and specified to perform their intended function.

- The applicant has met the requirements of 10 CFR §72.122(h)(1) and §72.236(h). The design of the dry cask storage system and the selection of materials adequately protect the spent nuclear fuel cladding against degradation that might otherwise lead to gross rupture of the cladding.

- The structures, systems, and components important to safety are designed to accommodate the combined loads of normal, off-normal, accident, and natural phenomena events with an adequate margin of safety. The structural analysis performed by PG&E demonstrates that structures, systems, and components important to safety are designed to resist the loads based on the site characteristics and environmental conditions under normal operations and under postulated off-normal and accident events. The PG&E structural analysis demonstrates that structures, systems, and components important to safety are designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions. Stresses at critical locations of structures, systems, and components for bounding design loads are determined by analysis. The section properties are adjusted to ensure that the capacity of all structural elements at all locations exceeds the demand. Total stresses for the combined loads of normal, off-normal, accident, and natural phenomena events are acceptable and found to be within the limits of applicable codes, standards, and specifications. The loads on the MPC and fuel assemblies remain bounded by the loads considered in the HI-STORM 100 System FSAR. Therefore, the requirements of 10 CFR §72.122(b)(1) and §72.122(b)(2) are satisfied.

- PG&E specified that the reinforced concrete storage pads are designed in accordance with ACI 349-97 (American Concrete Institute, 1998). The final design that is covered in Calculation No. PGE–009–CALC–007 (ENERCON Services Inc., 2003a) demonstrates compliance with the applicable codes and standards. Structural analysis completed by PG&E demonstrated that the reinforced concrete storage pads are designed to resist the loads based on the site characteristics and environmental conditions during normal operations and postulated off-normal and accident events in accordance with the requirements of 10 CFR §72.122(b)(1).

- The descriptions of structures, systems, and components important to safety include consideration of inspection, maintenance, and testing. Components requiring inspection and maintenance are identified, and operational procedures are summarized. Therefore, the requirements of 10 CFR §72.122(f) are satisfied.

- This design also allows for emergency capabilities because access to critical locations and regions in the event of emergencies is possible. In addition, the lifting components are designed to hold the load in the event of emergencies. Therefore, the requirements of 10 CFR §72.122(g) are satisfied.

- The design allows for handling and storage of the limited radioactive waste generated at the ISFSI within the low-level waste storage room. Therefore, the requirements of 10 CFR §72.128(b) are satisfied.

- The applicant has met the requirements of 10 CFR §72.236(h) and §72.236(m). The material properties of structures, systems, and components important to safety will be maintained during normal, off-normal, and accident conditions of operation so the spent nuclear fuel can be readily retrieved for further processing or disposal without posing operational safety problems.

- The applicant has met the requirements of 10 CFR §72.236(g). The material properties of structures, systems, and components important to safety will be maintained during all conditions of operation so the spent nuclear fuel can be safely stored for a minimum of 20 years and maintenance can be conducted as required.

- The applicant has met the requirements of 10 CFR §72.236(h). The HI-STORM 100 System employs materials that are compatible with wet and dry spent nuclear fuel loading and unloading operations and facilities. These materials should not degrade over time or react with one another during any conditions of storage.

5.3 References

American Concrete Institute. *Code Requirements for Nuclear Safety Related Concrete Structures.* ACI 349-85. Detroit, MI: American Concrete Institute. 1985.

American Concrete Institute. *Code Requirements for Nuclear Safety Related Concrete Structures.* ACI 349-97. Detroit, MI: American Concrete Institute. 1998.

American Concrete Institute. *Building Code Requirements for Structural Concrete.* ACI 318-02. Detroit, MI: American Concrete Institute. 2002.

American National Standards Institute/American Nuclear Society. *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).* ANSI/ANS 57.9. La Grange Park, IL: American National Standards Institute/American Nuclear Society. 1992.

American National Standards Institute/American Nuclear Society. *Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More.* ANSI N14.6. La Grange Park, IL: American National Standards Institute. 1993.

American Society for Testing and Materials. *Annual Book of ASTM Standards, Section 1, Volume 01.04.* Philadelphia, PA: American Society for Testing and Materials. 1990.

ASME International. *ASME Boiler and Pressure Vessel Code, Section III, Division 1.* New York City, NY: ASME International. 1995a.

ASME International. *ASME Boiler and Pressure Vessel Code, Section V, Nondestructive Examination Methods.* New York City, NY: ASME International. 1995b.

ASME International. *ASME Boiler and Pressure Vessel Code, Section IX, Welding and Brazing Qualifications.* New York City, NY: ASME International. 1995c.

ASME International. *ASME Boiler and Pressure Vessel Code, Section II, Materials, Part A–Ferrous Materials.* New York City, NY: ASME International. 1995d.

ASME International. *ASME Boiler and Pressure Vessel Code, Section II, Materials, Part B–Non Ferrous Materials.* New York City, NY: ASME International. 1995e.

ASME International. *ASME Boiler and Pressure Vessel Code, Section II, Materials, Part D–Properties.* New York City, NY: ASME International. 1995f.

ASME International. *Rules for Construction of Overhead and Gantry (Top Running Bridge, Multiple Girders).* New York City, NY: ASME International. 1995g.

ASME International. *Slings. B30.9-1996 through B309C-2000.* Addenda, New York City, NY: ASME International. 2000.

American Welding Society. *AWS Standard A2.4—Standard Symbols for Welding, Brazing, and Nondestructive Examination.* Miami, FL: American Welding Society. 1998.

American Welding Society. *AWS D1.1 Structural Welding Code Steel.* Miami, FL: American Welding Society. 2002.

ENERCON Services Inc. *Cask Transfer Facility (Reinforced Concrete).* PGE–009–CALC–002, Rev 1. Tulsa, OK: ENERCON Services Inc. December 17, 2001a.

ENERCON Services Inc. *ISFSI Cask Storage Pad Seismic Analysis.* PGE–009–CALC–003, Rev 2. Tulsa, OK: ENERCON Services Inc. December 14, 2001b.

ENERCON Services Inc. *ISFSI Cask Storage Pad Steel Reinforcement.* PGE-009-CALC-007. Tulsa, OK: ENERCON Services Inc. March 11, 2003a.

ENERCON Services Inc. *Embedment Support Structure.* PGE-009-CALC-001, Rev 5. Tulsa, OK: ENERCON Services Inc. March 11, 2003b.

ENERCON Services Inc. *ISFSI Cask Storage Pad Concrete Shrinkage and Thermal Stresses.* PGE–009–CALC–006, Rev. 1 Tulsa, OK: ENERCON Services Inc. March 3, 2003c.

ENOVA Engineering Services. *Seismic Stability Analysis of Transporter on Soil.* Calculation No. 0104-021-C01. Walnut Creek, CA: ENOVA Engineering Services. September 23, 2002.

Holtec International. *Acceptable Flaw Size in MPC Lid-To-Shell Welds. Position paper DS–213 Rev. 2.* Marlton, NJ: Holtec International. February 23, 1999.

Holtec International. *Analysis of Transmission Tower Collapse Accidents at the Diablo Canyon ISFSI Pad and CTF.* HI–2012634, Rev 1. Marlton, NJ: Holtec International. March 21, 2001a.

Holtec International. *Evaluation of Site-Specific Blast and Explosions for the Diablo Canyon ISFSI.* HI–2002512, Rev 2. Marlton, NJ: Holtec International. March 29, 2001b.

Holtec International. *Analysis of Anchored HI-STORM Casks at the Diablo Canyon ISFSI.* HI–2012618, Rev 5. Marlton, NJ: Holtec International. December 11, 2001c.

Holtec International. *Design Criteria Document for the Diablo Canyon Cask Transfer Facility.* HI–2002570, Rev 3. Marlton, NJ: Holtec International. October 22, 2001d.

Holtec International. *Design Basis Wind and Tornado Evaluation for DCPP.* HI–2002497, Rev 1. Marlton, NJ: Holtec International. April 20, 2001e.

Holtec International. *Transporter Stability on Diablo Canyon Dry Storage Travel Paths.* HI–20012768, Rev 2. Marlton, NJ: Holtec International. November 14, 2001f.

Holtec International. *Evaluation of the Effects of Lightning and a 500 KV Line Break on Holtec Casks.* HI–2002559, Rev 1. Marlton, NJ: Holtec International. February 26, 2001g.

Holtec International. *Structural Evaluation of Diablo Canyon Cask Transfer Facility.* HI–2012626, Rev 7. Marlton, NJ: Holtec International. December 12, 2001h.

Holtec International. *Functional Specification for the Diablo Canyon Cask Transporter.* HI–2002501, Rev 5. Marlton, NJ: Holtec International. November 9, 2001i.

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System Revision 1 (HI-STORM 100 Cask System).* Vols I and II. HI–2002444. Docket 72-1014. Marlton, NJ: Holtec International. 2002.

MSC Software Corporation. *VisualNastran Desktop.* Version 2001. Santa Ana, CA: MSC Software Corporation. 2001.

Pacific Gas and Electric Company. *ISFSI Foundation Pad—Thermal and Shrinkage Values.* Calculation 52.27.100.701, Rev 2. Avila Beach, CA: Pacific Gas and Electric Company. November 8, 2001a.

Pacific Gas and Electric Company. *Cask Transfer Facility Seismic Restraint Configuration.* Calculation M–1058, Rev 2. Avila Beach, CA: Pacific Gas and Electric Company. December 11, 2001b.

Pacific Gas and Electric Company. *Development of Lateral Bearing Capacity for DCPP CTF Stability Analysis.* Calculation 52.27.100.716. Avila Beach, CA: Pacific Gas and Electric Company. November 15, 2001c.

Pacific Gas and Electric Company. *Non-Linear Seismic Sliding Analysis of the ISFSI Pad.* Calculation 52.27.100.704, Rev 0. Avila Beach, CA: Pacific Gas and Electric Company. November 30, 2001d.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation Safety Analyses Report Amendment 1.* Docket No. 72-26. Avila Beach, CA: Pacific Gas and Electric Company. 2002a.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation License Application Attachment C, Proposed Technical Specifications.* Docket No. 72-26. Avila Beach, CA: Pacific Gas and Electric Company. 2002b.

Pacific Gas and Electric Company. *Risk Assessment of Dry Cask/Spent Fuel Transportation Within the DCPP Owner Controlled Area.* Calculation File No. PRA01.01, Revision 01. Avila Beach, CA: Pacific Gas and Electric Company. 2002c.

Pacific Gas and Electric Company. *Response to NRC Request for Additional Information for the Diablo Canyon Independent Spent Fuel Storage Installation Application (TAC No. L23399).* Letter DIL–02–009. Avila Beach, CA: Pacific Gas and Electric Company. 2002d.

U.S. Nuclear Regulatory Commission. *Control of Heavy Loads at Nuclear Power Plants, Resolution of Generical Technical Activity A–36.* NUREG–0612. Washington, DC: U.S. Nuclear Regulatory Commission. 1980.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Spent Fuel Dry Storage Facilities.* NUREG–1567. Washington, DC: U.S. Nuclear Regulatory Commission. 1998.

U.S. Nuclear Regulatory Commission. *10 CFR Part 72 Certificate of Compliance No. 1014, Amendment 1, for the HI-STORM 100 Cask System.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. July 15, 2002a.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report, Amendment 1.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. July 15, 2002b.

6 THERMAL EVALUATION

6.1 Conduct of Review

Review of the thermal evaluation included Safety Analysis Report (SAR) Sections 2.2.2.2, "Hazards From Fires;" 2.2.2.3, "Onsite Explosion Hazards;" 2.3.2, "Local Meteorology;" 3.1.1, "Materials to be Stored;" 3.2.7, "Temperature and Solar Radiation;" 3.3.1.6, "Fire and Explosion Protection;" 4.2.3, "Storage Cask Description;" 4.4.3.6, "Thermal Performance;" 5.1, "Operation Description;" 8.1.2, "Off-Normal Environmental Temperatures;" 8.1.4, "Partial Blockage of Air Inlets;" 8.2.5, "Fire;" 8.2.6, "Explosion;" and 10.2, "Development of Operating Controls and Limits;" of the SAR (Pacific Gas and Electric Company, 2002a) and responses to U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAI).[1] A Probabilistic Risk Assessment[2] pertaining to explosion hazards was also reviewed. The design of the proposed Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) Facility is based on the use of the HI-STORM 100 System as certified in accordance with 10 CFR Part 72 by the U.S. Nuclear Regulatory Commission (2002a) and as described in the Final Safety Analysis Report (FSAR) for the HI-STORM 100 System (Holtec International, 2002).

6.1.1 Decay Heat Removal Systems

The staff reviewed the discussion on decay heat removal systems with respect to the following regulatory requirements:

- 10 CFR §72.122(h)(1) requires the spent fuel cladding to be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

- 10 CFR §72.128(a) requires spent fuel storage, and other systems that might contain or handle radioactive materials associated with spent fuel must be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with (1) A capability to test and monitor components important to safety, (2) Suitable shielding for radioactive protection under normal and accident conditions, (3) Confinement structures and systems, (4) a heat-removal capability having testability and reliability consistent with its importance to safety, and (5) means to minimize the quantity of radioactive wastes generated.

---

[1]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Spent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

[2]Womack, L.F. Letter (July 28) DIL–03–010, Enclosure 2 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

The HI-STORM 100SA storage cask is designed to remove decay heat primarily by convective heat transfer. An active cooling system is not used. The storage cask is equipped with four inlet vents at the bottom and four outlet vents on top. Cool air is drawn into the annulus between the canister and storage cask through the bottom inlet vents. The buoyancy created by the heating of the air creates a chimney effect, and the air flows back into the environment through the outlet vents at the top of the cask. The Certificate of Compliance (CoC) (U.S. Nuclear Regulatory Commission, 2002a, Appendix A) includes surveillance requirements for ensuring that the cask heat-removal system is operational during storage (i.e., the air ducts are inspected or the temperature differential between the convected cooling air exiting the outlet vents and ambient air is measured every 24 hours to ensure that the ducts are free of blockages). The storage cask design and heat-removal capability are described and evaluated in the HI-STORM 100 System FSAR (Holtec International, 2002). As documented in the HI-STORM 100 System Safety Evaluation Report (SER) (U.S. Nuclear Regulatory Commission, 2002b), the staff has previously determined that the HI-STORM 100SA storage cask provides adequate heat-removal capacity under normal storage conditions as long as the fuel specifications and loading conditions as defined in the CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix B) are adhered to and the environmental characteristics of the site are bounded by the corresponding design criteria (see Section 6.1.3 of this SER).

As with the HI-STORM 100SA storage cask, the HI-TRAC 125 Transfer Cask is designed to remove decay heat primarily by convective heat transfer. An active cooling system is not used. From the outer surface of the Multi-Purpose Canister (MPC) to the ambient atmosphere, heat is transported within the transfer cask through multiple concentric layers of air, steel, and shielding materials. Heat must propagate through a total of six concentric layers, representing the air gap, the HI-TRAC 125 inner shell, the lead shielding, the outer shell, the water jacket, and the enclosure shell. Heat is dissipated to the atmosphere predominantly by natural convection from the surface of the enclosure shell. The transfer cask design and heat-removal capability are described and evaluated in the HI-STORM 100 System FSAR (Holtec International, 2002). As documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b), the staff has previously determined that the HI-TRAC 125 Transfer Cask provides adequate heat-removal capacity during normal transport conditions as long as the fuel specifications and loading conditions as defined in the CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix B) are adhered to and the environmental characteristics of the site are bounded by the corresponding design criteria (see Section 6.1.3 of this SER). Furthermore, the CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix B) limits the minimum ambient temperature for conducting transfer cask loading, onsite transport, and unloading operations to -18 °C [0 °F].

The staff reviewed the information provided by the applicant regarding the spent nuclear fuel heat-removal capacity of the HI-STORM 100 System for normal, off-normal, and accident conditions. The staff found the analysis acceptable because

- The staff has previously determined (U.S. Nuclear Regulatory Commission, 2002b) that the HI-STORM 100 System provides adequate heat-removal capacity for normal, off-normal, and accident conditions so long as the fuel specifications as defined in the CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix B) are adhered to and the environmental characteristics of the site are bounded by the corresponding design criteria (see Section 6.1.3 of this SER).

- The Diablo Canyon ISFSI fuel specifications and normal, off-normal, and accident-loading conditions identified in the SAR are sufficient to ensure the decay heat-removal capacities of the HI-STORM 100 System will not be exceeded.

Staff review of the submitted information gave reasonable assurance that the Diablo Canyon ISFSI spent nuclear fuel will not exceed the decay heat-removal capacities of the HI-STORM 100 System. Based on the foregoing evaluation, the staff finds that the requirements of 10 CFR §72.122(h)(1) and §72.128(a) have been adequately satisfied.

6.1.2 Material Temperature Limits

The staff reviewed the discussion on material temperature limits with respect to the following regulatory requirement:

- 10 CFR §72.122(h)(1) requires the spent nuclear fuel cladding to be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This condition may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

The material temperature limits for components of the HI-STORM 100 System storage and transfer casks and spent nuclear fuel cladding are given in the HI-STORM 100 System FSAR, which was previously reviewed and found to be acceptable by the staff (U.S. Nuclear Regulatory Commission, 2002a,b). These material temperature limits have been established for normal, off-normal, and accident conditions. In the case of the spent nuclear fuel cladding, the established temperature limits also take into consideration the fuel age at initial loading and the level of burnup.

The characteristics of the spent nuclear fuel to be stored at the Diablo Canyon ISFSI are bounded by the previously approved contents for the HI-STORM 100SA storage cask. The proposed Diablo Canyon ISFSI is designed to provide interim storage for 4,400 fuel assemblies, which will accommodate the number of assemblies planned to be used during the licensed operating life of the Diablo Canyon Power Plant (DCPP) (Pacific Gas and Electric Company, 2002a). Approved contents for the HI-STORM 100 System storage and transfer casks are provided in the CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix B). The characteristics of the spent nuclear fuel to be stored at the Diablo Canyon ISFSI are provided in the SAR (Pacific Gas and Electric Company, 2002a, Tables 3.1-1 and 3.1-2 and Section 10.2). A description of the methods used to assess damaged fuel is provided in the Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2002a, Section 10.2.1.1).

Low-burnup fuel (i.e., burnups less than or equal to 45,000 MWd/MTU) proposed to be stored at the Diablo Canyon ISFSI is clad with zirconium-based alloys, including ZIRLO and Zircaloy-4, and have been previously approved for storage in the HI-STORM 100 System (U.S. Nuclear Regulatory Commission, 2002a,b). The low-burnup fuel is subject to the assembly-specific physical parameters, burnup, cooling time, and decay heat limits specified in the CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix B), in the proposed Diablo Canyon ISFSI technical specifications (Pacific Gas and Electric Company, 2002b, Attachment C:

Tables 2.1-1 through 2.1-10), and in the Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2002a, Tables 10.2-1 through 10.2-9).

High-burnup fuel is limited to fuel with Zircaloy-4 cladding. The spent nuclear fuel cladding integrity for high-burnup fuel for the proposed ISFSI will be evaluated by calculating the allowable corrosion reserve using the methodology documented in the HI-STORM 100 System CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix A: Section 5.6). The maximum allowable fuel cladding oxidation layer thicknesses for LOPAR and VANTAGE 5 fuel assemblies are listed in the proposed Diablo Canyon ISFSI technical specifications (Pacific Gas and Electric Company, 2002b, Attachment C, Section 5.1.3) and were determined using the methodology documented in the HI-STORM 100 System CoC (U.S. Nuclear Regulatory Commission, 2002a Appendix A, Section 5.6) using DCPP specific fuel characteristics and internal rod pressure. The site-specific value for internal fuel rod pressure is more conservative than the generic thermal evaluations provided in the HI-STORM 100 System FSAR (Holtec International, 2002, Table 4.A.4). The staff independently calculated the allowable corrosion reserve and verified the maximum allowable average fuel-cladding oxidation layer thickness listed in the proposed technical specifications for the Diablo Canyon ISFSI (Pacific Gas and Electric Company, 2002b, Attachment C) and in the HI-STORM 100 System FSAR (Holtec International, 2002, Table 4.A.4).

Fuel-cladding temperatures under normal, off-normal, and accident conditions are provided in the HI-STORM 100 System FSAR (Holtec International, 2002, Sections 4.4, 4.5, 11.1, 11.2). Fuel-cladding temperature limits for long-term storage of zirconium-alloy clad, low-burnup fuel applicable to the proposed Diablo Canyon ISFSI are listed in HI-STORM 100 System FSAR (Holtec International, 2002, Table 4.3-7). Long-term Zircaloy-4 high-burnup fuel-cladding temperature limits are also provided in the HI-STORM 100 System FSAR (Holtec International, 2002, Table 4.A.2).

Allowable long term peak cladding temperature limits for low- and high-burnup Zircaloy-clad Pressurized Water Reactor fuel are summarized in Table 6-1. For low-burnup fuel ($\leq$ 45,000 MWd/MTU), the temperature limit is determined by the bounding values of cladding stress and the permissible Pacific Northwest National Laboratory cladding temperature limit (Levy, et al., 1987). For low burnup fuel, a short-term temperature limit of 570 °C [1058 °F] is applicable to off-normal and accident conditions (Pacific Gas and Electric Company, 2001a). For MPCs containing moderate burnup fuel ($\leq$ 45,000 MWd/MTU), vacuum drying can be used to remove the remaining moisture in the MPC cavity. The temperature limits for normal operations are consistent with the approved Pacific Northwest Laboratory cladding temperature limits for low burnup fuel (Levy et al., 1987)

Temperature limits for high burnup fuel (> 45,000 MWd/MTU) were determined using the creep model developed by Holtec and documented in the HI-STORM 100 System FSAR (Holtec International, 2002, Section 4.A). For high burnup fuel, temperature limits for transfer and vacuum drying operations correspond to the normal storage temperature limits delineated in Table 6-1 (U.S. Nuclear Regulatory Commission, 2002b, Section 4.1.5). A forced helium dehydration system is required to remove the remaining moisture in the canister cavity for MPCs that contain at least one high burnup fuel assembly. For high burnup fuel, the temperature limits during normal and short-term operations are below the allowable limits based on the approved Holtec International creep model. These limits are consistent with guidance

Table 6-1. Allowable temperature limits for low and high-burnup Pressurized Water Reactor fuels (Holtec International, 2002; Tables 4.3.7, 4.A.2).

Fuel Age at Initial Loading Years	Low-Burnup Zircaloy-Clad Temperature Limit °C [°F]	High-Burnup Zircaloy-Clad Temperature Limit °C [°F]
5	366.0 [691]	359.7 [679]
6	358.0 [676]	348.7 [660]
7	335.0 [635]	335.0 [635]
10	329.6 [625]	327.2 [621]
15	323.2 [614]	321.9 [611]

provided in Interim Staff Guidance-11, Revision 2 (U.S. Nuclear Regulatory Commission, 2002c). For high burnup fuel the temperature limit of 570 °C [1,058 °F] for off-normal and accident conditions is consistent with the guidance provided in Interim Staff Guidance-11, Revision 2 (U.S. Nuclear Regulatory Commission, 2002c).

The staff reviewed the information provided by the applicant pertaining to the Diablo Canyon ISFSI material temperature limits for normal, off-normal, and accident conditions. The staff found the analysis acceptable because

The staff has previously determined (U.S. Nuclear Regulatory Commission, 2002b) that the material temperature limits of the HI-STORM 100 System will not be exceeded for normal, off-normal, and accident conditions so long as the fuel specifications as defined in the CoC (U.S. Nuclear Regulatory Commission, 2002a Appendix B) are adhered to and the environmental characteristics of the site are bounded by the corresponding design criteria (see Section 6.1.3 of this SER).

- The temperature limits for normal operations are consistent with the approved Pacific Northwest National Laboratory cladding temperature limits for low-burnup spent nuclear fuels (Levy, et al., 1987).
- For high-burnup spent nuclear fuels, the temperature limits during normal operations are below the allowable limits based on the approved Holtec International creep model (Holtec International, 2002, Section 4.A).
- The short-term temperature limit of 570 °C [1,058 °F] for both low- and high-burnup spent nuclear fuel is consistent with the guidance provided in Interim Staff Guidance-15 (U.S. Nuclear Regulatory Commission, 2001).

Staff review of the submitted information gave reasonable assurance that the Diablo Canyon ISFSI Facility is adequately designed to maintain important to safety SSC material temperatures below the specified limits. Based on the foregoing evaluation, the requirements of 10 CFR §72.122(h)(1) have been adequately satisfied.

6.1.3 Thermal Loads and Environmental Conditions

The staff reviewed the discussion on thermal loads and environmental conditions with respect to the following regulatory requirements:

- 10 CFR §72.92(a) requires that the natural phenomena that may exist or that can occur in the region of a proposed site be identified and assessed according to their potential effects on the safe operation of the ISFSI. The important natural phenomena that affect the ISFSI design must be identified.

- 10 CFR §72.122(b) requires that (1) structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. (2)(i) structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform their intended design functions. The design bases for these structures, systems, and components must reflect: (A) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (B) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (ii) The ISFSI also should be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel waste or on to structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety.

The meteorological conditions of the proposed ISFSI are documented in the DCPP FSAR Update (Pacific Gas and Electric Company, 2001, Section 2.3), which is maintained in accordance with 10 CFR §50.71(e). The SAR for the proposed ISFSI (Pacific Gas and Electric Company, 2002a, Section 2.3) provided a summary of the relevant meteorological information and data documented in the DCPP FSAR Update. The meteorological conditions for the proposed ISFSI site and the adjacent power plant are expected to be the same because of their close proximity to each other {i.e., approximately 0.35 km [0.22 mi]} and negligible difference in elevation {i.e., approximately 68.6 m [225 ft]}. The highest hourly temperature recorded at the site was 36 °C [97 °F], which occurred in October 1987. Temperatures below freezing were measured over a several hour period in December 1990. The average annual temperature at the site is approximately 13 °C [55 °F]. In addition, Table 6-2 conveys relevant monthly and annual mean temperatures for Morro Bay, California (Pacific Gas and Electric Company, 2001, Table 2.3-7), which is located on the California coast approximately 16 km [10 mi] northwest of the DCPP site. The mean temperatures documented in Table 6-2 were derived from data accumulated during a 14-year period.

Table 6-2. Temperatures for Morro Bay, California (Pacific Gas and Electric Company, 2001, Table 2.3-7)

Month	Mean Temperature °C [°F]	Mean Maximum Temperature °C [°F]	Mean Minimum Temperature °C [°F]	Extreme Maximum Temperature °C [°F]	Extreme Minimum Temperature °C [°F]
January	11.4 [52.6]	16.7 [62.0]	6.2 [43.2]	27.8 [82]	-1.1 [30]
February	12.1 [53.8]	17.2 [63.0]	7.0 [44.6]	27.8 [82]	-1.1 [30]
March	11.7 [53.1]	16.9 [62.5]	6.4 [43.6]	29.4 [85]	0.0 [32]
April	12.3 [54.1]	17.5 [63.5]	7.1 [44.7]	33.9 [93]	0.6 [33]
May	12.8 [55.1]	17.2 [62.9]	8.5 [47.3]	36.7 [98]	0.6 [33]
June	14.2 [57.5]	18.0 [64.4]	10.3 [50.5]	36.7 [98]	4.4 [40]
July	14.6 [58.2]	18.4 [65.1]	10.7 [51.3]	31.7 [89]	1.1 [34]
August	13.1 [55.5]	19.3 [66.7]	11.5 [52.7]	34.4 [94]	7.2 [45]
September	15.9 [60.7]	20.4 [68.8]	11.4 [52.5]	38.3 [101]	6.1 [43]
October	16.0 [60.8]	21.4 [70.5]	10.6 [51.0]	37.2 [99]	3.3 [38]
November	13.9 [57.0]	18.9 [66.0]	8.8 [47.8]	33.3 [92]	0.0 [32]
December	11.3 [52.4]	16.4 [61.6]	6.2 [43.2]	26.1 [79]	-1.7 [29]
Annual	13.3 [55.9]	18.2 [64.8]	8.7 [47.7]	38.3 [101]	-1.7 [29]

The extreme ambient temperature range design criteria for the proposed ISFSI is -4 °C [24 °F] to 40 °C [104 °F] (Pacific Gas and Electric Company, 2002a, Table 3.4-1). These temperatures correspond to the extreme minimum and maximum temperatures recorded at Pismo Beach, California (Pacific Gas and Electric Company, 2001, Table 2.3-7) during a 12-year period. Pismo Beach is also located on the California coast approximately 24 km [15 mi] east-southeast of the proposed Diablo Canyon ISFSI Facility site. These extreme temperatures bound the values recorded for Morro Bay, California (see Table 6-2). In addition, the mean daily maximum temperature (defined as the mean of peak temperatures for a month) for Morro Bay, California, is 21.4 °C [70.5 °F].

According to the HI-STORM 100 System CoC (U.S. Nuclear Regulatory Commission, 2002a), the maximum average yearly temperature allowed for the site is 26.7 °C [80 °F]. Moreover, the allowed temperature extremes, averaged during a 3-day period, shall be greater than -40 °C [-40 °F] and less than 51.7 °C [125 °F]. Based on information provided in the DCPP FSAR (Pacific Gas and Electric Company, 2001, Table 2.3-7) and proposed ISFSI SAR (Pacific Gas and Electric Company, 2002a), the cask design criteria (Holtec International, 2002) bound the temperatures measured at the site and nearby towns.

According to the proposed ISFSI SAR (Pacific Gas and Electric Company, 2002a, Section 2.3.2), solar-insolation data collected by the California Polytechnic State University, Department of Water Resources, and cataloged in the California Irrigation Management Information System, are applicable to the Diablo Canyon ISFSI site. These data are measured at approximately 19.3 km [12 mi] northeast of the proposed Diablo Canyon ISFSI Facility. For data collected between 1986 and 1999, the maximum measured insolation for a 24-hour period was 766 g-cal/cm^2 per day {371 W/m^2 [118 BTU/hr-ft^2]} and, for a 12-hour period, 754 g-cal/cm^2 per day {365 W/m^2 [116 BTU/hr-ft^2]}. The daily (24 hour) average for the period of record was 430 g-cal/cm^2 per day {208 W/m^2 [66 BTU/hr-ft^2]}.

As reported by Holtec International (2000, Table 4.1; 2002, Section 4.4.1.1), the models analyzing the transport of heat from the heat-generation regions of the HI-STORM 100 System storage and transfer casks to the outside environment use insolation boundary conditions on the cask surfaces that are based on the 12-hour levels prescribed in 10 CFR Part 71, averaged over a 24-hour period, after accounting for partial blockage conditions on the sides of the overpack. These insolation values correspond to 800 g-cal/cm^2 per day {387 W/m^2 [123 BTU/hr-ft^2]}. Based on the foregoing information, the staff concludes that the insolation values used to evaluate the thermal performance of the HI-STORM 100 System storage and transfer casks adequately bound the site-specific insolation design parameters.

Table 4-2 of the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b) indicates that the bounding annual average earth temperature for the HI-STORM 100SA storage cask is 25 °C [77 °F]. Although the average storage pad or earth temperature was not explicitly provided nor addressed in the SAR for the proposed ISFSI, it can be inferred that this temperature is less than the bounding 25 °C [77 °F] requirement. This inference is based on the recognition that the average annual earth temperature will not be greater than the average annual air temperature at the site, which is reported to be approximately 13 °C [55 °F].

The staff reviewed the local meteorological data and discussions presented in the SAR (Pacific Gas and Electric Company, 2002a) and found them acceptable because reliable data sources were used and the data are appropriately summarized. The applicant adequately presented information regarding temperatures recorded by the onsite measurement program and at other nearby sites, and, therefore, satisfied the requirement of 10 CFR §72.92(a). The staff confirmed that temperatures and solar loads at the site are bounded by the HI-STORM 100 System storage and transfer casks design parameters.

The staff reviewed the information provided by the applicant pertaining to the Diablo Canyon ISFSI thermal loads and environmental conditions. The staff found the analysis acceptable because

- Reliable data sources have been used to present temperatures and solar insolation at nearby sites.
- Environmental data recorded during the onsite measurement program correlate well with data recorded at nearby sites.
- The temperatures and solar loads at the Diablo Canyon ISFSI site are bounded by the HI-STORM 100 System design parameters.

Staff review of the submitted information gave reasonable assurance that the Diablo Canyon ISFSI Facility thermal loads and environmental conditions are bounded by the HI-STORM 100 System design parameters. Based on the foregoing evaluation, the requirements of 10 CFR §72.92(a) and §72.122(b) have been adequately satisfied.

6.1.4 Analytical Methods, Models, and Calculations

The staff reviewed the discussion on analytical methods, models, and calculations with respect to the following regulatory requirements:

- 10 CFR §72.122(h)(1) requires the spent fuel cladding to be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

- 10 CFR §72.128(a) requires spent fuel storage, and other systems that might contain or handle radioactive materials associated with spent fuel must be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with (1) a capability to test and monitor components important to safety, (2) suitable shielding for radioactive protection under normal and accident conditions, (3) confinement structures and systems, (4) a heat-removal capability having testability and reliability consistent with its importance to safety, and (5) means to minimize the quantity of radioactive waste generated.

The staff reviewed the information provided by the applicant pertaining to the Diablo Canyon ISFSI analytical methods, models, and calculations. The staff found the analysis acceptable because

- The analytical methods, models, and calculations used to establish the thermal characteristics of the HI-STORM 100 System were previously reviewed and found to be acceptable by the staff (U.S. Nuclear Regulatory Commission, 2002a,b).

Staff review of the submitted information gave reasonable assurance that the Diablo Canyon ISFSI Facility is adequately designed to protect SSCs important to safety from all postulated normal, off-normal, and accident thermal loads and environmental conditions. Based on the foregoing evaluation, the requirements of 10 CFR §72.122(h)(1) and §72.128(a) have been adequately satisfied.

6.1.5 Fire and Explosion Protection

6.1.5.1 Fire

The staff reviewed the discussion on fire with respect to the following regulatory requirement:

- 10 CFR §72.122(c) requires that structures, systems, and components important to safety be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

The proposed Diablo Canyon ISFSI Facility is collocated with the DCPP. The dry-storage facility will be within the Pacific Gas and Electric Company (PG&E) owner-controlled DCPP area. A plan drawing of the proposed ISFSI was provided in Figure 2.1-2 of the SAR (Pacific Gas and Electric Company, 2002a). The loaded HI-STORM 100SA storage casks will be anchored to concrete storage pads within a separate protected area. A farm-type security fence is used to demarcate the protected area. According to Figure 2.1-2 of the SAR, the restricted area fence, which encompasses the security fence, is at least 30.5 m [100 ft] from the storage casks, except where the local roadway that provides access to the vehicle maintenance shop passes within 15.3 m [50 ft] of the dry-storage area. Up to seven storage pads will be constructed within the protected area to accommodate 140 storage casks (i.e., 20 storage casks per pad arranged in a 4 by 5 rectilinear array) over a 152 × 32-m [500 × 105-ft] footprint. The dry-storage protected area will have applicable barrier, access, and surveillance controls consistent with the requirements of 10 CFR §73.55. The Canister Transfer Facility (CTF) is located next to the storage area but outside the protected area.

Other than the DCPP, no industrial facilities, public transportation routes, rail lines, or military bases are within 8 km [5 mi] of the dry-storage site. Local shipping tankers may come within 16 km [10 mi] of the coast site, but will remain outside a 8-km [5-mi] range. No natural gas or other pipelines pass within 8 km [5 mi] of the site. Similarly, no combustible or explosive materials, beyond those associated with the DCPP, are stored within 8 km [5 mi] of the site.

Locations pertaining to the proposed ISFSI that fall within the purview of the 10 CFR Part 72 review are the transport route from the DCPP Fuel-Handling Building/Auxiliary Building (FHB/AB) to the CTF, the CTF, and the cask storage area. The postulated fire events that could potentially affect these areas that have been identified by Pacific Gas and Electric Company (2002a, Subsection 8.2.5) and the Probabilistic Risk Assessment[3] are

(1) An onsite cask transporter fuel tank fire
(2) Other onsite vehicle fuel tank fires
(3) Combustion of other local stationary fuel tanks
(4) Combustion of other local combustible materials
(5) Fire in the surrounding vegetation
(6) Electrical transformer fire

[3]Womack, L.F. Letter (July 28) DIL–03–010, Enclosure 2 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

The onsite cask transporter is used to move the spent nuclear fuel (contained within the MPC) from the FHB/AB to the CTF using the HI-TRAC 125 Transfer Cask. After the MPC has been transferred to the HI-STORM 100SA storage cask at the CTF, the cask transporter is used to move the storage cask onto the storage pad. To ensure the potential exposure of the HI-TRAC 125 transfer and HI-STORM 100SA storage casks to a fire remain within preapproved limits, the diesel fuel tank used for the transporter is limited to a 189-L [50-gal] capacity. The ability of the HI-TRAC 125 Transfer Cask and HI-STORM 100SA storage casks to provide confinement and protect the spent nuclear fuel from gross degradation as the result of a 189-L [50-gal] diesel fuel fire was previously reviewed and found to be acceptable by the staff (U.S. Nuclear Regulatory Commission, 2002a,b).

Administrative controls will be implemented to ensure transient sources of fuel in volumes larger than 189 L [50 gal] will be a sufficient distance from the dry-storage area pads at all times, the CTF during active MPC transfer operations, and the transport route during cask transport (Pacific Gas and Electric Company, 2002a, Section 8.2.5.2). According to the Probabilistic Risk Assessment,[4] however, a 3,028-L [800-gal] fuel tanker truck will use the transport route near the storage area to deliver fuel to the vehicle maintenance shop located approximately 610 m [2,000 ft] northeast of the storage area six times per week. This amount of fuel exceeds the 189-L [50-gal] limit. Recall that the transport route passes within 15.3 m [50 ft] of the storage area. To determine the potential consequences of a fuel tanker truck fire occurring near the storage area, a bounding 7,570-L [2,000-gal] fire-loading analysis was submitted[5] to assess the potential effects on the HI-TRAC 125 Transfer Cask, which bounds the potential effects on a HI-STORM 100SA storage cask. This fire-loading analysis adequately demonstrated that a nonengulfing 7,570-L [2,000-gal] fuel tanker fire will not adversely affect the HI-TRAC 125 Transfer Cask and HI-STORM 100SA storage casks.

There is at least a 30.5-m [100-ft] clearance between the storage area, CTF, and the cask transport route and any onsite stationary fuel tanks (Pacific Gas and Electric Company, 2002a, Section 2.2.2.2). Onsite stationary fuel tanks include

(1) Three fuel tanks {946 L [250 gal] propane; 7,571 L [2,000 gal] No. 2 diesel; and 11,356 L [3,000 gal] gasoline} located beside the main plant road, 366 m [1,200 ft] from the cask transport route at its nearest point

(2) The Unit 2 main bank transformer mineral oil storage tank, located approximately 73 m [240 ft] from the transport route

(3) Bulk hydrogen storage facility (consisting of six individual hydrogen storage tanks) located approximately 4.6 m [15 ft] from the transport route

The stationary propane, diesel, and gasoline fuel tanks are far enough away from the transporter route so as not to pose a fire hazard to the HI-TRAC 125 Transfer Cask. The 7,570-L [2,000-gal] fuel tanker fire-loading analysis adequately demonstrated that nonengulfing

---

[4]Womack, L.F. Letter (July 28) DIL–03–010, Enclosure 2 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[5]Womack, L.F. Letter (March 27) DIL–03–005, Attachment 1, Calculation No. M–1052 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

fires originating from the mineral oil and hydrogen storage tanks will not adversely affect the HI-TRAC 125 transfer cask.

No combustible materials will be stored within the confines of the storage area demarcated by the security fence.

The native vegetation surrounding the storage area is primarily grass. Maintenance programs will prevent uncontrolled growth of the surrounding vegetation. Moreover, the consequences of this potential fire hazard is bounded by the 7,570-L [2,000-gal] fuel tanker fire-loading analysis.

Electrical transformers are located approximately 73 m [240 ft] from the HI-TRAC 125 Transfer Cask transport route. The mineral oil within these transformers could be ignited by lightning strike, vehicle crash, or internal electrical faults.[6] Administrative procedures that prohibit transport of the transfer cask during inclement weather and DCPP transition operations significantly reduce the potential for transformer mineral oil fire being ignited by lightning or internal electrical faults. Administrative procedures will also prohibit the use of onsite vehicles during transport of the transfer cask, negating the potential for a vehicle accident serving as the initiating event for a transformer fire. Moreover, even if a transformer mineral oil fire were to occur, its effect on the transfer cask during transport is bounded by the nonengulfing 7,570-L [2,000-gal] fire-loading analysis.

The potential for a fire within the CTF as the result of a cask transporter fuel spill was addressed in response to a Request for Additional Information.[7] To mitigate the potential for these postulated fire events, the CTF opening will be located at a higher elevation than the surrounding area so any fuel spilled will flow away from the Diablo Canyon ISFSI Facility. Moreover, administrative controls will prohibit any transient fuel sources other than the cask transporter from coming into close proximity of the CTF during transfer operations.

The emergency plan for the DCPP has been augmented to address the potential fire hazards that are uniquely associated with the proposed dry-storage area. The DCPP Emergency Plan also provides for the availability of a fire brigade and fire-fighting equipment and gear. The fire brigade is organized, operated, trained, and equipped in accordance with the requirements of 10 CFR Part 50. The equipment and gear will be stocked and maintained in accordance with 10 CFR Part 50. This emergency plan was reviewed and found to be acceptable by the staff (see Chapter 10 of this SER for additional details associated with this review).

The staff reviewed the information provided by the applicant pertaining to the Diablo Canyon ISFSI fire protection. The staff found the analysis acceptable because

- The HI-STORM 100 System has been evaluated for a bounding, hypothetical fully engulfing fire caused by a 189-L [50-gal] spill of diesel fuel and the evaluation, described in detail in the HI-STORM 100 System FSAR, was

---

[6]Womack, L.F. Letter (July 28) DIL–03–010, Enclosure 1 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[7]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Spent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

previously reviewed and found to be acceptable by the staff (U.S. Nuclear Regulatory Commission, 2002a,b).

- Based on the assessment of the potential fire hazards and the fire-protection measures at the Diablo Canyon ISFSI Facility, reasonable assurance that the HI-STORM 100 System will not be exposed to fires that exceed the design basis fire has been provided.

- The restricted area has been adequately described.

- Noncombustible and heat-resistant materials will be used wherever practical.

- Through design of the CTF and administrative procedures, combustible material (e.g., spill of diesel fuel from cask transporters) will be kept out of the canister transfer cell during canister transfer operations.

- The DCPP Emergency Plan provides for the availability of a fire brigade and fire-fighting equipment and gear.

- The information provided is also acceptable for use in other sections of the SAR to develop the design bases of the Diablo Canyon ISFSI Facility and perform additional safety analyses.

Staff review of the submitted information gave reasonable assurance that the Diablo Canyon ISFSI is adequately designed to protect SSCs important to safety from all postulated onsite fires and wildfires. Based on the foregoing evaluation, the requirements of 10 CFR §72.122(c) have been adequately satisfied.

Further evaluation of the effects of credible fires at the Diablo Canyon ISFSI is given in Chapter 15, "Accident Analysis," of this SER.

6.1.5.2 Explosion

The information presented in Section 2.2.1.2, "Hazards from Facilities and Ground Transportation;" Section 2.2.2.3, "Onsite Explosion Hazards;" Section 3.3.1.6, "Fire and Explosion Protection;" and Section 8.2.6, "Explosion," of the SAR (Pacific Gas and Electric Company, 2002a) and a Probabilistic Risk Assessment[8] in connection with the protection against potential onsite and offsite explosions have been reviewed for conformance with the following regulatory requirement.

- 10 CFR §72.122(c) requires that structures, systems, and components important to safety be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive

[8]Womack, L.F. Letter (July 28) DIL–03–010, Enclosure 2 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

U.S. Highway 101 passes approximately 14.4 km [9 mi] east of the proposed site. The Irish Hills separate the highway from the site. The proposed site is also approximately 16 km [10 mi] south of U.S. Highway 1. The nearest county roads in Clark Valley and See Canyon are 8 km [5 mi] from the proposed site. A county road, Avila Beach Drive, provides access to the PG&E DCPP private road system.

The Southern Pacific Transportation Company rail line runs adjacent to U.S. Highway 101 and, therefore, is located approximately 14.4 km [9 mi] from the proposed site. The Irish Hills shield the DCPP site from any potential explosion hazards that may be transported along these two routes. There is no spur track into the PG&E DCPP property.

Shipping lines near the proposed facility are approximately 32 km [20 mi] offshore. Since 1998, tanker traffic into either Port San Luis or Estero Bay has stopped. Some petroleum products and crude oil are currently stored at Estero Bay, approximately 16 km [10 mi] from the proposed site. Port San Luis Harbor is approximately 9.6 km [6 mi] south-southeast of the proposed Facility.

Based on Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978), the maximum probable hazardous solid cargo for a single highway truck is 23,000 kg [50,000 lb]; for a single railroad box car, approximately 60,000 kg [132,000 lb]; and for a ship, approximately 4,500,000 kg [10,000,000 lb]. No potential explosion hazards that exceed these limits within the prescribed safe distances for each (U.S. Nuclear Regulatory Commission, 1978, Figure 1) were identified.

The potential onsite explosion hazards that could affect the HI-TRAC 125 Transfer Cask or HI-STORM 100SA storage casks identified in the SAR (Pacific Gas and Electric Company, 2002a, Subsection 8.2.6) are as follows:

(1) Detonation of a transporter or onsite vehicle fuel tank
(2) Detonation of mineral oil from the DCPP Unit 2 main bank transformers
(3) Detonation of propane bottles transported past the ISFSI storage pad
(4) Detonation of acetylene bottles transported past the ISFSI storage pad
(5) Explosive decompression of a compressed gas cylinder
(6) Detonation of large stationary fuel tanks in the vicinity of the transport route
(7) Detonation of the bulk hydrogen facility
(8) Detonation of acetylene bottles stored on the east side of the cold machine shop

Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) recommends that the setback distance for a potential explosion hazard correspond to an air overpressure of 6.9 kPa [1 psi], below which the potential effects of explosion-generated missiles do not have to be evaluated. Alternatively, Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978)

states that if estimates of explosion rate, frequency of shipment, and exposure distance are made on a realistic or best-estimate basis, an exposure rate less than 10^{-7} per year is sufficiently low to screen out potential explosion hazards. If conservative estimates are used, an exposure rate less than 10^{-6} per year is sufficiently low.

The flash point for diesel fuel {52 °C [125 °F]} exceeds the maximum expected ambient temperature {40 °C [104 °F]} for the proposed site. As a result, an additional accident event sequence would be required to produce the conditions needed to ignite this particular type of fuel. No credible accidents capable of producing the conditions necessary to cause an explosion of the transporter diesel fuel were identified. The flash point of the mineral oil {135 °C [275 °F]} from the DCPP Unit 2 main bank transformers also exceeds the maximum expected ambient temperature. Although an electrical fault may occur within one of the transformers, the resulting rupture of the transformer case may cause the mineral oil to ignite and burn, but not explode. Therefore, the annual frequency of occurrence for this event was not considered in assessing the potential explosion hazard exposure for the HI-TRAC 125 Transfer Cask.[9]

The potential hazard arising from standard vehicle fuel tank explosions was assessed assuming an average tank capacity of 76 L [20 gal].[10] Two standard vehicle fuel tank explosions scenarios were addressed in the Probabilistic Risk Assessment.[11] The first scenario considered the annual frequency of occurrence for a parked vehicle fuel tank explosion, applicable to the HI-TRAC 125 Transfer Cask explosion hazard assessment. The second scenario evaluated the frequency of occurrence for a fuel tank explosion of vehicles passing within close proximity of the storage area (i.e., vehicles passing within the 53-m [175-ft] setback distance of the HI-STORM 100SA storage casks).

In addition to the standard vehicle fuel tank explosion hazard is the hazard posed by a fuel tanker truck that passes within 15.2 m [50 ft] of the storage casks on the north side of the proposed dry-storage area approximately six times per week. The setback distance calculated for the 3,028-L [800-gal] tanker truck,[12] based on an explosion overpressure limit of 6.9 kPa [1 psi], is 183 m [600 ft]. The fuel tanker truck passing within 15.2 m [50 ft] of the dry-storage area violates the required setback distance. A Probabilistic Risk Assessment[13] associated with the fuel tanker coming into close proximity of the storage area was submitted.

No onsite stationary fuel storage facilities pose an explosion hazard to the ISFSI storage area. It is estimated, however, that one acetylene bottle per year and one propane bottle per week will pass within the required setback distance of the ISFSI storage area. Therefore, the annual

---

[9]Womack, L.F. Letter (July 28) DIL–03–010, Enclosure 1 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[10]Ibid.

[11]Ibid.

[12]Ibid.

[13]Ibid.

frequency of occurrence for this particular explosion hazard was accounted for in the Probabilistic Risk Assessment.[14]

The potential effect of explosion-generated missiles originating from the explosive decompression of a compressed gas cylinder on the HI-TRAC 125 Transfer Cask and HI-STORM 100SA storage casks were shown to be inconsequential (Pacific Gas and Electric Company, 2002a, Subsection 8.2.6.2.2).

Three onsite stationary fuel tanks {i.e., 946 L [250 gal] propane, 7,571 L [2,000 gal] No. 2 diesel, and 11,356 L [3,000 gal] gasoline} could pose an explosion hazard to the HI-TRAC 125 Transfer Cask during transport of spent nuclear fuel to the CTF. These tanks are located beside the main plant road, approximately 366 m [1,200 ft] from the cask transport route at its nearest point. Given their distance from the transport route, however, it was shown (Pacific Gas and Electric Company, 2002a, Subsection 8.2.6.2.1) that an explosion of the propane and gasoline tanks would subject the HI-TRAC 125 Transfer Cask to an explosion overpressure less than 6.9 kPa [1 psi] (U.S. Nuclear Regulatory Commission, 1978). As a result, this particular explosion hazard does not have to be considered in the Probabilistic Risk Assessment.[15] Note that an explosion of the diesel fuel tank is considered to be an incredible event.

The bulk hydrogen facility is comprised of six individual tanks that represent a combined 8.5-m^3 [300-ft^3] storage capacity. This facility is located near the FHB/AB and is approximately 4.6 m [15 ft] from the HI-TRAC 125 Transfer Cask transport route. The hydrogen storage tanks are enclosed within a seismic-qualified vault on three sides. The open face of the vault is oriented toward the transfer cask transport route. Therefore, the annual frequency of occurrence for this event was considered in assessing the potential explosion hazard exposure for the HI-TRAC 125 transfer cask.[16]

According to Probabilistic Risk Assessment,[17] the potential detonation of seismically constrained acetylene bottles stored on the east side of the cold machine shop is not a credible event. The basis for this conclusion is that administrative procedures will prohibit transport of the transfer cask during inclement weather, significantly reducing the potential for lightning being the initiating event for an explosion. Moreover, administrative procedures will also prohibit the use of on-site vehicles during transport of the transfer cask, negating the potential for a vehicle accident being the initiating event for an acetylene bottle explosion. As a result, the occurrence of an explosion from this potential hazard does not have to be included in the total explosion frequency of occurrence for the HI-TRAC 125 Transfer Cask.

In summary, the potential explosion hazards that could affect the ISFSI storage area are transient in nature. These explosion hazards are 76-L [20-gal] standard vehicle fuel storage tanks, 3,028-L [800-gal] fuel tanker trucks, and propane and acetylene bottles that pass within

---

[14]Womack, L.F. Letter (July 28) DIL–03–010, Enclosure 1 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[15]Ibid.

[16]Ibid.

[17]Ibid.

close proximity of the storage area on a regular basis. The potential explosion hazards that could affect the transporter during transfer of the spent nuclear fuel from the FHB/AB to the CTF are stationary standard vehicle fuel storage tanks (parked in the lot adjacent to the FHB/AB) and the bulk hydrogen facility.

The staff reviewed the information provided by the applicant pertaining to the Diablo Canyon ISFSI explosion protection. The staff found the analysis acceptable because

- Descriptions of potential explosion sources are adequate.
- The credible explosion hazard sources are sufficient distances from important-to-safety SSCs, including the storage cask and transfer cask during spent fuel transport, to ensure the air overpressure 6.9-kPa [1-psi] limit is not exceeded.
- A Probabilistic Risk Assessment[18] was performed to demonstrate the following explosion hazards for the storage area that could exceed the 6.9-kPa [1-psi] air overpressure limit are not credible [i.e., the combined annual frequency of occurrence for these potential explosion hazards is below the 1×10^{-6} threshold defined by Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978)]: (1) 3,028-L [800-gal] tanker truck that routinely uses the transport route near the area to deliver fuel to the vehicle maintenance shop located approximately 610 m [2,000 ft] northeast of the area, (2) 76-L [20-gal] fuel tanks associated with standard vehicles routinely passing within 53 m [175 ft] of the area, and (3) transport of acetylene bottles along the transport route near the area for delivery to the vehicle maintenance shop.
- A Probabilistic Risk Assessment[19] was performed to demonstrate the following explosion hazards along the HI-TRAC 125 Transfer Cask transport route that could exceed the 6.9-kPa [1-psi] air overpressure limit are not credible [i.e., the combined annual frequency of occurrence for these potential explosion hazards is below the 1×10^{-6} threshold defined by Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978)]: (1) the bulk hydrogen tank facility and (2) standard 76-L [20-gal] fuel tank vehicles parked near the FHB/AB.

Staff review of the submitted information gave reasonable assurance that the Diablo Canyon ISFSI is adequately designed to protect SSCs important to safety from all postulated explosion hazards. Based on the foregoing evaluation, the requirements of 10 CFR §72.122(c) have been adequately satisfied.

---

[18]Womack, L.F. Letter DIL–03–010, Enclosure 2 (July 28) to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[19]Ibid.

6.2 Evaluation Findings

Thermal evaluation of the Diablo Canyon ISFSI Facility, as presented in the SAR, is based on the assumption that the HI-STORM 100 System will be used for storage and transporting the spent nuclear fuel to the Facility. Findings based on this review follow.

Sufficient information was provided by the applicant to demonstrate the Diablo Canyon ISFSI spent nuclear fuel specifications will not exceed the decay heat-removal capacities of the HI-STORM 100 System under normal, off-normal, and accident loading conditions. Therefore, the requirements of 10 CFR §72.122(h)(1) and §72.128(a) have been adequately demonstrated.

The temperature limits for normal operations are consistent with the approved Pacific Northwest National Laboratory cladding temperature limits for low-burnup spent nuclear fuels (Levy, et al., 1987). For high-burnup spent nuclear fuels, the temperature limits during normal operations are below the allowable limits based on the approved Holtec International creep model (Holtec International, 2002, Section 4.A). The short-term temperature limit of 570 °C [1,058 °F] for both low- and high-burnup spent nuclear fuel is consistent with the guidance provided in Interim Staff Guidance-15 (U.S. Nuclear Regulatory Commission, 2001). As a result, the requirements of 10 CFR §72.122(h)(1) have been adequately demonstrated.

Reliable data sources have been used to present temperatures and solar insolation at nearby sites. Data recorded by the DCPP onsite measurement program have also been presented. These data correlate well with data recorded at nearby sites. Therefore, the SAR shows that information on temperatures and solar insolation at the proposed site is acceptable and in compliance with 10 CFR 72.92(a). The temperatures and solar loads at the site are bounded by the HI-STORM 100 System design parameters.

The analytical methods, models, and calculations used to establish the thermal characteristics of the HI-STORM 100 System were previously reviewed and found to be acceptable by the staff (U.S. Nuclear Regulatory Commission, 2002a,b).

The SAR adequately describes the potential fire hazards for the Diablo Canyon ISFSI Facility storage area and HI-TRAC 125 Transfer Cask transport route. In addition, adequate descriptions of potential sources of accidental onsite and offsite explosions have been presented. Through design of the CTF and administrative procedures, combustible material (e.g., spill of diesel fuel from cask transporters) will be kept out of the canister transfer cell during canister transfer operations. A Probabilistic Risk Assessment[20] was used to determine whether or not potential fire and explosion hazards are credible. In summary, the SAR shows that the fire and explosion hazards at the site are acceptable and in compliance with the requirements of 10 CFR 72.122(c). Based on the assessment of the fire protection measures and the potential fire and explosion hazards at the site, there is reasonable assurance that the HI-STORM 100 System will not be exposed to fires or explosions that are beyond the design basis for the cask system.

---

[20]Womack, L.F. Letter DIL–03–010, Enclosure 2 (July 28) to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

6.3 References

Holtec International. *Topical Report on the HI-STAR/HI-STORM Thermal Model and Its Benchmarking with Full-Size Cask Test Data.* HI–992252, Revision 1. Marlton, NJ: Holtec International. 2000.

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System).* Vols I and II, Rev 1. HI–2002444. Docket 72-1014. Marlton, NJ: Holtec International. 2002.

Levy, I.S., B.A. Chin, E.P. Simonen, C.E. Beyer, E.R. Gilbert, and A.B. Jonson, Jr. *Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircaloy Clad Fuel Rods in Inert Gas.* PNL–6189. Richland, WA: Pacific Northwest National Laboratory. 1987.

Pacific Gas and Electric Company. *Diablo Canyon Power Plant Final Safety Analysis Report Update.* Rev 14. Avila Beach, CA: Pacific Gas and Electric Company. 2001.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Safety Analysis Report.* Amendment 1. Avila Beach, CA: Pacific Gas and Electric Company. 2002a.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: License Application.* Amendment 1. Avila Beach, CA: Pacific Gas and Electric Company. 2002b.

U.S. Nuclear Regulatory Commission. *Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants.* Regulatory Guide 1.91. Rev 1. Washington, DC: U.S. Nuclear Regulatory Commission. 1978.

U.S. Nuclear Regulatory Commission. 10 CFR Part 72 *Certificate of Compliance No. 1014, Amendment 1, for the HI-STORM 100 Cask System.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. July 15, 2002a.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report.* Amendment 1. Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. July 15, 2002b.

U.S. Nuclear Regulatory Commission. *Interim Staff Guidance-11: Cladding Considerations for the Transportation and Storage of Spent Fuel.* Rev. 2. Washington, DC: U.S. Nuclear Regulatory Commission. 2002c.

7 SHIELDING EVALUATION

7.1 Conduct of Review

The objective of this review was to determine if the shielding design features of the proposed Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) meet U.S. Nuclear Regulatory Commission (NRC) criteria for radiation protection to workers and to the public against direct radiation emanated by the materials to be stored during all operations at the ISFSI. The shielding evaluation includes a review of the information in Chapter 7, "Radiation Protection," of the Safety Analysis Report (SAR) (Pacific Gas and Electric Company, 2002) and relevant sections of Chapter 3, "Principal Design Criteria;" Chapter 4, "ISFSI Design;" and Chapter 8, "Accident Analyses," of the SAR.

Pacific Gas and Electric Company (PG&E) proposes to use the HI-STORM 100 System, which consists of interchangeable Multi-Purpose Canister (MPC) that contain the fuel, a storage overpack, and a transfer cask. The HI-TRAC transfer cask contains the MPCs during loading, unloading, and transfer operations. The system has been reviewed and approved by NRC under the general license provision of 10 CFR Part 72. Amendment 1 of the HI-STORM 100 System Certificate of Compliance (CoC) became effective on July 15, 2002 (U.S. Nuclear Regulatory Commission, 2002). Cask-specific information reported in HI-STORM 100 System Final Safety Analysis Report (FSAR), Revision 1 (Holtec International, 2002) was reviewed as part of the shielding evaluation of the Diablo Canyon ISFSI.

The shielding review considered how the information in the SAR addressed the following regulatory requirements:

- 10 CFR §72.24(b) requires that the SAR contains a description and discussion of the ISFSI structures with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

- 10 CFR §72.24(c)(3) requires that the SAR describe all information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all structures, systems, and components important to safety, in sufficient detail to support a finding that the ISFSI will satisfy the design bases with an adequate margin for safety.

- 10 CFR §72.24(e) requires that SAR describe the means of controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20, and for meeting the objective of maintaining exposures as low as reasonably achievable.

- 10 CFR §72.104(a) requires that during normal operations and anticipated occurrences, the annual dose equivalent to any real individual located beyond the controlled area be limited to 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid or 0.25 mSv (25 mrem) to any other critical organ.

- 10 CFR §72.106(b) requires that 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). The minimum distance from the spent fuel, waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters.

- 10 CFR §72.126(a)(6) requires that structures, systems, and components for which operations, maintenance, and required inspections may involve occupations designed, fabricated, located, shielded, controlled, and tested to control external and internal radiation exposures to personnel. This design must means to shield personnel from radiation exposure.

- 10 CFR §72.128(a) requires that spent fuel storage, and other systems that might contain or handle radioactive materials associated with spent fuel, must be designed to ensure adequate safety under normal and accident conditions.

- 10 CFR §20.1201(a) requires that the licensee shall control the occupational dose to individual adults, except for planned special exposures under §20.1206, to the following dose limits. (1) An annual limit, which is the more limiting of (i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or (ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or the lens of the eye being equal to 50 rems (0.5 Sv). (2) The annual limits to the lens of the eye, to the skin of the whole body, and to the skin of the extremities, (i) A lens dose equivalent of 15 rems (0.15 Sv), and (ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

- 10 CFR §20.1301(a) requires that each licensee shall conduct operations so that (1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under §35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with §20.2003, and (2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with §35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

- 10 CFR §20.1302(b) requires that each licensee shall conduct operations so that (1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under §35.75, from voluntary participation in medical

research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with §20.2003, and (2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with §35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

7.1.1 Contained Radiation Source

The gamma and neutron source specifications are presented in Section 10.2 of the Diablo Canyon ISFSI SAR. The sources of gamma and neutron radiation are the intact spent nuclear fuel assemblies, damaged fuel assemblies, fuel debris and nonfuel hardware stored in the HI-STORM 100 System. Only fuel and associated hardware irradiated at Diablo Canyon Power Plant (DCPP) Units 1 and 2 will be stored at the ISFSI. The spent nuclear fuel assemblies to be stored at the proposed ISFSI consist of two zircaloy-clad Pressurized Water Reactor (PWR) designs. The two designs are the Westinghouse LOPAR 17 × 17, and the Westinghouse VANTAGE 5, 17 × 17, assemblies. Their characteristics are described in the Diablo Canyon ISFSI SAR Tables 3.1-1 and 3.1-2. Initial U-235 enrichment of the fuel to be stored at the proposed ISFSI will be limited to 5 percent. Average burnup will be limited to a maximum of 58,000 MWd/MTU and cooling time must exceed 5 years. Burnable poison rod assemblies, thimble plug devices, and rod cluster assemblies that will also contribute to the radiation source are described in Section 7.2 of the SAR. The gamma source term is composed of three distinct components. The first gamma source term is decay of fission products in the active fuel region. The second gamma source term is Co-60 activity of the steel structural material in the fuel element above and below the active fuel region. The third gamma source term is from (neutron, gamma) reactions.

For the ISFSI storage pads, PG&E conducted a shielding analysis that estimated bounding dose rates from direct radiation by assuming overpacks with identical multi-purpose canister (MPC)–32s are completely loaded with fuel assemblies having burnup of 32,500 MWd/MTU and 5-year cooling times. A Babcock and Wilcox 15 × 15 fuel assembly, a design-basis assembly for the HI-STORM 100 System, was used in the ISFSI shielding analyses. Additional credit was taken in the dose analyses for longer cooling times because the casks are to be placed at the ISFSI at the rate of only eight per year.

The PG&E shielding analysis of the transfer cask was performed for the MPC–24 using burnup and cooling time of 55,000 MWd/MTU and 12 years. These dose rates bound those for other MPC, burnup, and cooling time combinations. Gamma and neutron source terms were generated using the SAS2H (Hermann and Parks, 1995) and ORIGEN-S sequences from the SCALE 4.3 system (Hermann and Westfall, 1995).

The Co-60 gamma source for nonfuel hardware was evaluated. The Co-59 impurity level was assumed to be 0.8 g/kg [8×10^{-4} lb/lb] or 800 ppm in stainless steel and 4.7 g/kg [4.7×10^{-3} lb/lb] or 4,700 ppm in inconel. Burnup of 40,000 MWd/MTU and cooling time of 12 years were assumed for a design-basis burnable poison rod assembly. It was assumed that all overpacks were filled with the design-basis burnable poison rod assemblies.

For the shielding analysis, it was assumed that overpacks are placed on flat ground, so shielding-design features included only overpack and transfer cask and no credit was taken for any additional radiation shielding provided by a hilly terrain surrounding ISFSI storage pads.

The dose-versus-distance analysis was conducted for a single cask loaded with MPC-32 with design-basis burnup and cooling times using MCNP-4A Code (Briesmeister, 1993). The dose rates of direct radiation emanating from 140 casks were calculated at distances perpendicular to the long side of the ISFSI and assuming that eight overpacks are loaded per year. Radiation from 28 casks located along the long side of the ISFSI was accounted for as unshielded, and radiation from the remaining casks was accounted for as partly shielded by the casks located between those casks and dose locations.

The staff evaluated analyses of the bounding radiation source terms for the Diablo Canyon ISFSI. The staff found the description of radiation sources and calculation methods of the shielding analysis to be consistent with the information provided in HI-STORM 100 System FSAR Revision 1 (Holtec International, 2002). The HI-STORM 100 System was approved by NRC for use under general license provisions of 10 CFR Part 72 (U.S. Nuclear Regulatory Commission, 2002). The staff found that combinations of enrichment, burnup, and cooling time for DCPP spent nuclear fuel are conservative and determined correctly.

7.1.2 Storage and Transfer Systems

7.1.2.1 Design Criteria

The design criteria for the proposed Diablo Canyon ISFSI are the regulatory dose limit requirements delineated in 10 CFR Part 20, §72.104(a), and §72.106(b). The Diablo Canyon SAR specifies the shielding design criteria in Sections 3.3.1.1.2, 3.3.1.1.3, and 3.3.1.5.2. The SAR sections reference Chapters 1, 2, and 5 of the HI-STORM 100 System FSAR, Revision 1 (Holtec International, 2002). The HI-STORM 100 System storage cask is designed to limit maximum average surface dose radially, on top, and in areas adjacent to the inlet and exit vents to 600 μSv/hr [60 mrem/hr]. The transfer cask is designed to maintain personnel doses as low as reasonably achievable (ALARA). The staff finds the use of these design criteria to be appropriate. These design criteria provide reasonable assurance that the ISFSI will meet the dose limits delineated in 10 CFR Part 20, §72.104(a), and §72.106(b). The Diablo Canyon ISFSI will provide adequate radiological safety based on the use of suitable shielding for radiation protection in accordance with 10 CFR §72.128(a)(2).

7.1.2.2 Design Features

The Diablo Canyon ISFSI system is designed to provide both gamma and neutron shielding for all fuel loading, transfer, and storage conditions. The system shielding design features are described in Section 7.3 of the SAR. Facility design features that ensure that dose rates are ALARA include

- There are no radioactive systems at the ISFSI storage pads other than overpacks containing MPCs.
- The MPC shielding is composed of 1.3-cm-[0.5-in-] thick steel canister with a 6.4-cm-[2.5–in-] thick baseplate and a 24.1-cm-[9.5-in-] thick steel lid for the MPC–24 or a 25.4-cm [10-in] lid for the MPC–68.
- The MPCs are heavily shielded by the overpack that consists of 67.95-cm-[26.75-in-] thick concrete encased in inner and outer steel cases with a total

thickness of 6.99 cm [2.75 in]. The top of the overpack is shielded by 26.7-cm- [10.5-in-] thick concrete encased in inner and outer steel cases with a total thickness of 13.3 cm [5.25 in]. The bottom of the overpack is shielded by the pedestal shield and baseplate with a total of 17.8 cm [7 in] of steel and 43.2 cm [17 in] of concrete.

- The 1.13×10^5-kg [125-ton] HI-TRAC Transfer Cask radial shield is composed of a 4.45-cm- [1.75-in-] thick steel layer, an 11.4-cm-[4.5-in-] thick lead layer, and a 13.61-cm- [5.36-in-] thick water jacket. Its top lid consists of a 8.26-cm- [3.25-in-] thick Holtite-A encased in steel plates with a total thickness of 3.8 cm [1.5 in].

- The Canister Transfer Building positions the overpack below ground during the loading operations. This position reduces dose rates during loading operations.

The staff evaluated the Diablo Canyon ISFSI shielding design features and found them acceptable. The Diablo Canyon ISFSI SAR provided reasonable assurance the shielding design features of the storage and transfer systems can meet the requirements in 10 CFR Part 20, §72.104(a), and §72.106(b). The staff evaluated the radiation protection design features of the Diablo Canyon ISFSI in Chapter 11 of this Safety Evaluation Report (SER).

7.1.3 Shielding Composition and Details

7.1.3.1 Composition and Material Properties

The composition of the materials used in the shielding analysis are presented in Sections 3.3.1.5.2 and 7.3.2. These sections reference HI-STORM 100 System FSAR Revision 1 (Holtec International, 2002). The primary shielding is from the concrete and steel in the HI-STORM 100 System storage cask during storage and from the steel, lead, water, and neutron shield in the HI-TRAC 125 Transfer Cask during transfer operations.

The staff found the description of the shielding composition to be sufficient to meet the requirements of 10 CFR §72.24(b) and §72.24(c) by describing the design, the system shielding composition, and materials important to safety. This description is sufficiently detailed for evaluation of shielding effectiveness in reducing the dose rates around the Diablo Canyon ISFSI to within regulatory limits.

7.1.3.2 Shielding Details

The shielding details are described in Section 7.3.1 of the SAR. The HI-STORM 100 System storage casks will be stored on seven concrete pads in 4 × 5 arrays of casks. The total Diablo Canyon ISFSI storage capacity is 140 casks, which will be positioned on a 5.18 m [17-ft], center-to-center pitch.

The shielding models included streaming paths for the radial steel fins and pocket trunnions of the HI-TRAC 125 Transfer Cask, drain and vent ports in the MPC, annulus between the MPC and the concrete cask, annulus between the MPC and the transfer cask, and labyrinthine air inlet and outlet passages. Gamma and neutron doses were calculated using three-dimensional models with MCNP–4A Code (Briesmeister, 1993).

The staff evaluated the shielding details and found that the description satisfies the requirements of 10 CFR §72.126(a)(6) and provides reasonable assurance that the radiation protection systems were adequately modeled in the shielding analysis.

7.1.4 Analysis of Shielding Effectiveness

7.1.4.1 Computational Methods and Data

The computational methods and data used to analyze shielding effectiveness in reducing the dose rates at the ISFSI are presented in Section 7.3.2 of the SAR and in the HI-STORM 100 System FSAR Revision 1 (Holtec International, 2002). Analyses were conducted to determine the surface and 1-m [3.28-ft] dose rates for storage and transfer casks and dose rates at the restricted area fence, the makeup water facility, the power plant, the controlled-area boundary, and the nearest resident location.

The shielding analysis of the HI-STORM 100 System casks was performed using MCNP-4A Code (Briesmeister, 1993). The MCNP Code is a general-purpose, continuous-energy, generalized geometry, time-dependent, coupled neutron-photon-electron Monte Carlo transport code system. The code system is able to model the complex surfaces associated with the storage casks. For neutrons, continuous cross-sectional data are accounted for in all reactions in any given cross-section evaluation such as ENDF/B-VI (Los Alamos National Laboratory, 1994). Thermal neutrons are described by both the free gas and S (alpha beta) models. For photons, the code accounts for incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation and bremsstrahlung.

The gamma flux-to-dose conversion factors used in the Diablo Canyon ISFSI SAR were from American National Standards Institute/ American Nuclear Society (ANSI/ANS)–6.1.1 (American Nuclear Society Standards Committee Working Group, 1977). The computer code and the ANSI/ANS–6.1.1 flux-to-dose conversion factors used for shielding analysis are considered acceptable by the staff for use in the shielding evaluations.

7.1.4.2 Dose Rate Estimates

The estimates of dose rates and annual doses caused by direct neutron and gamma radiations at various on-site and off-site locations are presented in Sections 7.3.2.1, 7.3.2.2, 7.5.1, 7.5.3, and 7.5.4 of the SAR.

The HI-TRAC 125 Transfer Cask is designed to reduce dose rates from direct radiation emanated from a loaded MPC to levels that are ALARA. The design-basis MPC for the transfer cask analysis is the MPC–24 loaded with a fuel having a burnup of 55,000 MWd/MTU and 12-year cooling time period. The contact surface dose rate for the HI-STAR transfer cask was estimated to be approximately 3,893 μSv/hr [389.3 mrem/hr] outside the lid in its center. The transfer of MPC from the transfer cask to the overpack will take place outside the Fuel Handling Building/Auxiliary Building (FHB/AB) at the Cask Transfer Facility (CTF). The impact on the offsite dose of the loaded MPC while it is transported inside the transfer cask from the plant to the CTF was considered in the analysis.

The design-basis MPC for the storage cask analysis is the MPC–32 with a burnup of 32,500 MWd/MTU and 5-year cooling time period. The average contact surface dose rate for the HI-STORM 100 System storage cask (side-dose value) was estimated to be approximately 348 μSv/hr [34.8 mrem/hr] at the midplane of the overpack. To evaluate onsite and offsite dose rates from direct and scattered radiation emanated from the spent nuclear fuel stored at the ISFSI, the applicant did not build an integrated MCNP model of the 7 ISFSI storage pads with 140 casks and each loaded with a design-basis fuel. Instead, the applicant employed an approach described in Section 5.4.3 of the HI-STORM 100 System FSAR (Holtec International, 2002). In this approach, the dose rates were estimated along the line perpendicular to the long side of the array of casks situated at the storage pad. In the first stage of analysis, the MCNP binary surface source file was generated for a single cask. This surface source was based on the real configuration of design-basis fuel in the cask, and those data were used in the second stage of the analysis. In the second stage of analysis, the contribution of the cask positioned in the second row of the array of casks and blocked from direct view line to the dose point was estimated. Then, the total direct and scattered dose rates were summed and dose rates at various distances were estimated for several small arrays of casks. The side-dose rate values, top-dose rate values, and in-air scattering of radiation (skyshine) dose rates from both side and top radiation components were taken into accounted. Results of the calculations employing this approach are presented in Section 5.4.3 of the HI-STORM 100 System FSAR and are demonstrative in nature and not site specific. It is important to highlight that the contribution of the back row of casks to the total dose at a point 300 m [984 ft] away was estimated as 16 percent of the total dose. The dose fraction caused by neutrons from one cask at a point 200 m [656 ft] away from the cask was estimated at 7 percent of the total dose.

In addition to this approach, the applicant applied credit for the additional cooling times for the stored fuel based on its ISFSI loading plan. PG&E assumed in its dose rate analysis that eight overpacks per year will be loaded until all the ISFSI pads are filled to capacity. PG&E presented the reduction of gamma and neutron source intensity with time because of the longer cooling periods of design-basis fuel. Using this methodology instead of a full-scale 140-cask array modeling, the applicant calculated a site-boundary dose rate at a point located 427 m [1,400 ft] away from the ISFSI storage pad as 0.027 μSv/hr [0.0027 mrem/hr] from direct and scattered radiation exposure for 140 casks stored in a 5 × 28 array as in Chapter 7 of the SAR. This rate corresponds to a 56-μSv [5.6-mrem] annual dose for a 2,080-hr/yr occupancy rate for a hypothetical person located at the site boundary. The applicant estimated an annual direct and scattered radiation dose to the nearest resident located 2,414 m [1.5mi] away from the ISFSI as 0.0035 μSv [0.00035 mrem]. These dose values are less than the 250 μSv/yr [25 mrem/yr] whole-body dose limit specified in 10 CFR §72.104(a).

The staff evaluated the approach taken by the applicant in direct radiation dose analysis and found that the approach provides a reasonable assurance that onsite and offsite dose rates at various distances from the array of casks will be estimated correctly. No off-normal events or accidents resulting in a loss of radiation shielding were identified in the SAR. Therefore, it was concluded in the applicant analyses that dose rates at the controlled-area boundary and on site would not be affected by the minor damage to the transfer or storage casks.

The staff evaluated the Diablo Canyon ISFSI SAR shielding calculations and found them to be acceptable. The dose rates, at the on-site and off-site locations were found to be below the limits specified in 10 CFR §20.1201, §20.1301, §20.1302, and §72.104(a). The applicant description, combined with a sample input file in the HI-STORM 100 System FSAR, Revision 1

(Holtec International, 2002), provides reasonable assurance that the ISFSI shielding was adequately evaluated. The applicant analysis demonstrates that no credible accident will cause a significant increase in public or personnel dose rates from direct radiation. This provides reasonable assurance that during accident conditions, dose rates from direct radiation will be below limits specified in 10 CFR §72.106(b).

Chapter 11 of this SER discusses the overall offsite dose rates from the Diablo Canyon ISFSI estimated from the combined radiation exposure to the direct radiation, scattered radiation, and potential radioactive effluents. The staff has reasonable assurance that compliance with 10 CFR §20.1201, §20.1301, §20.1302, §72.104(a), and §72.106(b) can be achieved by the applicant by the means of its radiation protection design and radiological protection program described in the SAR.

7.1.5 Confirmatory Calculations

The staff examined the proposed contents listed in Tables 7.2.1 through 7.2.4 of the SAR. The staff performed independent calculations of the bounding source terms for the stored fuel at the proposed Diablo Canyon ISFSI. Neutron and gamma source terms, as well as the radionuclide inventory, were generated by the ORIGEN-ARP code (Bowman, 2000) for various cooling periods. These calculations provided reasonable assurance that design-basis gamma and neutron source terms for MPC–32 are acceptable for the shielding analyses. The staff performed an independent calculation of the dose rates that could be expected around the storage casks and at the edge of the Diablo Canyon ISFSI Facility controlled area. The staff used the MCNP–4C2 Code, cross-sectional data supplied with the code distribution (Briesmeister, 2000), and gamma flux-to-dose conversion factors from ANSI/ANS 6.1.1 (American Nuclear Society Standards Committee Working Group, 1977).

The staff analyses were conducted in three stages. First, a single cask bounding surface source was determined. Second, a dose rate from the surface source was calculated at various distances. Third, the dose rates that would result from the maximum number of 140 casks planned to be stored at the ISFSI Facility were calculated. A back-row shadow factor reported by PG&E was confirmed in separate calculations. Variable cask fuel cooling periods and back-row shade factors were taken into account in the total on-site and off-site dose calculations.

The annual doses were calculated for an individual located on the nearest boundary of the ISFSI Facility controlled area and for the resident nearest to the ISFSI Facility. These calculations confirmed the on-site dose rates calculated by the applicant and also confirmed that the off-site dose rates would be less than the 0.25-mSv/yr [25-mrem/yr] whole-body dose allowable to a member of the public as required by 10 CFR §72.104. Based on these confirmatory calculations, the staff concluded that the applicant's shielding analysis is acceptable.

7.2 Evaluation Findings

Evaluation of shielding at the Diablo Canyon ISFSI assumed that only the HI-STORM 100 System will be used. The staff made the following findings regarding the shielding evaluation of the Diablo Canyon ISFSI.

- The design of the shielding system of the Diablo Canyon ISFSI satisfies the design criteria for radiological protection of 10 CFR §72.126(a)(6).

- The design of the Diablo Canyon ISFSI provides acceptable means for controlling occupation radiation exposures within the limits given in 10 CFR §20.1201 and for meeting the objective of maintaining exposures ALARA, in compliance with 10 CFR §72.24(e).

- The design of the Diablo Canyon ISFSI provides acceptable means for controlling exposures of the public to direct and scattered radiation within the limits given in 10 CFR §72.104.

- The design of the Diablo Canyon ISFSI provides suitable shielding for radioactive protection during normal and accident conditions in compliance with 10 CFR §72.128(a)(2).

7.3 References

American Nuclear Society Standards Committee Working Group. *Neutron and Gamma Ray Flux-to-Dose-Rate Factors.* ANSI/ANS 6.1.1-1977. Washington, DC: American National Standards Institute. 1977.

Bowman, S.M. and L.C Leal. *ORIGEN-ARP: Automatic Rapid Process for Spent Fuel Depletion, Decay, and Source Term Analysis.* NUREG/CR–0200, Rev 6, (ORNL/NUREG/CSD–2/V1/R6). Oak Ridge, TN: Oak Ridge National Laboratory. 2000.

Briesmeister, J.F., ed. *MCNP—A General Monte Carlo N-Particle Transport Code, Version 4A.* LA-12625-M. Los Alamos, NM: Los Alamos National Laboratory. 1993.

Briesmeister, J.F. *MCNP—A General Monte Carlo N-Particle Transport Code, Version 4C.* LA–13709–M. Los Alamos, NM: Los Alamos National Laboratory. April 2000.

Hermann, O.W. and C.V. Parks. *SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module.* NUREG/CR–0200, Rev 5, (ORNL/NUREG/CSD–2/V2/R5). Oak Ridge, TN: Oak Ridge National Laboratory. 1995.

Hermann, O.W. and R.M. Westfall. *ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms.* NUREG/CR–0200, Rev. 5, (ORNL/NUREG/CSD–2/V2/R5). Oak Ridge, TN: Oak Ridge National Laboratory. 1995.

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System).* HI–2002444, Rev. 1. Docket 72-1014. 2002.

Los Alamos National Laboratory. *ENDF/B-V1, Data for MCNP.* LA–12891. Los Alamos, NM: Los Alamos National Laboratory. 1994.

Pacific Gas and Electric Company. *Diablo Canyon ISFSI Safety Analysis Report, Amendment 1, Pacific Gas and Electric Company.* Docket No. 72-26. Avila Beach, CA: Pacific Gas and Electric Company. October 2002.

U.S. Nuclear Regulatory Commission. *10 CFR Part 72 Certificate of Compliance No.1014, Amendment No.1, for the Holtec International HI-STORM 100 Cask System.* Docket 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. July 15, 2002.

8 CRITICALITY EVALUATION

8.1 Conduct of Review

The review of criticality evaluation included Chapter 3, "Principal Design Criteria;" and Chapter 4, "ISFSI Design" of the Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) Safety Analysis Report (SAR) (Pacific Gas and Electric Company, 2002). The criticality review is to ensure that the stored materials remain subcritical under normal, off-normal, and accident conditions during all operations, transfers, and storage at the proposed Diablo Canyon ISFSI. This review considers how the information in the SAR addresses the following regulatory requirements:

- 10 CFR §72.40(a)(13) requires that there is reasonable assurance that: (i) The activities authorized by the license can be conducted without endangering the health and safety of the public and (ii) these activities will be conducted in compliance with the applicable regulations of this chapter.

- 10 CFR §72.124(a) requires that spent fuel handling, packaging, transfer, and storage systems for the radioactive materials are designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes must occur in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions.

- 10 CFR §72.124(b) requires that when practicable, the design of an ISFSI must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design must provide for positive means of verifying their continued efficacy. For dry spent fuel storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of the neutron absorbing materials cannot occur over the life of the facility.

- 10 CFR §72.124(c) requires that a criticality monitoring system shall be maintained in each area where special nuclear material is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration under a license issued under this subpart is not required.

The applicant proposes to use the HI-STORM 100 System. This cask system consists of (1) interchangeable Multi-Purpose Canister (MPC) that contain the fuel; (2) a storage overpack that contains the MPC during storage; and (3) a transfer cask that contains the MPC during loading, unloading, and transfer operations. The system has been reviewed and approved by U.S. Nuclear Regulatory Commission (NRC) under the general license provision of

10 CFR Part 72. Amendment 1 of the HI-STORM 100 System Certificate of Compliance (CoC), which became effective on July 15, 2002 (U.S. Nuclear Regulatory Commission, 2002a).

8.1.1 Criticality Design Criteria and Features

This section evaluates whether the proposed criticality safety design criteria and features will maintain the stored materials in a subcritical configuration. The Diablo Canyon ISFSI conditions for criticality safety are based on acceptance criteria outlined in NUREG–1567, Chapter 8, "Criticality Evaluation" (U.S. Nuclear Regulatory Commission, 2000). The Diablo Canyon ISFSI design criteria and features are described in SAR, the Sections 3.3.1.4, 3.2, and 3.4. Section 4.2.3.3.5, "Criticality Design," addresses criticality safety of the HI-STORM 100 System. The applicant did not rely on the use of burnup credit or fuel-related burnable neutron absorbers for the criticality safety analysis. The applicant took no more than 75 percent of B-10 isotope credit for fixed neutron absorbers in the analysis.

8.1.1.1 Criticality Design Criteria

The description of the design criterion for criticality safety in SAR Section 3.3.1.4, "Nuclear Criticality Safety," is clearly identified and adequately described. The design criterion for criticality safety is that the effective multiplication factor, k_{eff}, including statistical biases and uncertainties, shall not exceed 0.95 during all credible normal, off-normal, and accident conditions and events. The staff reviewed the proposed design criterion of the ISFSI and the proposed HI-STORM 100 System design criteria to ensure consistency with the ISFSI.

The proposed cask system, the HI-STORM 100 System, is autonomous and provides a subcritical configuration of stored materials independent of any other ISFSI structures or components. The design criterion for criticality safety is consistent with the 10 CFR §72.124(a) requirement that at least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions must occur before an accidental criticality (defined as exceeding k_{eff} of 0.95 at a 95-percent confidence level) is possible.

The staff found that the proposed design criterion will meet the double contingency requirements of 10 CFR §72.124(a).

8.1.1.2 Features

The Diablo Canyon ISFSI criticality safety features are described in SAR Section 4.2.3.3.5, "Criticality Design," which focuses on the HI-STORM 100 System proposed for use at the Diablo Canyon ISFSI. This cask system maintains the stored materials in a subcritical configuration independent of the ISFSI design. The cask system is NRC certified and is described in Chapter 6, "Criticality Evaluation," of the HI-STORM 100 System Final Safety Analysis Report (FSAR) (Holtec International, 2002). For criticality prevention, the cask system relies on the MPC, which provides the confinement system for the stored fuel. At the Diablo Canyon ISFSI, the fuel will be dry and sealed within a welded MPC. Thus, there are no credible accidents in which water could enter the MPC during transportation inside the HI-TRAC 125 Transfer Cask, reloading into the HI-STORM 100 System storage overpack, and storage on the pad inside the storage overpack.

There are seven types of MPCs: the MPC–24, MPC–24E, MPC–24EF, MPC–32, MPC–68, MPC–68F, and MPC–68FF. A loaded MPC is stored within the storage overpack in a vertical orientation. The criticality safety features are the inherent geometry of the fuel basket designs within the canisters that provides sufficient separation of stored fuel assemblies and permanent neutron-absorbing Boral panels fixed to the fuel cell walls.

The criticality monitoring system requirements of 10 CFR §72.124(c) are being addressed within a pending exemption request to 10 CFR §70.24 and §50.68, which is being administered under the 10 CFR Part 50 license for the DCPP.

The staff found that the design features important to nuclear criticality safety are clearly identified and adequately described, that the stored material will be maintained in a subcritical configuration, and that the design-basis, off-normal, and postulated accident events will not have an adverse effect on the design features important to criticality safety. Therefore, the staff conclude that the design features meet the requirements of 10 CFR §72.124(b) and §72.40(a)(13).

8.1.2 Stored Material Specifications

The proposed stored materials specifications are described in Section 3.1.1 of the Diablo Canyon ISFSI SAR. The materials include intact Diablo Canyon Power Plant (DCPP) fuel assemblies, damaged fuel assemblies, fuel debris, and nonfuel hardware approved for storage in the HI-STORM 100 System. Appendix B of CoC No. 1014, Amendment No. 1 (U.S. Nuclear Regulatory Commission, 2002a) describes materials approved for storage in the system.

The staff found that the materials proposed for storage are bounded by the approved contents for the HI-STORM 100 System and the material specifications are adequate to ensure that the stored materials will be maintained subcritical. The staff finds that the proposed material specifications are adequate to ensure that the contents will be maintained subcritical and that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes must occur in the conditions essential to nuclear criticality safety, in compliance with 10 CFR §72.124(a).

8.1.3 Analytical Means

The staff reviewed the analytical means used by PG&E to demonstrate that the materials stored in the ISFSI will remain subcritical. SAR Section 4.2.3.3.5, "Criticality Design," contains relevant information.

8.1.3.1 Model Configuration

The storage cask model configuration was reviewed and approved by the staff during the certification process of the HI-STORM 100 System certification process. Results are documented in the HI-STORM 100 System, Amendment No. 1, Safety Evaluation Report (SER) (U.S. Nuclear Regulatory Commission, 2002b). The applicant assumed in the cask system analysis that fresh fuel with a maximum possible enrichment is stored in a configuration that yields maximum reactivity and is flooded with fresh water at various densities. Nonfuel hardware, single cask, and arrays of storage casks were also considered in the analysis. No

additional modeling was conducted for the ISFSI because there were no site-specific conditions that would affect the generic criticality safety analysis of the HI-STORM 100 System.

8.1.3.2 Material Properties

The material properties are the same as assumed in the Holtec International HI-STORM 100 System FSAR (Holtec International, 2002), as amended by Holtec LAR 1014-1. Material properties were reviewed and approved by the staff during the certification process, and review results were documented in the HI-STORM 100 System Amendment No. 1 SER (U.S. Nuclear Regulatory Commission, 2002b).

8.1.4 Applicant Criticality Analysis

The staff found that the applicant addressed the most reactive configurations and conditions in the cask system analysis. The results of the analysis are documented in the HI-STORM 100 System FSAR Revision 1 and a brief description of the results is presented in SAR Section 4.2.3.3.5, "Criticality Design." Staff agreed that no additional criticality analysis is necessary for the Diablo Canyon ISFSI.

8.1.4.1 Computer Program

The applicant's principal criticality analysis code was Monte Carlo N-Particle (MCNP) 4A, a three-dimensional, continuous-energy, Monte Carlo N-Particle Code. MCNP4A and the KENO Va computer code, which was employed for verification purposes, are described in the HI-STORM 100 System FSAR Revision 1. The NRC accepted use of both codes for criticality analyses. No additional criticality codes are necessary for analysis of criticality at the proposed Diablo Canyon ISFSI.

8.1.4.2 Multiplication Factor

All results of the applicant analyses for all proposed fuel loadings yielded values for k_{eff}, including all biases and uncertainties, less than the staff acceptance value of 0.95 for normal, off-normal, and accident conditions. These results are discussed in the HI-STORM 100 System FSAR, Revision 1. The HI-STORM 100 System for cask transfer, the HI-TRAC 125 Transfer Cask with lead overpack, was analyzed in a flooded condition for loading and unloading operations. The flooded condition represents the limiting case with highest reactivity. No additional calculations were performed for the ISFSI.

8.1.4.3 Benchmark Comparisons

The applicant relied on the benchmark analysis in the HI-STORM 100 System FSAR, Revision 1 where criticality benchmark experiments were discussed. The value of bias correction used for k_{eff} was 0.0021 with an uncertainty of 0.0006.

8.1.4.4 Independent Criticality Analysis

No confirmatory independent calculations were performed because no additional criticality calculations are necessary for the ISFSI.

8.2 Evaluation Findings

Based on a review of the SAR and the presentations and information supplied by PG&E, the staff finds, with reasonable assurance, that

- The design, procedures, and materials to be stored at the proposed Diablo Canyon ISFSI provide reasonable assurance that the activities authorized by the license can be conducted without endangering the health and safety of the public in compliance with 10 CFR §72.40(a)(13).

- The design and proposed use of the Diablo Canyon ISFSI handling, packaging, transfer, and storage systems for the radioactive materials to be stored reasonably ensure that the materials will remain subcritical and, that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes must occur in the conditions essential to nuclear criticality safety. The revised Diablo Canyon ISFSI SAR analyses and confirmatory analysis by the NRC adequately show that acceptable margins of safety will be maintained in the nuclear criticality parameters commensurate with uncertainties in the data and methods used in calculations, and demonstrated safety for the handling, packaging, transfer, and storage during normal, off-normal, and accident conditions in compliance with 10 CFR §72.124(a) and §72.124(b).

8.3 References

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System).* HI–2002444, Rev. 1. Docket 72-1014. Marlton, NJ: Holtec International. 2002.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation. Safety Analysis Report, Amendment 1.* Docket No. 72-26. Avila Beach, CA: Pacific Gas and Electric Company. 2002.

U.S. Nuclear Regulatory Commission. *10 CFR Part 72 Certificate of Compliance No.1014, Amendment No.1, for the Holtec International HI-STORM 100 Cask System.* Docket 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2002a.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Amendment No. 1 Safety Evaluation Report.* Docket 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2002b.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Spent Fuel Dry Storage Facilities.* NUREG–1567. Washington, DC: U.S. Nuclear Regulatory Commission. 2000.

9 CONFINEMENT EVALUATION

9.1 Conduct of Review

The staff reviewed the confinement evaluation presented in the Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) Safety Analysis Report (SAR) (Pacific Gas and Electric Company, 2002) and the amended HI-STORM 100 System Final Safety Analysis Report (FSAR) (Holtec International, 2002).

The Diablo Canyon ISFSI Facility will use the HI-STORM 100 System, which has been approved by U.S. Nuclear Regulatory Commission for use under the general license provisions of 10 CFR Part 72. The design specifications for the confinement function of the HI-STORM 100 System are addressed in Chapter 7 of the amended HI-STORM 100 System FSAR (Holtec International, 2002).

Based on the statements in the amended HI-STORM 100 System FSAR (Holtec International, 2002), Pacific Gas and Electric Company conducted an analysis of a hypothetical radiological release. The confinement evaluation submitted by the applicant relies on the analyses performed by Holtec International to demonstrate compliance with 10 CFR Part 72 and includes a discussion of radiological release calculations and an evaluation of stored material degradation. Information about chemical composition and mechanical properties of materials for construction of critical cask components is provided in the amended HI-STORM 100 System FSAR (Holtec International, 2002).

This review was conducted in accordance with the guidance presented in Chapter 9, "Standard Review Plan for Spent Fuel Dry Storage Facilities," of NUREG–1567, except that no independent confirmatory calculations were performed for this review (U.S. Nuclear Regulatory Commission, 2000). The review focused on analyses and results presented and referenced by the applicant in the Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2002). The referenced analyses include those performed by Holtec International (2002) for the HI-STORM 100 System proposed for use at the Diablo Canyon ISFSI.

9.1.1 Radionuclide Confinement Analysis

The application was reviewed for identification of the quantity of radionuclides that hypothetically could be released during normal, off-normal, and accident conditions, including design-basis accidents. The staff reviewed Sections 3.3.1.2, 3.3.1.5, 3.3.1.7, 4.2.3.2, 6.1, 8.1.3, and 8.2.7 and Chapter 7 of the Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2002), Section 1.5 and Chapter 4, and Chapter 7 of the HI-STORM 100 System FSAR (Holtec International, 2002). The information presented has been reviewed for conformance with the following regulatory requirements:

- 10 CFR §72.24(l)(1) requires a description of the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal operations and expected operational occurrences. The description must identify the design objectives and the means to be used for keeping levels of radioactive material in effluents to the environment as low as is reasonably achievable and within the exposure limits stated in §72.104. The

description must include an estimate of the quantity of each of the principal radionuclides expected to be released annually to the environment in liquid and gaseous effluents produced during normal ISFSI operations.

- 10 CFR §72.44(c)(1)(i) requires that each license issued under this part include technical specifications for functional and operating limits and monitoring instruments and limiting control settings. The functional and operating limits for an ISFSI are limits on fuel or waste handling and storage conditions that are found to be necessary to protect the integrity of the stored fuel or waste container, to protect employees against occupational exposures and to guard against the uncontrolled release of radioactive materials.

- 10 CFR §72.106(b) requires that 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]. The minimum distance from the spent fuel, high-level radioactive waste, or reactor-related GTCC waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters [300 ft].

- 10 CFR §72.122(b)(4) requires that if the ISFSI is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

- 10 CFR §72.122(h)(3) requires that ventilation systems and off-gas systems must be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions.

- 10 CFR §72.126(d) requires that the ISFSI be designed to provide means to limit to levels as low as is reasonably achievable the release of radioactive materials in effluents during normal operations; and control the release of radioactive materials under accident conditions. Analyses must be made to show that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limit given in §72.104. Analyses of design basis accidents must be made to show that releases to the general environment will be within the exposure limits given in §72.106. Systems designed to monitor the release of radioactive materials must have means for calibration and testing their operability.

- 10 CFR §72.128(a)(3) requires that spent fuel storage systems be designed with confinement structures and systems.

The HI-STORM 100 System is designed for long-term confinement and dry storage of pressurized water reactor and Boiling Water Reactor spent nuclear fuel. The design of the HI-STORM 100 System is discussed in detail in Section 4.2.3 of the Diablo Canyon ISFSI SAR. In Section 4.2.3.3.6, the applicant states that all components of the confinement system are

classified as important to safety. The major components of the HI-STORM 100 System that are classified as important to safety include the sealed multi-purpose canister (MPC) and the storage cask. The MPC is designed to maintain a confinement barrier during all normal, off-normal, and accident conditions.

The confinement boundary for the Holtec HI-STORM 100 System includes the MPC shell, the bottom baseplate, the MPC lid (including the vent and drain port cover plates), the MPC closure ring, and the associated welds. The welds forming the confinement boundary are described in detail in Section 7.1.3 of the HI-STORM 100 System FSAR. The MPC is designed, fabricated, and tested in accordance with the applicable requirements of ASME Code, Section III, Subsection NB, to the maximum extent practicable (ASME International, 1998). The MPC lid and MPC closure ring seal welds are designed to maintain confinement during normal and design-basis accident conditions.

In Section 6.1 of the SAR, the applicant states that no releases of any type of radioactive material will occur during normal operations. This statement is in agreement with the statements in Section 7.2.3 of the HI-STORM 100 System FSAR, which were previously reviewed and found acceptable by the staff (U.S. Nuclear Regulatory Commission, 2002). Thus, leakage of the MPCs during normal conditions was not reevaluated for this safety evaluation.

In Section 8.2.7 of the SAR, leakage from the HI-STORM 100 System during hypothetical accident conditions was evaluated. Following the methodology in the HI-STORM 100 System FSAR and in accordance with Interim Staff Guidance Number 5 (U.S. Nuclear Regulatory Commission, 1999), the applicant calculated the dose to an individual continuously present at the controlled-area boundary for 30 days at the location nearest to the Diablo Canyon ISFSI. This hypothetical, worst-case calculation, using methodology from Regulatory Guide 1.145 (U.S. Nuclear Regulatory Commission, 1983), yielded a total effective dose equivalent of 8.3 μSv [0.83 mrem] from a single leaking MPC. This value is lower than that reported by Holtec International (2002); the primary contributor to the decrease is the increased distance to the receptor from 100 m [330 ft] for the generic HI-STORM 100 System analyses (Holtec International, 2002) to approximately 430 m [1,400 ft] for the Diablo Canyon ISFSI (Pacific Gas and Electric Company, 2002). The accident dose rates (caused by direct and scattered radiation and a hypothetical release) for the HI-STORM 100 System do not exceed limits specified in 10 CFR §72.106(b).

While a hypothetical accident condition leakage calculation was performed for the HI-STORM 100 System, the applicant expects that there will be no release of radioactive materials in effluents during normal and all credible accident conditions. This is supported by the applicant analyses, which demonstrate that the MPC would maintain its confinement integrity under the design-basis normal, off-normal, and accident conditions (including earthquake, tornado, flood, drops and tip-over, fire, explosions, leakage, electrical accident, loading of an unauthorized assembly, extreme environmental temperature, loss of neutron shielding, adiabatic heat-up, blockage of vents and inlets, fuel rod rupture, transmission tower collapse, and lift jack failure). Based on the results of the applicant analyses, the staff agrees that the MPC confinement integrity would be maintained under the design-basis normal, off-normal, and accident conditions.

The staff, therefore, concludes with reasonable assurance that the risk of radioactive effluents released to the general public from storing as many as 140 HI-STORM 100 System casks at the Diablo Canyon ISFSI is insignificant and meets the requirements of 10 CFR §72.106(b). The staff also concludes that the stainless steel welded casks (with redundant welds in the lid enclosure of the cask) manufactured and inspected according to ASME Code, as approved by the staff, are not expected to release radioactive effluents, meeting the requirements of 10 CFR §72.122(b), §72.126(d), and §72.128(a)(3).

The staff reviewed the applicable chapters of the SAR and found that the conclusions that were made by the applicant were in agreement with the Holtec HI-STORM 100 System FSAR, already determined acceptable by staff (U.S. Nuclear Regulatory Commission, 2002). The staff also reviewed the site technical specifications proposed by the applicant and found those portions related to the confinement integrity of the HI-STORM 100 System to be acceptable, meeting the requirements of 10 CFR §72.122(h), §72.126(d), and §72.128(a)(3).

9.1.2 Confinement Monitoring

The staff review of this section focused on two areas: the continuous monitoring of closure seal effectiveness and the measure of radionuclides released to the environment during normal and accident conditions. The staff reviewed Sections 3.3.1.3, 3.3.1.5, 3.3.1.7, 4.2.3.3, and Chapter 6 of the Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2002). The information presented has been reviewed for conformance with the following regulatory requirements:

- 10 CFR §72.24(l)(1) requires a description of the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal operations and expected operational occurrences. The description must identify the design objectives and the means to be used for keeping levels of radioactive material in effluents to the environment as low as is reasonably achievable and within the exposure limits stated in §72.104. The description must include an estimate of the quantity of each of the principal radionuclides expected to be released annually to the environment in liquid and gaseous effluents produced during normal ISFSI operations.

- 10 CFR §72.44(c)(3)(iv) requires confirmation that the limiting conditions required for safe storage are met.

- 10 CFR §72.122(h)(4) requires that storage confinement systems have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. For dry spent fuel storage, periodic monitoring is sufficient provided that periodic monitoring is consistent with the dry spent fuel storage cask design requirements. The monitoring period must be based upon the spent fuel storage cask design requirements.

- 10 CFR §72.126(c)(1) requires as appropriate for the handling and storage system, that effluent systems be provided. Means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions

must be provided for these systems. A means of measuring the flow of the diluting medium, either air or water, must also be provided.

- 10 CFR §72.128(a)(1) requires that spent fuel storage systems be designed with a capability to test and monitor components important to safety.

Based on the staff assessment of welded cask enclosures, as stated in NUREG–1536, "Standard Review Plan for Dry Cask Storage Systems;" Chapter 7 Section V.2 (U.S. Nuclear Regulatory Commission, 1997), the MPC, which is the confinement system for the HI-STORM 100 System, provides reasonable assurance that no effluents will be released and, therefore, requires no monitoring of the MPC for leakage. The seal weld will be inspected and tested in accordance with the requirements in Section 8.1.5 of the HI-STORM 100 System FSAR. These requirements were reviewed during the certification of the HI-STORM 100 System and were found to be acceptable by the staff (U.S. Nuclear Regulatory Commission, 2002).

The staff finds the proposal to provide no monitoring of the confinement barrier for the HI-STORM 100 System acceptable because the casks will be loaded, welded, inspected, and tested in accordance with appropriate cask design requirements, meeting the requirements of 10 CFR §72.122(h)(4) and §72.126(c)(1).

9.1.3 Protection of Stored Materials from Degradation

The application was reviewed to establish that the fuel cladding would not experience significant degradation during the requested 20-year storage period. The staff reviewed Sections 3.3.1.1, 3.3.1.2, 3.3.1.1.7.2, 4.4.1.2, and 5.1.1.2 and Table 3.4-2 of the Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2002). The information presented has been reviewed for conformance with the following regulatory requirements:

- 10 CFR §72.24(g) requires that the license application include an identification and justification for the selection of those subjects that will be probable license conditions and Technical Specifications.

- 10 CFR §72.122(h)(1) requires that the spent fuel cladding be protected during storage against degradation that leads to gross ruptures or be otherwise confined such that degradation of the fuel during storage does not pose operational safety problems with respect to its removal from storage.

Following the loading of the MPC, the main lid is welded and a helium leak test is performed on the seal weld. The MPC cavity is then vacuum dried and filled with helium fill gas. The vent and drain ports are then welded into place and a helium leak test is conducted on the vent and drain port covers. These steps are described in detail in the HI-STORM 100 System FSAR. The helium back-fill procedure ensures that the presence of oxidizing gasses in the MPC cavity will be minimized.

The thermal analysis of the HI-STORM 100 System indicates that the fuel cladding temperature will not exceed the limits established to prevent fuel clad degradation during storage.

The staff verified that the SAR was consistent with the information provided in the amended HI-STORM 100 System FSAR. The staff reviewed the proposed technical specifications and

found the portions related to the protection of stored materials from degradation in the HI-STORM 100 System to be acceptable, meeting the requirements of 10 CFR §72.122(h)(1).

9.2 Evaluation Findings

During the evaluation of confinement of the spent nuclear fuel stored at the Diablo Canyon ISFSI, the staff assumed that only the HI-STORM 100 System will be used. Based on the staff review of the applicant's submittal and the applicable technical specifications, the staff made the following findings.

- The radionuclide confinement analysis for the Holtec HI-STORM 100 System and the Diablo Canyon ISFSI Site has met the requirements of 10 CFR 72.24(l)(1) by providing a description of how radioactive materials in gaseous and liquid effluents will be controlled such that they are as low as is reasonably achievable. The requirements of 10 CFR §72.44(c) have been met based on the staff review of the Technical Specifications that have been submitted by the applicant. Because the MPC lid is welded and tested in accordance with ASME Code and is not expected to leak under normal, off-normal, and accident conditions, the staff finds that the requirements of 10 CFR §72.122(h)(3), §72.126(d), §72.128(a)(3), and §72.122(a) have been met.

- The staff concludes that the HI-STORM 100 System, which contains the MPC that has been welded and tested in accordance with ASME Code, is not expected to leak and therefore does not require confinement monitoring. Based on this finding, the requirements of 10 CFR §72.44(c), §72.122(h)(4), §72.126(c)(1), and §72.128(a)(3) are met.

- The staff concludes that the proposed Technical Specifications are sufficient to protect the stored materials from degradation in accordance with 10 CFR §72.24(g). The staff also finds that the proposed methods to protect the stored materials from degradation are acceptable to protect the spent fuel cladding from gross ruptures in accordance with 10 CFR §72.122(h)(1).

9.3 References

ASME International. *ASME Boiler and Pressure Vessel Code, Section III.* New York City, NY: American Society of Mechanical Engineers. 1998.

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Revision 1. Vols I and II.* HI–5014464. Docket 72-1014. Marlton, NJ: Holtec International. 2002.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation Safety Analysis Report, Amendment 1.* San Luis Obispo County, CA: Pacific Gas and Electric Company. 2002.

U.S. Nuclear Regulatory Commission. *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. Regulatory Guide 1.145. Rev. 1. Washington, DC: U.S. Nuclear Regulatory Commission. 1983.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Dry Cask Storage Systems*. NUREG–1536. Washington, DC: U.S. Nuclear Regulatory Commission. 1997.

U.S. Nuclear Regulatory Commission. *Normal, Off-Normal, and Hypothetical Dose Estimate Calculations*. Interim Staff Guidance Document-5, Rev. 1. Washington, DC: U.S. Nuclear Regulatory Commission. 1999.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Spent Fuel Dry Storage Facilities*. NUREG–1567. Washington, DC: U.S. Nuclear Regulatory Commission. 2000.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Amendment No. 1 Safety Evaluation Report*. Docket 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2002.

10 CONDUCT OF OPERATIONS EVALUATION

10.1 Conduct of Review

Chapter 9, "Conduct of Operations," of the Safety Analysis Report (SAR), describes the organization for the design, fabrication, construction testing, operation, modification, and decommissioning of the Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI), including the organizational structure, personnel responsibilities and qualifications, and the corporate interface with contractors and other outside organizations. The chapter includes discussions of the management and administrative control system, personnel qualifications, plans for preoperational and startup testing and for operations, operational readiness review, training, and emergency planning. The chapter includes descriptions of the responsibilities of key personnel, the training program, standards and procedures that govern daily operations, and records generated as a result of those operations. The purpose of the review is to ensure that the infrastructure to manage, test, and operate the Diablo Canyon ISFSI, including provisions for effective training, is acceptable.

The staff evaluated the proposed conduct of operations by reviewing Chapter 9 of the SAR, documents cited in the SAR, and other relevant literature. The staff also considered information related to the conduct of operations that was submitted by the applicant in response to the staff request for additional information.[1] The applicant has requested an exemption from the record keeping requirements of 10 CFR §72.72(d), which requires that spent nuclear fuel and high-level waste records be stored in duplicate at a separate location sufficiently remote that a single event would not destroy both sets. The applicant requested that the record keeping procedure used for records at the Diablo Canyon Power Plant (DCPP) be applied to the Diablo Canyon ISFSI records. The DCPP recordkeeping program satisfies the criteria of 10 CFR Part 50, Appendix B, and meets American National Standards Institute (ANSI) N45.2.9-1974. The staff reviewed this exemption request and considered it appropriate; the recordkeeping program was found acceptable for the DCPP because it (1) provides for a record keeping system equivalent to the requirements of 10 CFR §72.72(d) and (2) avoids a redundant and unnecessarily complex record keeping system. The exemption will only be issued on completion of the regulatory process described in 10 CFR Part 72.

The information in SAR Chapter 9 was reviewed with respect to the regulations in 10 CFR §72.24(h), §72.24(i), §72.24(j), and §72.24(k); §72.28; §72.40(a)(4), §72.40(a)(9), and §72.40(a)(13); §72.180; and §72.184 (including requirements of 10 CFR Part 73, Appendix C) and Subpart I. These regulations also require compliance with certain regulations at 10 CFR Part 73. Where appropriate, findings of regulatory compliance are made for the 10 CFR Part 72 requirements that are fully addressed in Chapter 9 of the SAR. Because compliance with some regulations can only be determined by the integrated review of several sections in Chapter 9 and/or other chapters within the SAR, a finding of regulatory compliance is not made in each major section unless the specific regulatory requirement is fully addressed.

---

[1]Hall, J.R. *Request for Additional Information for the Diablo Canyon Independent Spent Fuel Storage Installation Application.* Letter (August 29) to L.F. Womack (U.S. Nuclear Regulatory Commission). Diablo Canyon Power Plant. Washington, DC: Nuclear Regulatory Commission. 2002.

Findings of technical adequacy and acceptability, however, are made for each section in Chapter 9 as it relates to the regulatory requirements.

10.1.1 Organizational Structure

Section 9.1, "Organizational Structure," of the SAR describes the organizational structure to be used to manage and operate the Diablo Canyon ISFSI. The review considered how the information in the SAR addresses the following regulatory requirements:

- 10 CFR §72.28(a) requires that the application include the technical qualifications, including training and experience, of the applicant to engage in the proposed activities.

- 10 CFR §72.28(c) requires that the application include a description of the applicant's operating organization, delegations of responsibility and authority and minimum skills and experience qualifications relevant to the various levels of responsibility and authority.

- 10 CFR §72.40(a)(4) requires that the applicant is qualified by reason of training and experience to conduct the operations covered by 10 CFR Part 72.

10.1.1.1 Corporate Organization

Sections 9.1.1, "Corporate Organization;" 9.1.2, "Corporate Functions, Responsibilities, and Authorities;" and 9.1.3, "In-House Organization" of the SAR describe the corporate organization that will be used to manage and operate the Diablo Canyon ISFSI.

The Diablo Canyon ISFSI will be managed by the same corporate structure that manages the DCPP. While the DCPP units are operating, costs for construction and operation of the Diablo Canyon ISFSI will be funded from power plant operations. After both the DCPP power generating units are shut down, funding for decommissioning will be from the DCPP Decommissioning Trust, which has been approved by the California Public Utilities Corporation. During the preoperational period, costs would be monitored and controlled by an ISFSI Program Manager. During operations, the Station Director would have this responsibility.

After the DCPP power generating units are decommissioned and the 10 CFR Part 50 operating licenses are terminated, the corporate management of the Diablo Canyon ISFSI would change to a revised Pacific Gas and Electric Company (PG&E) organizational structure. At that time, the U.S. Nuclear Regulatory Commission (NRC) will be notified about the nature of this revised structure.

The Senior Vice President, Generation and Chief Nuclear Officer, has overall corporate responsibility for Diablo Canyon ISFSI safety. His responsibilities include performance of the staff in designing, fabricating, constructing, testing, operating, modifying, decommissioning, and providing technical support to the ISFSI. He reports to the President and Chief Executive Officer of PG&E.

Under the Senior Vice President, Generation and Chief Nuclear Officer, the Vice President, Nuclear Services is responsible for engineering and design services, safety assessments, and

licensing services for the Diablo Canyon ISFSI. He interacts with the California Public Utilities Commission for ISFSI cost matters. The Vice President, Nuclear Services, is responsible for the design, fabrication, and testing of the first cask. The Engineering Director, who reports to the Vice President, Nuclear Services, will be responsible for the design, fabrication, and testing of all subsequent casks.

The Vice President, Diablo Canyon Operations, also at the corporate level, has responsibility for Diablo Canyon ISFSI operations and reports to the Senior Vice President, Generation and Chief Nuclear Officer.

A Nuclear Safety Oversight Committee reports to the Senior Vice President, Generation and Chief Nuclear Officer. This committee is chaired by the Vice President, Nuclear Services. The committee functions and responsibilities encompass both the DCPPs and the Diablo Canyon ISFSI.

The corporate management for the Diablo Canyon ISFSI is the same as that for the DCPP. Programs that are used for the DCPP, such as radiation protection, environmental monitoring, emergency preparedness, quality assurance, and training, will be adopted as necessary and will be employed to ensure safe operation of the Diablo Canyon ISFSI. Legal support will be provided from PG&E headquarters, and technical and operational support will be available from DCPP personnel and outside consultants for licensing, quality assurance (QA), engineering, radiation protection, maintenance, testing, emergency planning, security, and decommissioning.

Quality control functions will be performed by individuals independent of the Diablo Canyon ISFSI line organization. Results of QA audits and recommendations for improvement will be provided to the Senior Vice President, Generation and Chief Nuclear Officer. Prior to operations, they will also be presented to the ISFSI Program Manager, and during operations they will be provided to the Station Director. The frequency and scope of QA audits is adequately addressed in the Diablo Canyon ISFSI QA Program, which was included as an attachment to the license application.

The primary difference in the corporate management structure between the preoperational and operational phases is that during operations, day-to-day management of the Diablo Canyon ISFSI activities shifts from the ISFSI Program Manager to the on-site Station Director.

The Diablo Canyon ISFSI Program Manager manages the day-to-day activities during the preoperational phase and ensures that the design, fabrication, construction, fuel loading, testing, and initial operation of the first cask are safely completed. He is also responsible for cost control for these activities. The ISFSI Program Manager develops the license application and is responsible for licensing coordination with federal and state officials. He reports to the Director, Strategic Projects, and Assistant to the Vice President, Nuclear Services, who reports to the Vice President, Nuclear Services.

During the Diablo Canyon ISFSI preoperational phase, the Vice President, Nuclear Services, is responsible for the overall safety as well as industrial safety for the ISFSI.

The staff review finds the corporate organizational structure acceptable because it defines the relationships between corporate organizations and delineates authority and responsibility. Responsibility is clear to specific individuals and parts of the organization, and the functions of

radiation protection and other safety agencies are provided organizationally separate lines of reporting from those responsible for Diablo Canyon ISFSI operations. The staff also determined that a Nuclear Oversight Committee exists and is properly organized and staffed and, therefore, is acceptable. The Diablo Canyon ISFSI corporate management will be the same as that used for the DCPP. This corporate management structure and functioning has consistently been found to be acceptable by the NRC.

10.1.1.2 On-Site Organization

Sections 9.1.3, "In-House Organization;" and 9.1.6, "Operating Organization, Management, and Administrative Control System" of the SAR present the on-site organization, including responsibilities and reporting relationships.

The Diablo Canyon ISFSI will be constructed, tested, and operated by the same organization responsible for the testing and operation of the DCPP. The only difference is that after the preoperational phase, responsibility for day-to-day operations will shift from the ISFSI Program Manager to the on-site Station Director. It is anticipated that approximately 11 full-time-equivalent personnel will be used to operate the Diablo Canyon ISFSI. These personnel will be from the existing DCPP organization but will be specifically trained as required to support ISFSI operations. The authorities, responsibilities, and reporting relationships of these personnel are presented in the SAR and will be updated in organization charts, functional descriptions, and job descriptions, as required.

During the Diablo Canyon ISFSI operational phase, the Station Director will be responsible for design, fabrication, construction, fuel loading, and testing of all casks after the first cask. He will also be responsible for the overall safety of ISFSI operations and for training and qualification of operations, maintenance, radiation protection, and security personnel. The Station Manager reports to the Vice President, Diablo Canyon Operations.

The Manager, Operations, reports to the Station Director. The Manager, Operations, is responsible for administering, coordinating, planning, and scheduling all Diablo Canyon ISFSI operating activities. The Manager, Operations, provides operating procedures and ensures operating personnel are familiar with and use them.

The Director, Maintenance Services, reports to the Station Director. The Director, Maintenance Services, supervises Diablo Canyon ISFSI maintenance and work planning.

The Diablo Canyon ISFSI specialists and security staff will conduct the day-to-day operations of the ISFSI. In conducting these activities, they will use license requirements, technical specifications, the physical security plan, plant procedures, and applicable regulations. The ISFSI specialists will report to either the Manager, Operations, or the Director, Maintenance Services, according to their discipline.

All operations associated with the Diablo Canyon ISFSI are managed and approved by PG&E. Contractors and consultants may support various design and engineering activities for the ISFSI and its components. During operations, the Station Director is responsible for oversight of consultant and contractor work.

During both preoperational and operational periods, functions, such as engineering design, construction, quality assurance, radiation protection, testing, operations, and security, will be performed by DCPP personnel. The existing DCPP Plant Staff Review Committee will review any issues affecting the safe storage of spent nuclear fuel. The Plant Staff Review Committee is chaired by the Station Director. The duties and responsibilities of this committee are adequately covered by the DCPP Final Safety Analysis Report.

A formal order of succession and delegation of authority will be established to ensure continuity of operations and the ability to respond to off-normal events. The Station Director will formally designate personnel qualified to act in his absence.

The staff review finds the on-site organizational structure acceptable because it defines relationships between on-site organizations and liaisons with outside organizations and delineates authority and responsibility. The position responsible for oversight of outside organizations that manufacture canisters is clearly defined. The functions of radiation protection and other safety agencies are provided organizationally separate lines of reporting from those responsible for Diablo Canyon ISFSI operations. The staff also determined that a Plant Staff Review Committee exists and is properly organized and staffed and, therefore, is acceptable. The Diablo Canyon ISFSI onsite organization will be the same as that used for the DCPP. This onsite management structure and functioning has consistently been found to be acceptable by the NRC.

10.1.1.3 Management and Administrative Controls

Sections 9.1.6.3, "Administrative Control;" 9.4.1, "Procedures;" and 9.4.2, "Records" of the SAR describe management and administrative controls that will be employed at the Diablo Canyon ISFSI.

In general, the NRC-approved management and administrative controls that are in effect at the DCPP will be used at the ISFSI. QA audits conducted in accordance with the DCPP QA Program will be used to evaluate the adequacy of management and administrative controls, including procedures. The DCPP QA Program has been adequate for defining audit frequencies, documenting and communicating results, resolving issues, and implementing corrective action.

The change control program used by the DCPP will be used for the Diablo Canyon ISFSI.

The Diablo Canyon ISFSI commits to conducting all activities important to safety using detailed, written procedures. In addition, preoperational, normal operating, maintenance, and surveillance testing will be in effect prior to beginning loading operations. The associated procedures will be prepared, reviewed, and approved in accordance with the DCPP administrative program used for these purposes. The Diablo Canyon ISFSI commits to preparing these procedures in sufficient detail that qualified and trained personnel can implement them without incident.

Diablo Canyon ISFSI records will be maintained using established practices employed by the DCPP and the DCPP QA Program. The scope of the recordkeeping procedures includes the records retention period; QA requirements; operating records that document principal maintenance, alterations, and additions to components or facilities; records of off-normal

occurrences and events associated with radioactive releases; records for decommissioning; and environmental surveys.

The staff found that the management and administrative controls committed to in the SAR are adequate and would provide reasonable assurance that the operations at the site will be properly controlled and documented. The applicant has described an organizational system for preparing and controlling procedures, including changes to procedures, and for generating and maintaining adequate records. The staff found this organizational system acceptable based on the descriptions and commitments given in the SAR.

10.1.2 Pre-operational Testing and Startup Operations

Section 9.2, "Preoperational and Startup Testing," of the SAR describes the startup testing plans for storage systems and any associated equipment and facility testing. The Diablo Canyon ISFSI commits to completing this testing before initial movement of any spent nuclear fuel for placement on the ISFSI storage pad.

The Diablo Canyon ISFSI commits to preparing, reviewing, approving, and performing test procedures in accordance with existing DCPP administrative controls and the DCPP QA Program. This commitment includes requiring that any test procedures used by outside vendors will meet the requirements of a PG&E-approved QA program, and that PG&E will approve any such procedures and witness their performance.

The review considered how the information in the SAR addresses the following regulatory requirements:

- 10 CFR §72.24(i) requires that if the ISFSI incorporates structures, systems, or components important to safety whose functional adequacy or reliability have not been demonstrated by prior use for that purpose or cannot be demonstrated by reference to performance data in related applications or to widely accepted engineering principles, an identification of these structures, systems, or components along with a schedule showing how safety questions will be resolved prior to the initial receipt of spent fuel, high-level radioactive waste, and/or reactor-related GTCC waste as appropriate for storage at the ISFSI.

- 10 CFR §72.24(p) requires that the application contains a description of the program covering preoperational testing and initial operations.

- 10 CFR §72.28 requires that the applicant technical qualifications to conduct the proposed operations be adequately demonstrated.

- 10 CFR §72.40(a)(4) requires that the applicant is qualified by reason of training and experience to conduct the operation covered by 10 CFR Part 72.

10.1.2.1 Pre-Operational Testing Plan

Sections 9.2.1, "Administrative Procedures for Conducting Test Program;" 9.2.2, "Test Program Description;" and 9.2.3, "Preoperational Test Plan" of the SAR describe various aspects of the Preoperational Test Plan.

Preoperational testing verifies that the individual components of the storage system, facilities, and equipment meet respective functional requirements as described in the SAR. Preoperational testing must be successfully completed prior to beginning startup testing.

Any discrepancies identified during preoperational testing will be resolved in accordance with the existing DCPP procedures and process for discrepancy resolution.

The preoperational test plan will include testing of the Cask Transfer Facility (CTF), the transporter, and storage system supporting systems such as welding equipment and dehydration equipment. These tests will confirm operation in accordance with functional specifications and the requirements of the SAR. Typical aspects tested will be controls, hydraulic systems, brakes, instruments, and protective devices. The existing DCPP Control of Heavy Loads Program will be used to conduct and manage load testing of components.

Other testing that will be performed according to the preoperational test plan includes security system testing and construction-related testing. Control and calibration of measuring and test equipment will be conducted according to the existing DCPP QA Program.

There are no unresolved safety issues regarding the proposed Diablo Canyon ISFSI and the associated storage system. If any such questions should arise, PG&E has committed to address them and will incorporate them in the formal Corrective Action Program, if necessary.

The staff review found that the preoperational test plan includes the necessary tests and provides for proper evaluation, approval, and use of the test results. Appropriate administrative procedures will be developed to support the preoperational testing and startup programs, and a staff review of operational readiness will be performed prior to operation. There are no unresolved safety issues regarding the proposed Diablo Canyon ISFSI.

10.1.2.2 Startup Plan

Sections 9.2.1, "Administrative Procedures for Conducting Test Program;" 9.2.2, "Test Program Description;" and 9.2.4, "Startup Test Plan" of the SAR describe various aspects of the Startup Test Plan.

Startup testing will verify that the complete loading and unloading sequence using the storage system components, facilities, and equipment works together in accordance with the requirements of the SAR and the Diablo Canon ISFSI Technical Specifications. The applicant commits to completing startup testing prior to handling spent nuclear fuel.

Any discrepancies identified during startup testing will be resolved in accordance with the existing DCPP procedures and process for discrepancy resolution.

Startup testing will be controlled using an overall startup testing plan. According to this overall procedure, individual test procedures will be used to supplement Diablo Canyon ISFSI operational procedures.

Startup testing will be conducted using a multi-purpose canister (MPC) handling simulator that will mimic the dimensions and center of gravity of an actual MPC. The simulator will also be equipped with lifting and handling fixtures similar to those of an actual canister. Similar

mockups will be used to test welding equipment including the actual welds, moisture removal, helium filling, and canister cool down.

The applicant states that the personnel conducting the startup testing will have completed the applicable training requirements.

Startup testing at the Diablo Canyon ISFSI will include the following.

(1) Preparing the transfer cask and MPC simulator for movement into the spent nuclear fuel pool

(2) Moving the transfer cask into the Fuel Handling Building/Auxiliary Building (FHB/AB), upending it, and placing it in the temporary seismic restraint structure

(3) Placing the transfer cask into the spent nuclear fuel pool and simulating movement of fuel, including a dummy fuel assembly, into the MPC

(4) Installing the MPC lid retention device, removing the transfer cask from the spent nuclear fuel pool, and moving it back to the cask wash-down area and into the temporary seismic restraint structure

(5) Decontaminating the transfer cask

(6) Removing the MPC lid retention device, welding the MPC lid, removing moisture, filling the MPC with helium, and removing the lid weld

(7) Installing the transfer cask top lid

(8) Loading the transfer cask onto the cask transport frame using the Fuel Handling Building/Auxiliary Building crane and removing it from the building

(9) Transporting the loaded transfer cask from the Fuel Handing Building/Auxiliary Building to the CTF using the transporter

(10) Moving the MPC simulator from the transfer cask into a storage cask at the CTF

(11) Placing the top lid on a loaded overpack and raising the storage cask in the CTF

(12) Transporting a loaded overpack from the CTF to the ISFSI pad location

(13) Positioning and fastening the loaded overpack to the ISFSI pad

(14) Removing the loaded overpack from the ISFSI pad

(15) Transporting the loaded overpack from the ISFSI pad to the CTF

(16) Removing the top lid from a loaded overpack

(17) Transferring the multi-purpose simulator from the overpack back into the transfer cask

(18) Transporting the loaded transfer cask to the FHB/AB using the onsite transporter

Section 9.2.5, "Operational Startup Testing" of the SAR provides for additional testing. The operational startup testing would be performed during the initial loading of an MPC. The applicant commits to limiting these tests to gathering information that is only available when nuclear fuel is loaded in an MPC or for final verification of data obtained during startup testing.

Section 9.2.6, "Operational Readiness Review Plan" of the SAR commits to execution of an operational readiness review prior to beginning Diablo Canyon ISFSI operations for the first set of casks. The purpose of this readiness review will be to verify that appropriate actions have been completed prior to initial MPC loading. The operational readiness review will ensure, at a minimum, that the final actions have been completed.

(1) Results from operational and startup testing are satisfactory, and all associated corrective actions or lessons learned have been properly incorporated in Diablo Canyon ISFSI procedures.

(2) Necessary radiological procedures and controls are in place.

(3) Required operational procedures are approved and in place for surveillance, operations, and emergency response.

(4) All engineering issues related to the storage system initial use have been resolved.

(5) Fire protection procedures are approved and in place.

(6) Maintenance procedures are approved and in place, and all required ISFSI components are ready for use.

(7) The Cask Transportation Evaluation Program is in place.

(8) Procedures for planning are approved and are in place to ensure that the characteristics of fuel assemblies meet requirements of the SAR and the Diablo Canyon ISFSI technical specifications.

The staff review found that the startup test plan includes the necessary tests and provides for proper evaluation, approval, and use of the test results. Appropriate administrative procedures will be developed to support the startup test program, and a review of operational readiness will be performed prior to operation. ISFSI staff will be properly trained to conduct the proposed operations.

10.1.3 Normal Operations

Section 9.4, "Normal Operations" of the SAR includes Subsections 9.4.1, "Procedures;" and 9.4.2, "Records." These sections describe administrative controls and the conduct of operations

for activities important to safety. They also describe the management controls applied to maintaining records. The review considered how the information provided in the SAR addresses the following regulatory requirements.

- 10 CFR §72.24(p) requires that the application contain a description of the program covering preoperational testing and initial operations.
- 10 CFR §72.28 requires that the applicant technical qualifications to conduct the proposed operations be adequately demonstrated.
- 10 CFR §72.40(a)(4) requires that the applicant is qualified by reason of training and experience to conduct the operation covered by 10 CFR Part 72.

10.1.3.1 Procedures

Section 9.4.1, "Procedures" of the SAR states that activities important to safety will be conducted in accordance with detailed, written, approved procedures. In addition, the applicant commits to having preoperational, normal operating, maintenance, and surveillance testing in place prior to beginning fuel loading. All procedures, and revisions to them, will be prepared, reviewed, and approved using existing DCPP administrative programs for procedure preparation, review, and approval. These procedures will also be compliant with the DCPP QA Program. All procedures will be sufficiently detailed that qualified and trained personnel would be able to perform the actions without incident. The SAR addresses administrative, radiation protection, maintenance and surveillance testing, operating, and QA-implementing procedures separately.

The Diablo Canyon ISFSI administrative procedures will provide operating personnel with a clear understanding of operating philosophy and management policies. The scope of these procedures will include personnel conduct; procedure preparation, review, approval, and revision; personnel safety; the working environment; and procurement. The objective of these procedures is to ensure that these activities are completed with a high degree of readiness, quality, and success.

Diablo Canyon ISFSI radiation protection procedures will implement a radiation protection program that demonstrates compliance with 10 CFR Part 20 requirements, including as low as reasonably achievable (ALARA) principles. The scope of these procedures will include acquisition of data, use of equipment, and qualification and training of radiation protection personnel. Existing DCPP radiation protection procedures will be revised as necessary to address ISFSI operations. These existing procedures have proven adequate for monitoring exposure of employees, radiation surveys, maintenance monitoring, and radiation protection records maintenance. The revised radiation protection procedures will specifically address the safety of personnel performing fuel loading, fuel transport, fuel unloading, surveillance testing, and maintenance. Any entrance to or work performed inside the ISFSI protected area will be controlled by a radiation work permit and appropriate security checks. The operation and use of radiation monitoring equipment and the use of measurement and sampling techniques will be covered by procedures.

Diablo Canyon ISFSI maintenance and surveillance testing procedures will be established for preventative and corrective maintenance and for surveillance testing of ISFSI equipment and

instrumentation. An appropriate periodicity will be established for preventive maintenance, surveillance testing, calibrations, and load testing to preclude degradation of systems, equipment, and components. Corrective maintenance to rectify unexpected system, equipment, or component failures will also be controlled using procedures. Any structures, systems, or components important to safety that are commercial grade will be qualification tested prior to use. This testing will verify the functionality and the ability to carry full-rated load, where appropriate.

Subsequent to the qualification testing, standard preventive maintenance, surveillance testing, and corrective maintenance will be performed.

Diablo Canyon ISFSI operating procedures will include instructions for routine and projected off-normal operations. These operations include handling, loading, sealing, transporting, storing, and unloading and other operations important to safety.

Diablo Canyon ISFSI QA implementing procedures will be prepared for important-to-safety activities to ensure compliance with the DCPP QA Program. Similarly, the requirements for qualification of personnel will be implemented through formal procedures, which will specify that responsibility for quality rests with each individual.

The staff review found that the control of procedures, including procedure changes, described in the SAR was adequate. Preparation of procedures and procedure changes will have the appropriate level of detail and safety review. Training and certification of personnel will be accomplished using formal, written procedures.

10.1.3.2 Records

Section 9.4.2, "Records" of the SAR specifies that records will be maintained in accordance with established PG&E policies. The records management program is a part of the DCPP QA Program.

PG&E has requested an exemption from 10 CFR §72.72(d), which requires that spent nuclear fuel and high-level waste records be stored in duplicate at a separate location sufficiently remote that a single event would not destroy both sets. Pursuant to 10 CFR §72.140(d), PG&E proposes to use an NRC-approved QA program that satisfies the criteria of 10 CFR Part 50, Appendix B for the Diablo Canyon ISFSI. That program meets ANSI N45.2.9–1974. The applicant states that an exemption from the records storage requirements of 10 CFR §72.72(d) would allow records of spent nuclear fuel storage to be maintained in the same manner as with the DCPP QA Program.

The staff review found that the recordkeeping procedures committed to in the SAR are adequate to assure that records will be properly developed and maintained. The exemption from the requirements of 10 CFR §72.72(d) is appropriate, given that the record keeping program has been found acceptable for the DCPP, and it will provide a level of records management control equivalent to that of 10 CFR §72.72(d) and will avoid redundant and unnecessarily complex recordkeeping systems.

10.1.4 Personnel Selection, Training, and Certification

Sections 9.1.7, "Personnel Qualification Requirements;" and 9.3, "Training Program" of the SAR define the minimum qualification and training requirements for operation of the Diablo Canyon ISFSI.

The review considered how the SAR addressed meeting the following regulatory requirements.

- 10 CFR §72.40(a)(4) requires that the applicant is qualified by reason of training and experience to conduct the operation covered by 10 CFR Part 72.

- 10 CFR §72.40(a)(9) requires that the personnel training program comply with Subpart I of 10 CFR Part 72. Subpart I, Training and Certification of Personnel, consists of 10 CFR §72.190, §72.192 and §72.194, summarized below.

- 10 CFR §72.190 requires that operation of equipment and controls that have been identified as important to safety in the Safety Analysis Report and in the license must be limited to trained and certified personnel or be under the direct visual supervision of an individual with training and certification in the operation. Supervisory personnel who personally direct the operation of equipment and controls that are important to safety must also be certified.

- 10 CFR §72.192 requires that the applicant for a license under this part shall establish a program for training, proficiency testing, and certification of ISFSI personnel. This program must be submitted to the Commission for approval with the license application.

- 10 CFR §72.194 requires that the physical condition and general health of personnel certified for the operation of equipment and controls that are important to safety must not adversely affect safe operation of the Facility. For example, a condition that might cause impaired judgment or motor coordination must be considered in the selection of personnel. The physical condition and the general health of personnel certified for the operation of equipment and controls that are important to safety must not be such as might cause operational errors that could endanger other in-plant personnel or the public health and safety. Any condition that might cause impaired judgment or motor coordination must be considered in the selection of personnel for activities that are important to safety. These conditions need not categorically disqualify a person, if appropriate provisions are made to accommodate such defect.

10.1.4.1 Personnel Organization

Section 9.3, "Training Program" of the SAR states that, pursuant to 10 CFR §72.190 and §72.192, Diablo Canyon ISFSI personnel will receive training and indoctrination designed to provide and maintain a well-qualified work force for safe and effective operation. The existing DCPP training programs will be used. These programs are accredited by the Institute of Nuclear Power Operations, and the General Employee Training portions are directly applicable to the Diablo Canyon ISFSI. Supplemental training will be provided to the

operations, maintenance, security, and emergency planning personnel who are assigned duties at the ISFSI.

Supplemental training will be developed under the PG&E training program, which uses a systematic approach to training to provide a comprehensive, site-specific training, assessment, and qualification program for the ISFSI. This training program will include periodic requalification and retraining, records keeping, and medical requirements.

The staff review found that the personnel organization and systematic approach to training are acceptable. The personnel organization identifies the position that has responsibility for the training program, including implementing the program and maintaining training records.

10.1.4.2 Selection and Training of Operating Personnel

Section 9.3.3, "Continuing Training" of the SAR states that procedures to implement retraining, proficiency testing, and requalification for ISFSI personnel will be prepared, as required.

Section 9.1.7 specifies that DCPP personnel operating or working at the Diablo Canyon ISFSI will meet or exceed the qualifications specified by NRC Regulatory Guide 1.8 (U.S. Nuclear Regulatory Commission, 1987), with specific exceptions as identified in the license application consistent with the DCPP QA Program.

The Station Director is required to have a minimum of 8 years of power plant experience, at least 3 years of which shall be nuclear power plant experience. At most, 2 years of the remaining 5 years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training, on a one-for-one basis. The Station Director must also be qualified in accordance with the Diablo Canyon ISFSI Operations Training Program.

ISFSI specialists and security staff shall have a high school diploma or have successfully completed the General Education Development test. The ISFSI specialists must have at least 2 years of power plant experience, at least 1 year of which must be nuclear power plant experience. The ISFSI specialists also shall have the required training for their specific assignments required by the Diablo Canyon ISFSI Operations Training Program.

DCPP security staff who support the Diablo Canyon ISFSI will be trained and qualified in accordance with the DCPP Security Training and Qualifications Plan.

Diablo Canyon ISFSI fuel handling operations will be performed by, or supervised by, personnel trained and qualified through the Diablo Canyon ISFSI Operations Training Program. During operations, operation of equipment and controls that are important to safety will be limited to those personnel who are qualified and trained through the Diablo Canyon Operations Training Program or personnel under the direct supervision of persons trained and qualified through the Diablo Canyon Operations Training Program.

The staff determined that the SAR provides an acceptable level of detail with respect to operator experience, instruction and training courses, examination and testing requirements, and the criteria for qualifications or revocations. Qualifications for operators must include applicable training and experience, which may be at facilities other than dry storage facilities. The minimum personnel qualification requirements are comparable to similar positions at

power reactor facilities described in Regulatory Guide 1.8 (U.S. Nuclear Regulatory Commission, 1987) and are generally equivalent to the qualification requirements that are in place at other ISFSIs. Tailoring the existing DCPP training program to the ISFSI is an acceptable approach.

In summary, the staff determined that the applicant has provided sufficient details concerning its personnel training and qualifications to provide reasonable assurance that its training and certification program will satisfy the requirements of 10 CFR Part 72, Subpart I. Certain operations will be performed only by trained and certified operators, and the physical condition and general health of operators will be considered in the qualification of operators, as required by 10 CFR §72.192 and §72.194 of Subpart I. The qualifications and certifications of the operators will be inspected and evaluated by NRC staff following the issuance of a license to ensure regulatory compliance prior to the conduct of licensed operations at the Facility.

As described in the previous text, the staff determined that the Diablo Canyon ISFSI training program, including the commitments made by the applicant, provides reasonable assurance of compliance with the standards in 10 CFR Part 72, Subpart I, and is consistent with the applicable regulatory guidance. This training program includes specific training in ALARA principles. Based on the description of its training program, the staff concludes that the training commitments are consistent with Regulatory Guide 8.8 (U.S. Nuclear Regulatory Commission, 1978), which provides guidance in training and instruction in ALARA principles for nuclear power plant personnel, and provides reasonable assurance that NRC requirements related to radiation protection training and ALARA principles will be satisfied.

10.1.4.3 Selection and Training of Security Guards

The requirements for the security organization are addressed in Chapter 19 of this Safety Evaluation Report (SER).

10.1.5 Emergency Planning

The DCPP Emergency Plan applicable to Diablo Canyon Nuclear Plant Units 1 and 2 describes the organization, assessment actions, conditions for activation of the emergency organization, notification procedures, emergency facilities and equipment, training, provisions for maintaining emergency preparedness, and recovery criteria used at the DCPP. This emergency plan will be used for the Diablo Canyon ISFSI. The Emergency Action Level classification for the Diablo Canyon ISFSI is Notification of Unusual Event. The applicant has proposed changes to the DCPP Emergency Plan necessary to accommodate the ISFSI.

The staff finds that the application of the DCPP Emergency Plan to the Diablo Canyon ISFSI provides reasonable assurance that any emergency conditions would be properly responded to and that the requirements of 10 CFR §72.24(k), §72.32(c), and §72.40(a)(11) would be met.

10.1.6 Physical Security and Safeguards Contingency Plans

The purpose of the Physical Security Plan for the Diablo Canyon ISFSI is to protect the stored spent nuclear fuel. The Physical Security Plan for the Diablo Canyon ISFSI is covered by the DCPP Physical Security Plan, the Safeguards Contingency Plan, and the Security Training

and Qualification Plan. These programs meet the requirements of 10 CFR Part 72, Subpart H, Physical Protection, 10 CFR §72.24(o), §72.40(a)(8), and the applicable portions of §73.55.

The DCPP security force will control access to the Diablo Canyon ISFSI protected area. This access will be limited to those who must enter for work-related activities. There will be a list of approved individuals, and identification badges will be required. Persons, vehicles, and hand-held items will be searched prior to entry to the protected area.

An intrusion detection system will be provided for the Diablo Canyon ISFSI protected area. Manned stations will be provided to monitor intrusion detector system alarms, coordinate security communications, and perform closed-circuit television surveillance and alarm assessment.

The DCPP Safeguards Contingency Plan addresses responses to potential threats and contains a responsibility matrix as guidance for security force actions as required by 10 CFR §72.184. The contingency planning includes detailed response procedures and means for obtaining assistance from local law enforcement agencies.

The DCPP Security Training and Qualification Plan defines training and qualification requirements for the security force as required by 10 CFR §73.55. The plan includes crucial security tasks and identifies the positions that must be trained in these tasks. The plan also provides requirements for initial and recurring training and a program for screening the background, physical condition, and mental qualifications of security force members.

The DCPP Physical Security Plan, Safeguards Contingency Plan, and Security Training and Qualification Plan will be implemented using written procedures, as required by 10 CFR §73.55(b)(3)(i).

Physical security is further addressed in Chapter 19 of this SER.

The staff finds that the DCPP Physical Security and Safeguards Contingency Plans would provide reasonable assurance that spent nuclear fuel at the Diablo Canyon ISFSI would be protected in accordance with requirements of 10 CFR Part 72, Subpart H, and 10 CFR Part 73, Appendix C.

10.2 Evaluation Findings

The staff reviewed the SAR and has determined that the Diablo Canyon ISFSI has established an acceptable plan to conduct operations. The staff has determined that

- The conduct of operations described for the Diablo Canyon ISFSI meets the requirements of 10 CFR §72.40(a)(4) in that PG&E will be qualified by training and experience to conduct the operations included in the license.

- The conduct of operations described for the Diablo Canyon ISFSI meets the requirements of 10 CFR §72.24(h), §72.24(i), §72.24(j), and §72.24(k); §72.28; §72.40(a)(9), §72.40(13); §72.180; §72.184; and Part 72, Subpart I, in that PG&E has provided a description of the procedures and policies that assure that operation of equipment and controls that are important to safety is limited to

trained and certified personnel; has provided an adequate operator training and certification program; has operator qualifications that assure that the physical condition and general health of operators will not cause operational errors that could endanger other workers or the health and safety of the public; and has an adequate physical security plan.

- The staff is granting an exemption to the recordkeeping requirements of 10 CFR §72.72(d) because an equivalent recordkeeping system has already been established at the DCPP and granting the exemption would avoid redundant recordkeeping systems.

10.3 References

U.S. Nuclear Regulatory Commission. *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be ALARA.* Regulatory Guide 8.8. Rev. 3. Washington, DC: U.S. Nuclear Regulatory Commission. 1978.

U.S. Nuclear Regulatory Commission. *Qualification and Training of Personnel for Nuclear Power Plants.* Regulatory Guide 1.8. Rev. 2. Washington, DC: U.S. Nuclear Regulatory Commission. 1987.

11 RADIATION PROTECTION EVALUATION

11.1 Conduct of Review

The objective of this section is to evaluate the capability of the organizational, design, and operational elements of the Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) radiation protection plan to meet regulatory requirements. The regulatory requirements for providing adequate radiation protection to personnel and members of the public include

- 10 CFR §20.1101(a) requires that a licensee develop, document, and implement a radiation protection program.
- 10 CFR §20.1101(b) requires that a licensee use sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable.
- 10 CFR §20.1101(c) requires that a licensee periodically (at least annually) review the radiation protection program content and implementation.
- 10 CFR §20.1101(d) requires that a licensee, as part of the radiation protection program, establish a constraint for air emissions of radioactive materials to the environment such that a member of the public is not expected to receive a total effective dose equivalent in excess of 0.1 mSv [10 mrem] per year from these emissions.
- 10 CFR §20.1201(a) requires that a licensee control occupational dose to the following annual dose limits: a total effective dose equivalent of 0.05 Sv [5 rem] or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye of 0.5 Sv [50 rem], whichever is most limiting, a dose equivalent of 0.15 Sv [15 rem] to the lens of the eye, and a shallow-dose equivalent of 0.50 Sv [50 rem] to the skin or an extremity.
- 10 CFR §20.1301(a) establishes dose limits for a member of the public, including a total effective dose equivalent of 1 mSv [0.1 rem] in a year, and a maximum dose in any unrestricted areas of 0.02 mSv [0.002 rem] in an hour.
- 10 CFR §20.1301(b) requires that if a licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals.
- 10 CFR §20.1301(d) requires that the licensee comply with the environmental radiation standards in 40 CFR Part 190.
- 10 CFR §20.1302(a) requires a licensee to perform radiation surveys and monitor radioactive materials in effluents in unrestricted and controlled areas to demonstrate compliance with the dose limits for members of the public in 10 CFR §20.1301.

- 10 CFR §20.1302(b) requires that the licensee show compliance with the limits in 10 CFR §20.1301, by either demonstrating compliance with the dose limit to an individual by calculation or measurement, or by demonstrating that radioactivity in gaseous and liquid effluents does not exceed the values in table 2 of Appendix B to Part 20, and the dose from external sources would not exceed 0.02 mSv [0.002 rem] in an hour and 0.5 mSv [0.05 rem] in a year.

- 10 CFR §20.1406 requires that an applicant describe how facility design and procedures for operation will minimize contamination and generation of radioactive waste, and facilitate decommissioning.

- 10 CFR §20.1501(a)(1) requires that a licensee make surveys necessary to comply with 10 CFR Part 20.

- 10 CFR §20.1501(c) requires that dosimeters that are used by a licensee are processed and evaluated by a processor holding accreditation from the National Voluntary Laboratory Accreditation Program.

- 10 CFR §20.1701 requires that a licensee use process or other engineering controls to control the concentrations of radioactive material in the air.

- 10 CFR §20.1702 requires that when it is not practicable to apply process or other engineering controls, that the licensee shall increase monitoring and limit intakes by use of other controls, including access control, limitation of exposure times, use of respiratory protection, etc.

- 10 CFR §72.24(e) requires that the SAR include the means for controlling and limiting occupational radiation exposure within the limits given in 10 CFR Part 20, and for meeting the objective of maintaining exposures as low as is reasonably achievable.

- 10 CFR §72.104(a) requires that, during normal operations and anticipated occurrences, the annual dose equivalent to any real individual 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 organ, from various sources, including planned discharges of radioactive materials to the environment.

- 10 CFR §72.104(b) requires that operational restrictions are established to meet objectives for maintaining doses as low as is reasonably achievable from radioactive materials in effluents.

- 10 CFR §72.104(c) requires that operational limits for radioactive materials in effluents are established to ensure that the dose limits in 72.104(a) are met.

- 10 CFR §72.106(b) requires that any individual located on or beyond the nearest controlled area boundary shall not receive a dose greater than 0.05 Sv [5 rem] to the whole body or any organ from any design basis accident, and that the

minimum distance from the spent fuel waste handling and storage facilities to the nearest boundary shall be at least 100 meters.

- 10 CFR §72.126(a) requires that radiation protection systems must be provided for areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and inspections may involve occupational exposure, must be designed, fabricated, located, shielded, controlled and tested to control external and internal radiation exposures. The design must include means to, among other things, control access to areas of potential contamination or high radiation, measure and control contamination, minimize worker time, and shield personnel.

- 10 CFR §72.126(c)(1) requires that, as appropriate for the handling and storage system, effluent systems must be provided, as well as methods for measuring the amount of radionuclides in the effluents.

- 10 CFR §72.126(c)(2) requires that areas containing radioactive materials must be provided with systems for measuring the direct radiation levels in and around these areas.

- 10 CFR §72.126(d) requires that the ISFSI be designed to limit effluents to levels that maintain doses as low as is reasonably achievable, and analyses must show that releases to the environment during normal operations and anticipated occurrences will be within the exposure limit given in 10 CFR §72.104.

Pacific Gas and Electric Company (PG&E) proposes to use the HI-STORM 100 System. This cask system consists of (1) interchangeable multi-purpose canisters (MPC) that contain the fuel; (2) a storage overpack that contains the MPC during storage; and (3) a transfer cask that contains the MPC during loading, unloading, and transfer operations. The system has been reviewed and approved by U.S. Nuclear Regulatory Commission (NRC) according to the general license provision of 10 CFR Part 72. Amendment 1 of the HI-STORM 100 System Certificate of Compliance (CoC) became effective on July 15, 2002 (U.S. Nuclear Regulatory Commission, 2002a).

The review included Chapter 7, "Radiation Protection" of the SAR (Pacific Gas and Electric Company, 2002). Chapter 7 of the SAR describes the radiation protection features of the proposed Diablo Canyon ISFSI that ensure that radiation exposures to personnel and members of the public meet the regulatory requirements. Information included in the HI-STORM 100 System Final Safety Analysis Report (FSAR) Revision 1 (Holtec International, 2002) that pertains to ISFSI radiation protection was also considered in the review.

11.1.1 As Low As Reasonably Achievable Considerations

The objective of this section is to evaluate whether PG&E has appropriately considered the goal of maintaining occupational doses and doses to the members of the public as low as reasonably achievable (ALARA) during the operation of the ISFSI. Considerations related to maintaining doses ALARA are described in Section 7.1 of the Safety Assessment Report (SAR).

11.1.1.1 As Low As Reasonably Achievable Policy and Program

The policy and program for maintaining doses ALARA are described in Section 7.1.1, "Policy Considerations and Organization" of the SAR. The primary objective of the Health Physics Program is to maintain radiation exposure to workers, visitors, and members of the public below regulatory limits and otherwise ALARA.

PG&E considers the Diablo Canyon Power Plant (DCPP) program for maintaining doses ALARA, which complies with the regulatory requirements of 10 CFR Parts 20 and 50, sufficient for the proposed ISFSI operations under 10 CFR Part 72. The program for maintaining doses ALARA follows the guidance in Regulatory Guides 8.8 (U.S. Nuclear Regulatory Commission, 1978) and 8.10 (U.S. Nuclear Regulatory Commission, 1975).

The DCPP Health Physics program objectives are described in Section 12.3.2 of the DCPP FSAR update (Pacific Gas and Electric Company, 2001).

11.1.1.2 Design Considerations

The description of the design considerations to maintain doses ALARA is provided in Section 7.1.2 of the SAR, "Design Considerations," which delineates the following specific features of the Diablo Canyon ISFSI Facility:

- Centralized location of the ISFSI within the DCPP site boundary to reduce offsite doses
- Placement of the storage pads at a sufficient distance from buildings and occupied places, so doses to workers are maintained ALARA
- Adequate spacing between storage casks on the pads to allow personnel to function efficiently during placement, surveillance, maintenance, and repair activities
- Use of a restricted area fence and a security perimeter fence that protect individuals against undue risks from radiation exposure and prevent unauthorized access to the ISFSI
- Use of a thick biological shielding in overpacks that provides gamma and neutron shielding
- Use of a dry environment inside the weld-sealed MPC to prevent potential release of radioactive liquids and other radioactive effluents from inside the canister
- Use of inflatable annulus seals to preclude spent nuclear fuel water contacting the exterior of the MPC and to minimize surface contamination

The site of the ISFSI is located in the DCPP Facility, which precludes the need for spent nuclear fuel transport from the spent nuclear fuel pool to the ISFSI on public roads. The ISFSI storage pads are located in a cut into a hill that provides natural shielding on one side and partial shielding on two other sides and is located at a sufficient distance from the controlled-area fence to aid in minimizing offsite exposures.

The staff finds that the design of the Diablo Canyon ISFSI provides reasonable assurance that doses to personnel and members of the public will be maintained ALARA and meet the requirements of 10 CFR §72.126(a). The staff finds that the requirements of minimization of contamination and amount of generated radioactive waste outlined in 10 CFR §20.1406 are satisfied. The staff finds that the design of weld-sealed MPCs that are not opened at the ISFSI and that allow no effluents meets the requirements of 10 CFR §72.126(d).

11.1.1.3 Operational Considerations

The operational considerations to maintain doses ALARA are described in Section 7.1.3 of the SAR, "Operational Considerations." The operating procedures for the Diablo Canyon ISFSI, such as cask loading and unloading, transfer to the Cask Transfer Facility (CTF), MPC transfer, and transfer to the storage pad were developed in accordance with Regulatory Guides 8.8 (U.S. Nuclear Regulatory Commission, 1978) and 8.10 (U.S. Nuclear Regulatory Commission, 1975). Specifically, the program to maintain doses ALARA includes the following operational elements:

- Use of classroom training, mockups, and dry-run training to train personnel about canister transfer procedures, to verify equipment operability and procedure efficiency to minimize radiation exposure

- Fuel loading procedures will follow accepted work practices that reflect lessons learned about maintaining doses ALARA from other facilities that use dry cask storage

- Use of a regionalized MPC loading strategy of placing less radioactive assemblies on the periphery of the MPC basket and more radioactive assemblies in the center to take advantage of the spent nuclear fuel self shielding and minimize onsite and offsite dose rates from direct radiation

- Use of power-operated tools in bolting operations to minimize personnel exposure time

- Use of temporary portable shielding during fuel transfer to minimize personnel exposure to direct radiation

The staff finds that the PG&E description of the operational considerations for maintaining doses ALARA satisfies the requirements of 10 CFR §72.24(e) and that use of Regulatory Guides 8.8 (U.S. Nuclear Regulatory Commission, 1978) and 8.10 (U.S. Nuclear Regulatory Commission, 1975) in planning operations is appropriate and provides reasonable assurance that doses to personnel and members of the public will be maintained ALARA.

11.1.2 Radiation Protection Design Features

Information relevant to the proposed radiation design features of the ISFSI is contained in Section 7.3, "Radiation Protection Design Features," of the SAR.

11.1.2.1 Installation Design Features

The installation radiation protection design features are described in Section 7.3.1 of the SAR, "Installation Design Features." The ISFSI will be located within the DCPP controlled-site boundary with the nearest site boundary point 427 m [1,400 ft] away and the nearest residence 2.4 km [1.5 mi] from the front row of the stored casks on the pad. The ISFSI will be surrounded by a restricted area fence. Periodic inspections, placement of loaded storage casks, and routine security checks are the planned operations to be conducted at the ISFSI.

The HI-STORM 100 System will be used at the ISFSI. The major components of the system include a stainless steel cylindrical MPC confinement vessel, a steel-lined concrete storage cask, and a transfer cask that is a steel-lead-steel layered cylinder with a water jacket attached to the exterior.

The fuel is stored dry inside the MPC, so no radioactive liquid is available for leakage. Airborne radioactive materials will be prevented by the weld-sealed canisters, and fuel is not removed from the MPCs at either the ISFSI storage pads or the CTF. The storage system is passive and requires little maintenance. The system is not expected to leak during normal, off-normal, or accident conditions, and therefore, staff concludes that airborne radioactive monitors specified in 10 CFR §72.126 are not required at the ISFSI. Spacing of the storage casks on the storage pads provides self shielding for interior casks. Labyrinthine paths for the air inlets and outlets are used to minimize radiation streaming.

The staff finds that the use of Regulatory Position 2 of NRC Regulatory Guide 8.8 (U.S. Nuclear Regulatory Commission, 1978) that provides guidance regarding facility and equipment features is appropriate. The staff finds that given the proposed design features described in the SAR, PG&E has satisfied the requirements of 10 CFR §72.126(a) and §72.128(a)(2). The distance between the spent nuclear fuel handling and storage areas and the nearest boundary of the controlled area of the proposed ISFSI {427 m [1,400 ft]} meets the minimum distance specified in 10 CFR §72.106(b), which is 100 m [30 ft]. Sections 3.3.1.5, 3.3.1.5.2, 4.2.3.2, 4.2.3.3, 7.1.2, 7.3.1, 7.3.2, 7.3.3, and 7.3.4 of the SAR discuss design features that address radiation monitoring, control of airborne contaminants, instrumentation and controls, and other considerations related to maintaining doses ALARA.

The staff finds that the SAR provides reasonable assurance that the use of the HI-STORM 100 System can meet the regulatory requirements in 10 CFR Part 20, §72.104(a), §72.104(b), §72.104(c), and §72.106(b). Chapters 7, 9, and 15 of this Safety Evaluation Report (SER) discuss staff evaluations of the radiation shielding features, confinement features and their capabilities during off-normal and accident conditions. Based on these evaluations, the staff finds that the radiation protection features for the proposed Diablo Canyon ISFSI are acceptable.

11.1.2.2 Access Control

The access control to the ISFSI is described in Sections 2.2.2, "Site Description;" 3.3.1.5.1, "Access Control;" 4.1, "Location and Layout;" 7.6.2, "Equipment, Instrumentation, and Facilities;" and 9.6, "Physical Security Plan" of the SAR. Access control to the restricted area provides for both personnel radiation protection and stored fuel physical protection. There are two fences that will surround the Diablo Canyon ISFSI. A security fence, with a locked gate, will circumscribe the ISFSI storage pads and will serve as the protected-area boundary in compliance with the requirements of 10 CFR §73.55. There is a minimum 12-m [40-ft] distance between the storage casks and the security fence. A second fence will be built around the protected area that is approximately 30 m [100 ft] from the storage casks. The second fence serves as a restricted-area boundary in compliance with §72.106(a). The location of the second fence is to ensure that the dose rate at this boundary will not exceed 20 μSv/hr [2 mrem/hr] as specified in 10 CFR Part 20.

Once the Diablo Canyon ISFSI is operational, entrance to and work within the ISFSI protected area will be controlled by radiation protection and security personnel who will maintain a list of individuals authorized for access. During normal storage operations, personnel will conduct infrequent, short-duration checks on the material condition of the casks to ensure the overpack air ducts are free of blockage. Higher occupancy activities will occur during construction of new pads and placement of loaded overpacks. Radiation work permits will be required for authorized work. The ISFSI protected area will have an intrusion detection system to detect unauthorized entry. The controlled area within the owner-controlled area is surrounded by a farm-type fence with the nearest boundary to the ISFSI site 427 m [1,400 ft] away. The dose rate outside the controlled area will not exceed 250 μSv/yr [25 mrem/yr], as specified in 10 CFR §72.104(a).

The staff finds that the description of access control at the Diablo Canyon ISFSI is acceptable and meets the requirements of 10 CFR §72.126(a)(3). The physical security plan prevents the entry of unauthorized personnel into radiologically controlled areas. The ISFSI security program contains information that is to be withheld from the public in accordance with 10 CFR §2.790(d) and §73.21. The information withheld for security purposes is not included in the SAR and will be submitted as a separate document.

11.1.2.3 Radiation Shielding

The radiation shielding evaluation is contained in Chapter 7 of this SER. As stated in Chapter 7 of this SER, the staff evaluated the Diablo Canyon ISFSI SAR shielding calculation and found them to be acceptable. The dose rates at the onsite and offsite locations were found to be below the limits specified in 10 CFR §20.1201, §20.1301, §20.1302, and §72.104(a). The applicant description, combined with a sample input file in the HI-STORM 100 System FSAR Revision 1 (Holtec International, 2002), provides reasonable assurance that the ISFSI shielding was adequately evaluated. The applicant analysis demonstrates that no credible accident will cause a significant increase in public or personnel dose rates from direct radiation. This provides reasonable assurance that during accident conditions, dose rates from direct radiation will be below limits specified in 10 CFR §72.106(b).

11.1.2.4 Confinement and Ventilation

The evaluation of the MPC confinement system is provided in Chapter 9 of this SER. The MPC is a welded cylindrical enclosure with no mechanical joints or seals in the confinement boundary and is not vented.

11.1.2.5 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Area radiation and airborne radioactivity monitoring instrumentation is described in Sections 3.3.1.3.2, "Instrumentation;" 3.3.1.5.3, "Radiological Alarm Systems;" 6.2, "Radioactive Wastes;" and 7.3.4, "Area Radiation and Airborne Radioactivity Monitoring Instrumentation." All spent nuclear fuel at the Diablo Canyon ISFSI will be stored in weld-sealed MPCs. There are no credible events that could result in the release of radioactive materials from within MPCs, and there are no credible events that might increase dose rates from direct radiation from the casks. Therefore, area radiation and airborne radioactivity monitors are not needed at the Diablo Canyon ISFSI storage pads. Continuous monitoring and audible high-radiation level alarms will be available in the fuel handling building auxiliary building. Thermoluminescent dosimeters will be used to monitor and record area doses at appropriate intervals in all four directions around the ISFSI restricted-area fence. Hand-held radiation protection instruments and dosimeters will be provided during fuel transfer operations at the CTF and routine maintenance at the ISFSI storage pads. Before discharge, all water collected in the CTF sump will be sampled and, if contamination is found, it will be disposed of in accordance with the DCPP Radioactive Waste Management program.

The staff finds that the radiation monitoring instrumentation described in the SAR meets the requirements of 10 CFR §72.126(c) and provides reasonable assurance that actual dose rates around the ISFSI will be adequately monitored to verify compliance with the radiological limits specified in 10 CFR Parts 20 and 72 for members of the public, and any unexpected increases in dose rates will be properly detected in a timely manner.

11.1.3 Dose Assessment

The dose assessments are presented in Sections 7.3.2, "Shielding;" 7.4, "Estimated Collective Dose Assessments;" 7.5, "Offsite Collective Dose;" 8.1.3.5, "Radiological Impact of Confinement Boundary Leakage;" 8.1.4.5, "Radiological Impact of Partial Blockage of Air Inlets;" 8.1.6.5, "Radiological Impact of Loss of Electric Power;" 8.1.7.5, "Radiological Impact of Cask Transporter Off-Normal Operation;" 8.2.6.3, "Accident Dose Calculations;" 8.2.7.3, "Dose Calculations for Hypothetical Accident Conditions;" and 8.2.15.3, "Dose Calculations for 100 Percent Blockage of Air Inlet Ducts."

PG&E proposes to use the HI-STORM 100 System. This cask system consists of (1) interchangeable MPCs that contain the fuel; (2) a storage overpack that contains the MPC during storage; and (3) a transfer cask that contains the MPC during loading, unloading, and transfer operations. The system has been reviewed and approved by NRC under the general license provision of 10 CFR Part 72. Amendment 1 of the HI-STORM 100 System CoC became effective on July 15, 2002 (U.S. Nuclear Regulatory Commission, 2002a). The staff evaluations of the shielding, confinement, radiation protection evaluation, and dose assessments for the cask system are documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). The staff concluded that the design of radiation protection features in the

HI-STORM 100 System is sufficient to meet the radiation protection requirements of 10 CFR Part 20, §72.104, and §72.106.

PG&E calculated offsite dose rates for 140 casks located on the storage pads. The bounding dose analyses are conducted by assuming that overpacks loaded with identical MPC–32s are completely loaded with fuel assemblies having burnups of 32,500 MWd/MTU [2.42×10^{12} Btu/ton uranium] and cooling times varying from 5 to 20 years. A Babcock and Wilcox 15 × 15 fuel assembly, a design-basis assembly for the HI-STORM 100 System, was used in the Diablo Canyon ISFSI shielding analyses. Additional credit was taken in the dose analyses for longer cooling times as the casks are placed at the ISFSI at the rate of eight casks per year. The transfer cask dose analysis was performed for the MPC-24 using burnup and cooling time of 55,000 MWd/MTU [4.09×10^{12} Btu/ton uranium] and 12 years. These dose rates bound those for other MPC, burnup, and cooling time combinations.

As discussed in Section 7.1.4.2 of this SER, the contact surface dose rate for the HI-STAR transfer cask was estimated to be approximately 3,893 μSv/hr [389.3 mrem/hr] outside the lid in its center, and the average contact surface dose rate for the HI-STORM 100 System storage cask (side-dose value) was estimated to be approximately 348 μSv/hr [34.8 mrem/hr] at the midplane of the overpack. Based on these values, PG&E estimated the controlled-area boundary dose rate as 0.027 μSv/hr [0.0027 mrem/hr] that corresponds to 56.2 μSv/yr [5.62 mrem/yr] for 2,080-hour annual occupancy with 140 casks from direct and scattered radiation exposure. As discussed in Chapter 9 of this SER, no release of radioactive materials in effluent is expected during normal operations, therefore, the doses caused by effluents are not considered. PG&E estimated an annual direct and scattered radiation dose to the nearest resident located 2.4 km [1.5 mi] away from the ISFSI as 0.0035 μSv [0.00035 mrem], assuming the resident is continually present at the residency for 8,760 hours per year. These dose values are less than the 250 μSv/yr [25 mrem/year] whole-body dose limit specified in 10 CFR §72.104(a).

The estimated dose from loading, transport, and emplacement of a single MPC in a storage cask in the storage pad was estimated at 21 person-mSv [2.1 person-rem].

The annual occupational exposures from routine maintenance activities, such as cask inlet or outlet duct surveillance, concrete inspections, and radiation protection surveys, were evaluated to result in 18 person-mSv [1.8 person-rem] dose. The estimated annual exposure for overpack repair activities was estimated as 16 person-mSv [1.6 person-rem] and for construction of the last storage pad as 432 person-mSv [43.2 person-rem]. There is a reasonable assurance that the personnel exposures will be below the annual occupation dose limit of 50 mSv [5 rem] specified in 10 CFR §20.1201.

The staff evaluation of the controlled-area boundary and the nearest residence dose assessment and shielding evaluation of direct and scattered radiation doses is contained in Chapter 7 of this SER. As discussed in Chapter 9 of this SER, the doses caused by effluents were not considered. The use of dosimeters and periodic radiological surveillance at the ISFSI is expected to detect any unexpected significant releases of radioactive materials and, therefore, meet requirements outlined in 10 CFR §72.126(c)(2) and §20.1301.

Contributions to the dose rates from other nuclear fuel-cycle facilities located within a 5-km [3-mi] radius of the proposed ISFSI, which are the two units of the DCPP, were taken into account in the total off-site collective dose assessment. The annual dose from the other uranium fuel cycle operations was estimated as 0.44 μSv/yr [0.044 mrem/yr] at the controlled-site boundary and 0.44 μSv/yr [0.044 mrem/yr] to the nearest resident (Pacific Gas and Electric Company, 1996). The combined annual dose from the proposed ISFSI (including off-normal effluent release from a single cask) and other nuclear fuel-cycle facilities was estimated as 58.4 μSv [5.84 mrem] at the controlled-site boundary and 4.5 μSv [0.45 mrem] to the nearest resident. The total off-site collective dose assessment presented in the SAR provides reasonable assurance that the cumulative effects of the combined operations of the ISFSI and DCPP will not constitute an unreasonable risk to the health and safety of the public, in compliance with 10 CFR §72.104(a) and §72.122(e).

The staff finds the occupational and off-site dose assessments for the ISFSI acceptable. Results of these assessments and evaluations of the previously approved HI-STORM 100 System (U.S. Nuclear Regulatory Commission, 2002b) provide reasonable assurance that doses to personnel and members of the public will be maintained ALARA and will meet requirements of 10 CFR Part 20, §72.24(m), and §72.104(a).

11.1.4 Health Physics Program

The health physics program is described in Section 7.6, "Health Physics Program" of the SAR.

11.1.4.1 Organization

The health physics program organization is described in Section 7.6.1, of the SAR, "Organization" and references the health physics program described in the DCPP FSAR update (Pacific Gas and Electric Company, 2001), Section 12.3. The Radiation Protection Manager is responsible for health physics activities related to ISFSI operations for the life of the facility. The Radiation Protection Manager is independent of the Operations Manager. Once the storage site is operational, entrance to and work within the ISFSI protected area will be controlled by Radiation Protection and Security personnel. The staff finds that the proposed protection program satisfies 10 CFR §20.1101(a).

11.1.4.2 Equipment, Instrumentation, and Facilities

The equipment, instrumentation, and facilities pertinent to the ISFSI health physics program are described in Section 7.6.2, "Equipment, Instrumentation, and Facilities," of the SAR. The ISFSI is located within the Diablo Canyon owner-controlled area; PG&E has full authority to control all activities within the ISFSI and owner-controlled area boundaries. Many of the equipment, instrumentation, and facilities described in the DCPP FSAR update (Pacific Gas and Electric Company, 2001) will be used for ISFSI operations and radiation surveys.

Once the ISFSI is operational, entrance to and work within the Diablo Canyon ISFSI protected area will be controlled by Radiation Protection and Security personnel, and radiation work permits will be required for conducting activities within the ISFSI protected area. The available DCPP radiological instrumentation and equipment proposed for use at the ISFSI include:

- Personal monitoring equipment
- Portable radiation measuring instruments
- Portable air sampling equipment
- External dosimetry devices used for monitoring whole-body exposure, including thermoluminescent dosimeters and self-reading dosimeters
- Facilities for internal radiation monitoring
- Count room equipment
- Decontamination equipment and facilities

The DCPP Radiological Environmental Monitoring Program will be used for the ISFSI. Additional thermoluminescent dosimeters will be used to determine dose rates at restricted-area and controlled-area boundaries. There will be no additional effluent monitoring because no radioactive effluents are expected during normal ISFSI operations. The ultimate compliance with the requirements specified in 10 CFR §72.104(a) will be demonstrated through the DCPP environmental monitoring program.

The staff finds that the requirements of 10 CFR §20.1101(a) are met because the health physics equipment, instrumentation, and facilities proposed for use at the Diablo Canyon ISFSI are adequate to perform surveys of direct radiation, airborne radioactivity, and potential intakes by inhalation, ingestion, or absorption through skin or wounds.

11.1.4.3 Policies and Procedures

The equipment, instrumentation, and facilities pertinent to the ISFSI Health Physics Program are described in Section 7.6.3, "Policies and Procedures," of the SAR. The Health Physics Program at the Diablo Canyon ISFSI will be implemented in accordance with PG&E program directives, administrative procedures, and working-level procedures, which will be revised as needed to address ISFSI operations prior to operation of the ISFSI. The operation and use of radiation monitoring equipment will be described in written procedures.

The staff concludes that the description of the Health Physics Program satisfies the requirements of 10 CFR §20.1101(a), §20.1101(b), §20.1302(a), §20.1406, and §20.1501(a)(1).

11.2 Evaluation Findings

Based on the review of information in the SAR, the staff makes the following findings regarding the Radiation Protection Program of the Diablo Canyon ISFSI:

- The design and operating procedures of the Diablo Canyon ISFSI provide acceptable means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20 and for meeting the objective of maintaining exposures ALARA, in compliance with 10 CFR §72.24(e).

- The SAR and other documentation submitted in support of the application are acceptable and provide reasonable assurance that the activities authorized by the license can be conducted without endangering the health and safety of the public, in compliance with 10 CFR §72.40(a)(13).

- The proposed Diablo Canyon ISFSI is to be on the same site as DCPP. The cumulative effects of the combined operations of these facilities will not constitute an unreasonable risk to the health and safety of the public, in compliance with 10 CFR §72.122(e).

- The SAR provides analyses showing that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limits given in 10 CFR §72.104.

- The design of the Diablo Canyon ISFSI provides suitable shielding for radiation protection under normal and accident conditions, in compliance with 10 CFR §72.128(a)(2).

11.3 References

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Revision 1.* Volumes I and II. HI–5014464. Docket 72-1014. Marlton, NJ: Holtec International. 2002.

Pacific Gas and Electric Company. *Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update, Revision 11.* Avila Beach, CA: Pacific Gas and Electric Company. 1996.

Pacific Gas and Electric Company. *Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update, Revision 14.* Avila Beach, CA: Pacific Gas and Electric Company. November 2001.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation Safety Analysis Report, Amendment 1.* Docket No. 72-26. Avila Beach, CA: Pacific Gas and Electric Company. October 2002.

U.S. Nuclear Regulatory Commission. *Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable. Regulatory Guide 8.10. Revision 1-R.* Washington, DC: U.S. Nuclear Regulatory Commission. 1975.

U.S. Nuclear Regulatory Commission. *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable.* Regulatory Guide 8.8. Rev. 3. Washington, DC: U.S. Nuclear Regulatory Commission. 1978.

U.S. Nuclear Regulatory Commission. *10 CFR Part 72 Certificate of Compliance No.1014, Amendment No.1, for the Holtec International HI-STORM 100 Cask System.* Docket 72-1014. Washington, DC: Nuclear Regulatory Commission. July 15, 2002a.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Amendment No. 1, Safety Evaluation Report*. Docket 72-1014. Washington, DC: Nuclear Regulatory Commission. 2002b.

12 QUALITY ASSURANCE

To be provided by NRC

13 DECOMMISSIONING EVALUATION

To be provided by NRC

14 WASTE CONFINEMENT AND MANAGEMENT EVALUATION

14.1 Conduct of Review

This chapter of the Safety Evaluation Report (SER) evaluates the waste management systems of the Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI). Chapter 6 of the Safety Analysis Report (SAR) provides information about the waste confinement and disposal systems that are a part of the facility. This review specifically focused on radioactive wastes that would be generated by site activities involving the handling and storage of spent nuclear fuel. These activities may produce (1) gaseous wastes, (2) liquid wastes, and (3) solid or solidified wastes during loading and unloading of the multi-purpose cask (MPC). Neither the actual spent nuclear fuel nor the waste generated by the Diablo Canyon Power Plant (DCPP) at the ISFSI fall within the scope of this review. The review objectives for this chapter are to determine whether the ISFSI provides safe confinement and management of radioactive waste generated, and the generation of radioactive waste and release of the radioactive material to the environment meet the regulatory standards.

14.1.1 Waste Source

A review of the sources of radioactive waste described in Chapter 6 of the SAR included consideration of various sources during the operation of the facility. The review considered how the SAR addresses the following regulatory requirements:

- 10 CFR §72.24(l) requires that information regarding the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal operations and expected operational occurrences needs to be included in the SAR describing the proposed ISFSI.

- 10 CFR §72.104(a) requires that, 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 from various sources, including planned discharges of radioactive materials to the environment.

- 10 CFR §72.128(a)(5) requires that spent fuel storage and handling systems must be designed with means to minimize the quantity of radioactive wastes generated.

- 10 CFR §72.128(b) requires that radioactive waste treatment facilities must be provided. Provisions must be made for packing site-generated low-level wastes in a form suitable for storage onsite awaiting transfer to disposal sites.

As described in Chapter 6 of Volume 1 of the Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2002), some amounts of liquid, gaseous, and solid radioactive wastes may be generated during loading and decontamination activities before storage.

A small quantity of low-level solid waste may be generated during MPC loading operations and will be processed using the existing DCPP radioactive waste control systems as described in the 10 CFR Part 50 license and supporting documentation. Contaminated water from the loaded MPCs is drained back into the spent nuclear fuel pool with no additional processing. Liquid wastes are also generated from transfer cask and MPC decontamination, and any contaminated water collected in the cask transfer facility (CTF) sump. The decontamination procedure may result in a small amount of detergent or demineralized mixture being collected in the Fuel Handling Building Auxiliary Building (FHB/AB). This liquid waste would be processed using existing DCPP radioactive waste control systems as described in the 10 CFR Part 50 license and supporting documentation. Potentially contaminated air and helium from the MPC during loading and unloading operations will be collected and processed through the existing gaseous radioactive waste system which is covered under the 10 CFR Part 50 license and supporting documentation.

The staff concludes that use of Diablo Canyon ISFSI facilities for processing solid and liquid wastes generated during fuel loading and decontamination activities satisfies the requirements of 10 CFR §72.128(b). The passive design minimizes the volume of radioactive waste that could be generated by the operation of the ISFSI. The staff concludes that the Diablo Canyon ISFSI satisfies the requirements of 10 CFR §72.128(a)(5). The details provided in the SAR regarding the treatment of the solid, liquid, and gaseous wastes generated satisfy the requirements of 10 CFR §72.24(l).

During transport to the ISFSI pad and during storage at the ISFSI, no radioactive waste material is generated. The system is a passive design requiring no active systems to ensure adequate decay heat removal and to ensure adequate confinement. The system also does not require intrusive periodic maintenance. The passive design minimizes the volume of radioactive waste that could be generated by the operation of the ISFSI.

14.1.2 Off-Gas Treatment and Ventilation

The review of the SAR regarding off-gas treatment and ventilation considered how the SAR addresses the following regulatory requirements:

- 10 CFR §72.104(a) requires that, 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 from various sources, including planned discharges of radioactive materials to the environment.

- 10 CFR §72.122 (h)(3) requires that ventilation systems and off-gas systems must be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions.

- 10 CFR §72.126(d) requires that the ISFSI must be designed to provide means to limit to levels as low as is reasonably achievable the release of radioactive materials in effluents during normal operations; and control the release of radioactive materials under accident conditions. Analyses must be made to show that releases to the general environment during normal operations and

anticipated occurrences will be within the exposure limit given in §72.104. Analyses of design basis accidents must be made to show that releases to the general environment will be within the exposure limits given in §72.106. Systems designed to monitor the release of radioactive materials must have means for calibration and testing their operability.

As described in the Diablo Canyon ISFSI SAR, the MPC is designed to endure normal, off-normal and accident conditions of storage with maximum decay heat loads without loss of confinement. Permanent area radiation and airborne radioactivity monitors are not needed at the Diablo Canyon ISFSI because the storage system is passive. During fuel loading, existing spent fuel pool instrumentation will monitor for any releases of airborne radioactivity. These monitors are designed to automatically change the building ventilation exhaust system from normal to emergency mode upon detection of radiation levels above preset alarm levels. The MPC confinement boundary ensures that there will be no release of radioactive materials during fuel loading and closure of the MPC and potentially contaminated air will be collected and processed through the gaseous radioactive waste system. This contaminated vented gas would be redirected and processed using existing plant facilities and procedures subject to the requirements of the DCPP 10 CFR Part 50 license. There are no radioactive wastes created by the HI-STORM 100 System while in storage at the storage pads during transport to or from the CTF, or at the CTF.

The staff therefore concludes that the applicant has provided sufficient design features and controls to ensure the confinement of airborne radioactive particulate during normal and off-normal conditions in compliance with 10 CFR §72.122(h)(3). In addition, the staff concludes that the proposed design and operation of the ISFSI satisfies the requirements of 10 CFR §72.104(a) and §72.126(d). Because no effluents are expected under normal or accident conditions, the requirements of 10 CFR §72.126(c)(1), regarding measurement and dilution of effluents, are considered not applicable.

14.1.3 Liquid Waste Treatment and Retention

The review of the SAR regarding liquid waste treatment and retention considered how the SAR addresses the following regulatory requirements:

- 10 CFR §72.104(a) requires that, 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 from various sources, including planned discharges of radioactive materials to the environment.

- 10 CFR §72.128(b) requires that radioactive waste treatment facilities must be provided. Provisions must be made for packing site-generated low-level wastes in a form suitable for storage onsite awaiting transfer to disposal sites.

Contaminated water from the loaded MPCs is drained back into the spent nuclear fuel pool with no additional processing. Liquid wastes are also generated from transfer cask and MPC decontamination and any contaminated water collected in the CTF sump. The decontamination procedure may result in a small amount of a detergent or demineralized mixture being collected

in the FHB/AB. This liquid waste would be processed using existing DCPP radioactive waste control systems as described in the 10 CFR Part 50 license and supporting documentation.

The staff finds that no special liquid radioactive waste treatment and retention systems are needed at the ISFSI Facility. The applicant has identified and described an appropriate method for treating the contaminated liquids, should it be needed. Therefore, the staff finds that the requirements of 10 CFR §72.128(b) are satisfied. Use of the DCPP facility, subject to the provisions of 10 CFR Part 50, to process radioactive waste generated during all phases of ISFSI operation satisfies the requirements of 10 CFR §72.128(b).

14.1.4 Solid Wastes

The review of the SAR regarding off-gas treatment and ventilation considered how the SAR addresses the following regulatory requirements:

- 10 CFR §72.104(a) requires that, 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 from various sources, including planned discharges of radioactive materials to the environment.

- 10 CFR §72.128(b) requires that radioactive waste treatment facilities must be provided. Provisions must be made for the packing of site-generated low-level wastes in a form suitable for storage onsite awaiting transfer to disposal sites.

A small quantity of low-level solid waste may be generated during MPC loading operations. The solid waste may include disposable anticontamination garments, paper, rags, tools, and such, which will be processed using the existing DCPP radioactive waste control systems as described in the 10 CFR Part 50 license and supporting documentation. Use of the DCPP facility, subject to the provisions of 10 CFR Part 50, to process radioactive waste generated during all phases of ISFSI operation satisfies the requirements of 10 CFR §72.128(b) and §72.104(a).

14.1.5 Radiological Impact of Normal Operations

Based on the staff assessment of welded cask enclosures, as stated in NUREG–1536, "Standard Review Plan for Dry Cask Storage Systems," Chapter 7 Section V.2, the MPC, which is the confinement system for the HI-STORM 100 System, provides reasonable assurance that no effluents will be released during normal, off-normal, or accident conditions and, therefore, requires no monitoring of the MPC for leakage. The seal weld will be inspected and tested in accordance with the requirements in Section 8.1.5 of the HI-STORM 100 System Final Safety Analysis Report (FSAR). These requirements were reviewed during the certification of the HI-STORM 100 System and were found to be acceptable by the staff. The staff finds the applicant proposal to provide no monitoring of the confinement barrier for the HI-STORM 100 System acceptable because the casks will be loaded, welded, inspected, and tested in accordance with appropriate procedures.

Section 8.2.7 of the SAR (Pacific Gas and Electric Company, 2002) evaluates the potential consequences of leakage because of a confinement boundary accident. The potential consequences of this postulated accident are determined by assuming that 100 percent of the cladding for the fuel rods have ruptured and the MPC pressure boundary has been breached. The staff previously determined that the methodology used to assess this postulated accident is acceptable and that there are no consequences that affect the public health and safety based on licensee commitments regarding fuel specifications and loading conditions as defined in the HI-STORM 100 System Certificate of Compliance and SER (U.S. Nuclear Regulatory Commission, 2002a,b).

14.2 Evaluation Findings

The staff made the following findings regarding waste confinement and management of the Diablo Canyon ISFSI:

- The Diablo Canyon ISFSI SAR adequately describes acceptable features of the ISFSI design and operating modes that reduce, to the extent practical, the radioactive waste volume generated by the installation in compliance with 10 CFR §72.24(f) and 72.128(a)(5).

- Use of DCPP facilities for processing solid and liquid wastes generated during loading and decontamination activities conducted under the provisions of the DCPP 10 CFR Part 50 license satisfies the requirements of 10 CFR §72.128(b).

- The design of the ISFSI provides acceptable means to limit to levels as low as reasonably achievable the release of radioactive materials in effluents during normal operation and to control the release of radioactive materials under accident conditions in compliance with 10 CFR §72.126(d) and §72.104(a).

- The waste confinement and management activities described in the Diablo Canyon ISFSI SAR support a conclusion that the activities authorized by the license can be conducted without endangering the health and safety of the public in compliance with 10 CFR §72.40(a)(13).

14.3 References

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation Safety Analysis Report, Amendment 1.* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

U.S. Nuclear Regulatory Commission. *10 CFR Part 72 Certificate of Compliance No. 1014, Amendment 0, for the HI-STORM 100 Cask System.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2000a.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2000b.

15 ACCIDENT ANALYSIS

15.1 Conduct of Review

The staff evaluated the Pacific Gas and Electric (PG&E) Company accident analysis by reviewing Chapter 8, "Accident Analysis," of the Safety Analysis Report (SAR) (Pacific Gas and Electric Company, 2002), documents cited in the SAR, and other relevant publicly available information, including web sites on the Internet.

In response to Request for Additional Information (RAI) 15-1[1] PG&E described the basis for selecting off-normal and accident events to ensure all relevant potential scenarios have been considered. The selection of these off-normal and accident event scenarios is based on NUREG–1567 (U.S. Nuclear Regulatory Commission, 2000). In addition, PG&E also reviewed other site-specific applications and associated U.S. Nuclear Regulatory Commission (NRC) evaluations.

The dry-cask storage system to be used at the proposed facility is the HI-STORM 100 System, which has been reviewed by the NRC and approved for general use under Certificate of Compliance (CoC) No. 1014-1 (U.S. Nuclear Regulatory Commission, 2002a). As discussed in Chapters 4 and 5 of this Safety Evaluation Report (SER), the design-basis loads considered in the HI-STORM 100 System Final Safety Analysis Report (FSAR) bound the loading conditions at the proposed Independent Spent Fuel Storage Installation (ISFSI). Thus, where applicable, the staff relied on the review carried out during the certification process of the cask system, as documented in the NRC HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b).

The staff reviewed the accident analysis to determine if the following regulatory requirements have been met:

- 10 CFR §72.90 requires that: (a) site characteristics that may directly affect the safety or environmental impact of the ISFSI must be investigated and assessed; (b) proposed sites for the ISFSI must be examined with respect to the frequency and the severity of external natural and man-induced events that could affect the safe operation of the ISFSI; (c) design basis external events must be determined for each combination of proposed site and proposed ISFSI design; (d) proposed sites with design basis external events for which adequate protection cannot be provided through ISFSI design shall be deemed unsuitable for the location of the ISFSI; (e) pursuant to subpart A of part 51 of Title 10 for each proposed site for an ISFSI, the potential for radiological and other environmental impacts on the region must be evaluated with due consideration of the characteristics of the population, including its distribution, and of the regional environs, including its historical and esthetic values; and (f) the facility must be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of floodplains.

---

[1]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

- 10 CFR §72.92 requires that: (a) natural phenomena that may exist or that can occur in the region of a proposed site must be identified and assessed according to their potential effects on the safe operation of the ISFSI. The important natural phenomena that affect the ISFSI design must be identified; (b) records of the occurrence and severity of those important natural phenomena must be collected for the region and evaluated for reliability, accuracy, and completeness. The applicant shall retain these records until the license is issued; and (c) appropriate methods must be adopted for evaluating the design basis external natural events based on the characteristics of the region and the current state of knowledge about such events.

- 10 CFR §72.94 requires that: (a) the region must be examined for both past and present man-made facilities and activities that might endanger the proposed ISFSI. The important potential man-induced events that affect the ISFSI design must be identified; (b) information concerning the potential occurrence and severity of such events must be collected and evaluated for reliability, accuracy, and completeness; and (c) appropriate methods must be adopted for evaluating the design basis external man-induced events, based on the current state of knowledge about such events.

- 10 CFR §72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI be identified.

- 10 CFR §72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR §72.98(a) and §72.98(b) be investigated as appropriate with respect to: (1) the present and future character and the distribution of population, (2) consideration of present and projected future uses of land and water within the region, and (3) any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR §72.102(f)(1) requires that the design earthquake for use in the design of structures be determined as follows: (1) for sites that have been evaluated under the criteria of Appendix A of 10 CFR Part 100, the design earthquake must be equivalent to the safe shutdown earthquake for a nuclear power plant; and (2) Regardless of the results of the investigations anywhere in the continental U.S., the design earthquake must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum.

- 10 CFR §72.106(b) requires that any individual located on or beyond the nearest boundary of the controlled area 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 of the eye dose equivalent shall not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or to any extremity shall not exceed 0.5 Sv (50 rem). The minimum distance from the spent fuel or high-level radioactive waste handling

and storage facilities to the nearest boundary of the controlled area must be at least 100 meters.

- 10 CFR §72.122(b) requires that (1) structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents; and (2) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lighting, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect: (i) structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform their intended design functions. The design bases for these structures, systems, and components must reflect: (A) appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (B) appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (ii) The ISFSI also should be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel, high-level radioactive waste or on to structures, systems, and components important to safety.

- 10 CFR §72.122(c) requires that structures, systems, and components important to safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

- 10 CFR §72.122(h)(1) requires that the spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

- 10 CFR §72.122(h)(4) requires that storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe

storage conditions. For dry spent fuel storage, periodic monitoring is sufficient provided that periodic monitoring is consistent with the dry spent fuel storage cask design requirements. The monitoring period must be based upon the spent fuel storage cask design requirements.

- 10 CFR §72.122(h)(5) requires that the waste must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of part 20 limits. The package must be designed to confine the high-level radioactive waste for the duration of the license.

- 10 CFR §72.122(i) requires that instrumentation and control systems must be provided to monitor systems that are important to safety over anticipated ranges for normal operation and off-normal operation.

- 10 CFR §72.122(l) requires that Storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste for further processing or disposal.

- 10 CFR §72.124(a) requires spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions.

- 10 CFR §72.128(a)(2) requires that spent fuel storage be designed with suitable shielding for radioactive protection under normal and accident conditions.

The proposed ISFSI facility must be sited, designed, constructed, and operated so the above-mentioned regulatory requirements are met to adequately protect public health and safety during all credible off-normal and accident events.

15.1.1 Off-Normal Events

The off-normal events are described in Section 8.1, "Off-Normal Operations," of the SAR. This section of the SER discusses results from the review of potential off-normal conditions, which are cask drop, partial vent blockage, and operational events. The cask system to be used at the proposed facility is the HI-STORM 100 System. Where applicable, the staff relied on the analyses in the HI-STORM 100 System FSAR and the related staff evaluation as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b).

15.1.1.1 Cask Drop Less Than Design Allowable Height

A potential drop of the HI-TRAC 125 Transfer Cask can only occur during transport from the Diablo Canyon Power Plant (DCPP) Fuel Handling Building and Auxiliary Building (FHB/AB) to the Cask Transfer Facility (CTF). Similarly, a drop of the HI-STORM 100SA storage cask can only occur during transport between the CTF and the dry storage area. In response to RAI 15–3[2] PG&E has committed to design the cask transporter so it will have redundant drop protection features and will conform to the requirements of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980), American National Standards Institute (ANSI) N14.6 (American National Standards Institute, 1993), and ASME B30.9-1996 (ASME International, 1996). The staff previously determined that the casks can be lifted to any height necessary during transportation between the FHB/AB and CTF (U.S. Nuclear Regulatory Commission, 2002a), if these cask transporter design requirements are met.

15.1.1.2 Partial Vent Blockage

The staff previously determined (U.S. Nuclear Regulatory Commission, 2002b) that the HI-STORM 100SA storage cask provides adequate heat removal capacity under partial vent blockage conditions so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to and the environmental characteristics of the site are bounded by the corresponding design criteria (see Section 6.1.3 of this SER). The HI-STORM 100 System CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix A) includes surveillance requirements for ensuring that the cask heat removal system is operational during storage (i.e., the air ducts are inspected or the temperature differential between the convected cooling air exiting the outlet vents and ambient air is measured every 24 hours to ensure that the ducts are free of blockages).

15.1.1.3 Operational Events

Failure of Instrumentation

No off-normal events that involve failure of instruments and control systems are postulated because there would be no permanent instruments to monitor the heat and radiation at the ISFSI storage pad site. The HI-STORM 100SA storage cask will be periodically inspected visually to ensure that the overpack inlet and outlet air ducts remain free from blockages. If a blockage is detected, it will be removed within one operating shift. Radiation and airborne radioactivity will be monitored using portable hand-held radiation protection instruments and dosimeters during transfer operations at the CTF and routine maintenance at the ISFSI storage area.

The review of the information provided regarding failure of instrumentation gave reasonable assurance that important to safety functions will not be affected for the proposed cask system or the proposed facility.

---

[2]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

Vehicular Impact

The staff reviewed the information presented in SAR Chapters 3, "Principal Design Criteria;" and 4, "ISFSI Design;" and Section 8.2.4, "Drops and Tip-Over." The applicant does not directly address vehicular impact. Vehicular impact is postulated by the staff to occur during transport from the FHB/AB to the CTF, during transport from the CTF to the storage pads, or in the storage pad area. Vehicular impacts are postulated to result from an interaction between a transportation cask, storage cask, cask transporter, onsite vehicle, or a vehicle used by site personnel. Equipment failure, operator error, or a natural event (e.g., tornado) may lead to this off-normal event. Occurrence of this event is easily identifiable from visual evidence, such as dents or scratches on casks, onsite vehicles, and other ISFSI facility structures, systems, and components (SSC).

As discussed in the HI-STORM 100 System FSAR, the HI-STORM 100SA storage cask and HI-TRAC 125 Transfer Cask are designed to withstand a tornado missile equivalent to the impact of an automobile weighing 1,800 kg [3,000 lb] traveling at a speed of 202 km/h [56 m/s] {126 mph [185 ft/s]} (SAR Table 3.2-2). This tornado-missile analysis for the storage cask and the staff evaluation are provided in the HI-STORM 100 System FSAR and the related NRC SER. Onsite vehicles will generally be traveling at a much lower speed, approximately 24 km/h [15 mph]. Therefore, vehicular impact for the HI-STORM 100 System is bounded by the tornado missile analysis.

The cask transporter and CTF are designed to withstand a tornado missile equivalent to the impact of an automobile weighing 1,800 kg [3,968 lb] traveling at a speed of 15 m/s {48.8 ft/s (33.3 mph)} (SAR Table 3.2-2) (Pacific Gas and Electric Company, 2002). The tornado missile analysis and the staff evaluation are provided in the SAR Sections 8.2.2 and 15.1.2.10 of this SER. Onsite vehicles will generally be traveling at a much lower speed, on the order of 24 km/h [15 mph]. Therefore, vehicular impact for the cask transporter and CTF is bounded by the tornado missile analysis.

The staff finds that potential vehicular impact will not impair the ability of the SSC to maintain subcriticality, confinement, and sufficient shielding of the stored fuel.

Loss of Electrical Power

The staff reviewed the information presented in Section 8.1.6, "Loss of Electrical Power," of the SAR (Pacific Gas and Electric Company, 2002) as an off-normal event. Total loss of external alternating current power is postulated to occur during the facility operations. The loss of electrical power at the Diablo Canyon ISFSI facility may occur because of natural phenomena, such as lightning, high wind, and such, or as a result of failure of the electrical distribution system or equipment. A loss of electrical power will be detected through loss of functions of the electric-powered equipment.

No safety features required for lifting, upending, and lowering of HI-TRAC 125 Transfer Cask, multi-purpose canister (MPC) and HI-STORM 100SA storage cask at the CTF will be affected by the loss of power because these operations will be conducted by the cask transporter, which is driven by an on-board diesel engine. Similarly, the emplacement operations of the HI-STORM 100SA storage cask in the ISFSI storage pad site are also conducted using the cask transporter and do not involve use of electric power.

Electric power is supplied to each of the three lifting screw jack motors and control systems that operate the CTF lifting platform. The CTF lifting platform will raise and lower the MPC during the transfer operation of the MPC from the HI-TRAC 125 Transfer Cask to the HI-STORM 100SA storage cask. In the event of a power loss during the operations of the lifting platform, all three screw jack motors would stop simultaneously to prevent a potential uncontrolled descent of the storage cask inside the CTF. The lift jacks will remain stopped and will require manual action to restart upon restoration of power. For an extended period of power loss, the storage cask (including the MPC) would be raised to grade level from the CTF lifting platform within 22 hours using the cask transporter to ensure that short-term cladding temperature is not exceeded.

No radiological impact because of loss of electric power is expected because there is no loss of MPC confinement during this off-normal event. In addition, the transfer cask is designed to provide adequate shielding and decay heat removal from the canisters. The operators would take measures to maintain adequate distance and additional shielding between themselves and the CTF to minimize exposure until power is restored and the transfer operation is resumed.

The staff concludes the applicant evaluation of loss of electrical power as an off-normal event is adequate in providing assurance that Diablo Canyon ISFSI operations can be conducted without endangering the health and safety of the public.

Cask Transporter Off-Normal Operation

Staff reviewed the information provided in Section 8.1.7, "Cask Transporter Off-Normal Operation" (Pacific Gas and Electric Company, 2002). The transporter with a loaded cask would travel a distance of 1.9 km [1.2 mi] along the transporter route from the DCPP to the CTF and would take approximately 3.0 hours per transport. The transporter is also used in the transfer operation of an MPC from a transportation cask to a storage cask at the CTF site and in the emplacement of storage casks on the ISFSI pads. The off-normal events from operation of the cask transporter could arise from driver error or incapacitation, transporter engine failure because of mechanical failure, and loss of hydraulic fluid in the hydraulic system. A support team would walk with the transporter and observe the driver and transporter movement. At the sight of driver distress or swerving of the transporter, the support personnel can stop the transporter using either of two stop switches located outside the transporter. The transporter is also equipped with automatic shutoff control to stop the vehicle in the event of incapacitation of the driver. The same control will also be used for emergency stops during the lifting operation at the CTF. Transporter engine failure would stop the vehicle or hydraulic brakes would engage to stop lifting operations. Hydraulic system failure would be detected by pressure instrumentation on the transporter, and any loss of hydraulic fluid will engage hydraulic brakes to stop lifting operations. The transporter is designed to operate in a "fail-safe" mode so any uncontrolled lowering of a transfer cask loaded with an MPC or storage cask is precluded.

Off-normal events associated with cask transporter operation are not expected to cause radiological dose as the confinement and shielding of spent nuclear fuel will not be affected.

The staff concludes the applicant assessment of cask transporter off-normal operation is adequate in providing assurance that Diablo Canyon ISFSI operations can be conducted without endangering the health and safety of the public.

15.1.1.4 Off-Normal Ambient Temperatures

The off-normal environmental temperature range for the Diablo Canyon ISFSI is -4.4 to 36.1 °C [24 to 97 °F]. This off-normal temperature range is bounded by the previously evaluated off-normal temperature ranges for the HI-STORM 100SA storage and HI-TRAC 125 Transfer Casks. Specifically, the previously evaluated off-normal temperature range for the HI-STORM 100SA storage cask is -40 to 38 °C [-40 to 100 °F] and for the HI-TRAC 125 Transfer Cask, -18 to 38 °C [0 to 100 °F]. The staff previously determined that the HI-STORM 100SA storage and HI-TRAC 125 Transfer Casks provide adequate heat removal capacity during off-normal ambient temperature conditions so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to.

15.1.1.5 Off-Normal Pressures

Section 8.1.1.1 of the Diablo Canyon ISFSI SAR indicates that the off-normal pressure within the MPC, which is the sole pressure boundary for the HI-STORM 100SA storage cask, is evaluated considering a concurrent rupture of 10 percent of the stored fuel rods while exposed to off-normal ambient temperatures of 38 °C [100 °F]. Note that this off-normal temperature bounds the off-normal temperature for the proposed Diablo Canyon site (see Section 6.1.3 of this SER). The staff previously determined that the methodology used to assess this off-normal condition is acceptable and that there are no consequences that affect the public health and safety so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to.

15.1.2 Accidents

The SAR includes a discussion of potential accidents resulting from both external natural and man-induced events at the proposed facility. Natural phenomena events are discussed in Chapter 2, "Site Characteristics" of the SAR. The staff evaluation is discussed in Chapter 2 of this SER. The accident analysis review focused on the effects of the natural phenomena and human-induced events on SSC important to safety. Analytical techniques, uncertainties, and assumptions were examined. Each event was examined to ensure that it includes: (1) a discussion of the cause of the event, (2) the means of detection of the event, (3) an analysis of the consequences and the protection provided by devices or systems designed to limit the extent of the consequences, and (4) any actions required of the operator.

The proposed facility will use the HI-STORM 100 System. Where applicable, the staff relied on the analyses in the HI-STORM 100 System FSAR and the related staff evaluation as documented in the HI-STORM 100 System SER.

15.1.2.1 Cask Tip-Over

The staff has previously determined that cask tip-over events need not be considered if the preapproved HI-STORM 100SA storage cask anchorage system is used and the storage pad design specifications are met (U.S. Nuclear Regulatory Commission, 2002a,b). The Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2002) indicates that the anchored HI-STORM 100SA storage cask will be used. The evaluation of the storage pad and anchorage system design can be found in Section 5.1.2 of this SER.

15.1.2.2 Cask Drop

A potential drop of the HI-TRAC 125 Transfer Cask can only occur during transport from the DCPP FHB/AB to the CTF. Similarly, a drop of the HI-STORM 100SA storage cask can only occur during transport between the CTF and the dry storage area. In response to RAI 15-3[3], PG&E committed to design the cask transporter so it will have redundant drop protection features and conform to the requirements of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980), ANSI N14.6 (American National Standards Institute, 1993), and ASME B30.9-1996 (ASME International, 1996). The staff has previously determined that the casks can be lifted to any height necessary during transportation between the FHB/AB and CTF (U.S. Nuclear Regulatory Commission, 2002a), if these cask transporter design parameters are met.

15.1.2.3 Flood

The applicant has not considered flooding a credible accident at the Diablo Canyon ISFSI. As discussed in Section 2.1.4, "Surface Hydrology," of this SER, PG&E demonstrated that local natural and man-made drainage systems are sufficient to prevent flooding of the ISFSI pad site and CTF.

15.1.2.4 Fire and Explosion

Fire

The staff reviewed the information presented in Section 8.2.5, "Fire," of the SAR. Additional information presented in SAR Sections 4.2.3.3.2.10, "Fire;" and 4.2.3.3.2.11, "Lightning," was also used in this review.

Locations pertaining to the proposed ISFSI that fall within the purview of 10 CFR Part 72 review are the transport route from the DCPP FHB/AB to the CTF, within the CTF, and withing the cask storage area. Credible fire accidents potentially affecting SSC important to safety at the proposed facility identified by PG&E (Pacific Gas and Electric Company, 2002) are

(1) An onsite cask transporter fuel tank fire

(2) Other onsite vehicle fuel tank fires

(3) Combustion of other local stationary fuel tanks

(4) Combustion of other local combustible materials

(5) Fire in the surrounding vegetation

---

[3]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

Additional information and evaluation are provided in Section 6.1.5.1, "Fire," of this SER.

The cask transporter will be used to move the spent nuclear fuel in MPC from the FHB/AB to the CTF using the HI-TRAC 125 Transfer Cask. After the MPC has been transferred to the HI-STORM 100SA storage cask at the CTF, the cask transporter will be used to move the storage cask onto the storage pad. To limit the potential exposure of the HI-TRAC 125 Transfer Cask and HI-STORM 100SA storage casks to a fire attributable to the transporter diesel fuel, the fuel tank used for the transporter is limited to a 189-L [50-gal] capacity.

One potential fire at the CTF and the storage pads involves a diesel-fueled cask transporter with a 189-L [50-gal] fuel tank. The tank may rupture, resulting in all 189 L [50 gal] of diesel fuel being spilled. The spilled diesel fuel may be ignited. The ability of the HI-TRAC 125 Transfer Cask and HI-STORM 100SA storage casks to provide confinement and protect the spent nuclear fuel from gross degradation as the result of a 189-L [50-gal] diesel fuel fire was previously reviewed and found to be acceptable by the staff (U.S. Nuclear Regulatory Commission, 2002a,b).

Administrative controls will be implemented to ensure that transient sources of fuel in volumes larger than 189 L [50 gal] will be at a sufficient distance from the dry storage area pads at all times, the CTF during active MPC transfer operations, and the transport route during cask transport (Pacific Gas and Electric Company, 2002, Section 8.2.5.2). There is at least a 30.5-m [100-ft] clearance between the storage area, CTF, and the cask transport route, and any onsite stationary fuel tanks (Pacific Gas and Electric Company, 2002, Section 2.2.2.2).

According to the SAR (Pacific Gas and Electric Company, 2002, Section 8.2.6), a 3,028-L [800-gal] gasoline tanker truck will use the transport route near the storage area to deliver fuel to the vehicle maintenance shop located approximately 610 m [2,000 ft] northeast of the storage area six times a week. The tanker truck transport route passes within 15.2 m [50 ft] of the storage casks on the north side of the proposed dry storage area (Pacific Gas and Electric Company, 2002, Section 2.2.2.3). To determine the potential consequences of a gasoline tanker truck fire occurring near the proposed storage facility, a bounding 7,570-L [2,000-gal] fire loading analysis was conducted[4] to assess the potential effects on the HI-TRAC 125 Transfer Cask, which bounds the potential effects on a HI-STORM 100SA storage cask. This fire loading analysis adequately demonstrated that a nonengulfing 7,570-L [2,000-gal] fuel tanker fire will not adversely affect the HI-TRAC 125 Transfer Cask or HI-STORM 100SA storage cask.

Onsite stationary fuel sources include

(1) Three fuel tanks {946 L [250 gal] of propane, 7,571 L [2,000 gal] of No. 2 diesel, and 11,356 L [3,000 gal] of gasoline} located beside the main plant road, 366 m [1,200 ft] from the cask transport route at its nearest point

(2) Unit 2 main bank transformer mineral oil storage tank

(3) Gas cylinders

---

[4]Womack, L.F. Letter (March 27) DIL–03–005, Attachment 1, Calculation No. —1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

(4) Bulk hydrogen storage facility

(5) Cold machine shop acetylene bottles

(6) Electrical transformer fire

No combustible materials will be stored within the confines of the storage area demarcated by the security fence. The separation distance between the three stationary fuel tanks and the transport route is 366 m [1,200 ft][5]. Because of the separation distance, radiation is the only mechanism through which released heat would be transferred to the cask. The surface area of a hemisphere with a 366-m [1,200-ft] radius is in excess of 836,131 m^2 [9×10^6 ft^2]. The projected area of the cask is approximately 20 m^2 [220 ft^2]. Therefore, only 0.0025 percent of the total heat energy released simultaneously from these tanks would be directed toward a single cask. This is a small amount of energy, and consequently, a fire in the transporter fuel tank would be bounding[6].

The potential for a fire within the CTF as the result of a cask transporter or gasoline tanker truck fuel spill was addressed in response to a RAI[7]. To mitigate the potential for these postulated fire events, the CTF opening will be located at a higher elevation than the surrounding area so any fuel spilled will flow away from the facility. Moreover, administrative controls will prohibit any transient fuel sources beyond that of the cask transporter from coming into close proximity of the CTF during transfer operations.

Vegetation surrounding the storage pad area is primarily grass with no significant brush or trees (Pacific Gas and Electric Company, 2002). A potential fire in the vegetation may be started by an offsite fire spreading onto the proposed site or by a lightning or a transmission line strike. As discussed in Section 8.2.5.2 of the SAR, "Accident Analysis" (Pacific Gas and Electric Company, 2002), no combustible materials will be stored within the security fence of the proposed facility at any time. A walk-down of the general area and the transport route will be conducted prior to any loaded cask transport to ensure that all combustible materials are controlled according to the administrative procedures. PG&E will implement a maintenance program to prevent uncontrolled growth of vegetation surrounding the storage area.

PG&E submitted an analysis of potential effects of wildfires on the HI-STORM 100SA storage casks (Holtec International, 2001a). This analysis evaluated two scenarios: (1) no wind and (2) 24-km/hr [15-mph] wind in the uphill direction. Although it is expected that facility personnel will try to suppress or control the fire quickly, it is postulated that no fire fighting activities occur. Using simulation codes FARSITE and FLAMMAP, Holtec International (2001a) developed the

---

[5]Womack, L.F. Letter (March 27) DIL–03–005, Attachment 1, Calculation No. —1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[6]Ibid.

[7]Womack, L.F. Letter (July 28) DIL–03–010, Enclosure 2, Calculation No. —1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

values for the parameters necessary to describe the wildfire characteristics (namely, fire intensity, rate of spread, and flame length).[8]

There will be a minimum of a 15.2-m [50-ft] gap between the storage pads and the security fence on the north side of the proposed facility. The gap will be at least 12.2 m [40 ft] on the other three sides. The restricted area fence surrounds the area protected by the security fence and is approximately 30.5 m [100 ft] from the storage pads. Holtec International (2001a) assumed that the area within the proposed storage facility nuisance fence would be covered with either gravel or concrete. Therefore, the area surrounding the storage pads would be covered with noncombustible materials, which will not only act as a barrier for progression of wild fires but also will not add any additional fuel to the fire.

Electrical transformers are located approximately 73 m [240 ft] from the transporter route. The mineral oil within these transformers could be ignited by lightning strike, vehicle crash, or internal electrical faults.[9] Administrative procedures would prohibit movement of the loaded transporter during inclement weather. Additionally, DCPP transition operations significantly reduce the potential for transformer mineral oil being ignited by lightning or internal electric faults. Each active transformer has a fire-suppression system that will activate in case of a fire. Administrative procedures will also prohibit use of onsite vehicles during transporter operation negating the potential of a vehicle accident initiating a transformer fire. Moreover, even if a transformer mineral oil fire were to occur, its effect on the transfer cask during transport would be bounded by the nonengulfing 7,570-L [2,000-gal] fire-loading analysis.

The staff reviewed the information provided by the applicant regarding potential wildfires and onsite fires at the proposed facility. The staff found the applicant analysis acceptable because

- Through design and administrative procedures, sources of ignition (e.g., spill of diesel fuel) will be kept out of the CTF.
- The storage casks are designed to withstand a fire from 50 gallons of diesel fuel in the fuel tank of the cask transporters.
- Both the transfer and storage casks will be able to withstand a nonengulfing 7,570-L [2,000-gal] fuel fire.
- Adequate analysis was presented about potential effects of the tanker truck in storage casks sitting on the pads.
- The area surrounding the storage pads would be covered with noncombustible materials.

---

[8]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

[9]Womack, L.F. Letter (March 27) DIL–03–010, Enclosure 2, Calculation No. —1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

Onsite and Offsite Explosion

The staff has reviewed the information presented in SAR Sections 2.2.2.3, "Onsite Explosion Hazards"; 8.2.6, "Explosion"; and 3.3.1.6, "Fire and Explosion Protection." In addition, the staff also reviewed analyses of potential explosion events in Holtec International (2002) and PG&E Calculation File (Afzali, 2002). Potential sources of explosions within the proposed facility include

(1) Detonation of a transporter or onsite vehicle fuel tank

(2) Detonation of a 3,028-L [800-gal] tanker truck while transporting fuel near the storage pad

(3) Detonation of a propane bottle transported past the ISFSI storage pad

(4) Detonation of an acetylene bottle transported past the ISFSI storage pad

(5) Explosive decompression of a compressed gas cylinder

(6) Detonation of large stationary fuel tanks in the vicinity of the transport route

(7) Detonation of mineral oil from the DCPP Unit 2 main bank of transformers

(8) Detonation of the bulk hydrogen storage facility

(9) Detonation of acetylene bottles stored on the east side of the cold machine shop

Important to SSC that are required to function after an explosion event include the storage casks, the transportation casks, the transporters, and the CTF. Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) provides an acceptable methodology to estimate the minimum separation distance between an explosion source and a structure so that the peak positive incident overpressure would be less than 6.9 KPa [1 psi]. If the minimum separation distances calculated by following the suggested methodology of Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) are not sufficiently large to allow a conclusion that the peak positive incident overpressure would be less than 6.9 kPa [1 psi], an analysis of the frequency of hazardous materials shipment may be used to show the associated risk is sufficiently low. If the hazardous materials are shipped by more than one transportation mode, the frequency of exposure for the modes should be summed. Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) also states that potential explosion hazards can be screened out if, based on realistic or best estimate bases, an exposure rate less than 10^{-7} per year can be demonstrated. If conservative estimates are used, an exposure rate less than 10^{-6} per year is sufficiently low.

Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) sets 6.9 kPa [1 psi] as the peak positive incident overpressure below which no significant damage to the structures would be expected to result from an explosion. Explosion-induced ground motions are bounded by the earthquake criteria. Similarly, effects of explosion-generated missiles would be bounded by those associated with the air overpressure levels if the threshold air overpressure from any

explosion source is kept below 6.9 kPa [1 psi], based on Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978).

A potential explosion event can affect (1) canister transfer operation at the CTF, (2) storage casks placed on the pads, and (3) transfer casks while being transported by the transporter from the FHB/AB of the power plant to the proposed facility. Potential sources of explosive materials that may affect the storage casks and the canister transfer operation are (1) detonation of a transporter and onsite vehicle fuel tank, (2) detonation of 3,028-L [800-gal] tanker truck while transporting gasoline past the ISFSI storage pads, (3) detonation of a propane bottle transported past the ISFSI storage pads, (4) detonation of an acetylene bottle transported past the ISFSI storage pads, (5) detonation of large stationary fuel tanks, and (6) an explosive decompression of a compressed gas cylinder. Other sources are far away from the proposed storage site and contain sufficiently small amounts of explosive materials to pose a credible hazard to the storage casks and canister transfer operations. A transfer cask loaded on a transporter could be affected by (1) detonation of the fuel tank of a transporter or an onsite vehicle (including the potential explosion of a parked vehicle fuel tank), (2) explosion of large stationary fuel tanks in the vicinity of the transport route, (3) explosion of mineral oil from the DCPP Unit 2 main bank of transformers, (4) explosion of the Bulk Hydrogen Storage Facility, and (5) explosion of acetylene bottles stored on the east side of the cold machine shop.

Transporter and/or Onsite Vehicle Fuel Tanks

Potential sources of explosion are the fuel tanks of onsite transporters and other onsite vehicles, including 3,028-L [800-gal] gasoline tanker trucks (Pacific Gas and Electric Company, 2002). The maximum capacity of the fuel tanks of the onsite transporters is 189 L [50 gal] of diesel. The average capacity of the fuel tank of any onsite vehicle is 76 L [20 gal].[10] Additionally, a 3,028-L [800-gal] capacity gasoline tanker truck will use the transport route near the storage pads on its way to and from the maintenance shop, located approximately 666 m [2,000 ft] northeast of the storage pads. In addition, PG&E will prevent the 15,142-L [4,000-gal] truck from passing near the proposed storage facility and will not allow it to enter the owner-controlled area while spent nuclear fuel is being transported (Afzali, 2002).

Detonation of the fuel tank of a transporter and/or an onsite vehicle can occur near the storage pads, CTF, and transport route. This event can potentially affect the storage cask, the transfer cask, and the structure of the CTF.

PG&E (2002) assumed that a minimum distance of 15 m [50 ft] will be maintained between the source of explosion and the nearest storage cask because

- No gasoline-powered vehicles will be allowed within the restricted area of the proposed facility
- A minimum distance will be maintained between the storage casks and the protected area fence at the north side of the proposed facility

[10]Womack, L.F. Letter (March 27) DIL–03–010, Enclosure 2, Calculation No. —1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

The flash point of diesel fuel is 51.7 °C [125 °F]. Based on the Fire Protection Association Handbook (National Fire Protection Association, 1997), the flash point of a liquid must be less than 37.8 °C [100 °F] to be classified as a flammable liquid. Therefore, diesel in the fuel tank of a transporter does not pose a credible explosion hazard.

Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) provides a methodology to estimate the exposure rate *r*:

$$r = n \cdot f \cdot s \tag{15-1}$$

where,

n	–	explosion rate (per mile)
f	–	frequency of shipment (per year)
s	–	exposure distance (miles)

Based on the National Highway Traffic Safety Administration of the U.S. Department of Transportation (2003a), a total of 6,323,000 crashes involving all types of motor vehicles took place in 2001. Additionally, approximately 4,450,339 million km [2,781,462 million mi] were traveled in that year by all types of vehicles. Therefore, the vehicle involvement rate would be 227 per 160 million km [100 million mi] of travel. Based on 2001 crash statistics compiled by U.S. Department of Transportation (2003a), approximately 30 percent of all vehicle crashes constitute a single-vehicle crash. Additionally, approximately 30 percent of all single-vehicle crashes took place at a speed below 48 km/hr [30 mph] (U.S. Department of Transportation, 2003a). Moreover, approximately 0.1 percent of all vehicle crashes resulted in a fire (U.S. Department of Transportation, 2003a).

PG&E, by Administrative Control, will prevent any vehicle from passing another within the setback distance of 52.5 m [175 ft] from the proposed facility.[11] Consequently, only a single-vehicle accident needs to be considered further. This setback distance was selected so that the resulting air overpressure from an exploding 76-L [20-gal] gasoline tank would be 6.9 kPa [1 psi]. Additionally, PG&E will use administrative controls to prevent any motor vehicles from exceeding the speed limit of 40 km/hr [25 mph] in the area of the proposed facility (Assumption 7).[12] Therefore, the frequency of vehicle fire has been estimated by PG&E to 3.26×10^{-10} per km [2.04×10^{-10} per mi]. Assuming conservatively that every vehicle fire leads to an explosion, the explosion rate of vehicle fire, *n*, would be 3.26×10^{-10} per km [2.04×10^{-10} per mi].

The exposure distance, *s*, is the distance along the road within the setback region of the proposed facility from which the storage casks would have the potential to receive an air overpressure greater than 6.9 kPa [1 psi]. This distance is estimated to be approximately 90 m [300 ft].[13] As stated by PG&E (Assumption 10),[14] a maximum of 140 gasoline-powered vehicles

---

[11]Womack, L.F. Letter (March 28) DIL–03–010, Enclosure 2, Calculation No. —1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[12]Ibid.

[13]Ibid.

[14]Ibid.

would pass by the proposed facility in a day. Consequently, approximately 51,100 times in a year all items important to safety at the proposed facility would be exposed to the explosion hazard from passing gasoline-powered motor vehicles. Therefore, the annual frequency of exposure, *r*, is

$$r = 2.04 \times 10^{-10} \times 51100 \times \frac{300}{5280} = 5.92 \times 10^{-7} \text{ per year} \qquad (15\text{-}2)$$

The staff concludes that the annual frequency of occurrence of a transporter and/or onsite vehicle fuel tank explosion was estimated in a conservative manner.

Parked Vehicle Fuel Tanks

PG&E has used probabilistic analysis to estimate the annual frequency of explosion of an on-site vehicle, parked in the power plant parking lots, that may have a potential to damage a transfer cask being hauled by the transporter on the transport route. Since the start of construction of DCPP 30 years ago, there has never been an explosion of a parked car, although one parked car caught fire. PG&E[15] considers this an incredible scenario as, by administrative procedures, walk-downs of the parking lots would be performed looking for any explosion hazards, such as gasoline leaking from a vehicle, before a loaded transporter passes by. Additionally, administrative and physical controls would prevent movement of any vehicle within 52.5 m [175 ft] of the transporter.

PG&E[16] conducted a search for industry information regarding the frequency of explosion of parked vehicles; however, no data have been found. Although administrative and physical controls would make an explosion of a parked car an incredible scenario, nevertheless, PG&E[17] conducted an analysis to estimate the magnitude of the potential hazard if any. Analysis of gasoline-powered moving vehicles estimated the frequency of fire (an explosion) to be 3.26×10^{-10} per km [2.04×10^{-10} per mi] based on a single-vehicle crash. Since the cars, parked within 52.5 m [175 ft] of the moving loaded transporter, would not be allowed to move, reduction of one order of magnitude in the explosion rate to 3.26×10^{-11} per km [2.04×10^{-11} per mi] would be reasonable.

The transporter carrying a HI-TRAC Transfer Cask will make eight trips per year from the protected area of the power plant to the proposed storage facility. Therefore, frequency, *f*, would be 8/yr. The exposure distance, *s*, is estimated to be 333 m [1,000 ft]. Assuming a maximum of 200 vehicles would be within the setback distance of 52.5 m [175 ft] at any moment while the transporter is moving, the annual frequency of exposure, *r*, is

$$r = 2.04 \times 10^{-11} \times 200 \times 8 \times \frac{1{,}000}{5{,}280} = 6.18 \times 10^{-9} \text{ per year} \qquad (15\text{-}3)$$

---

[15]Womack, L.F. Letter (March 28) DIL–03–010, Enclosure 2, Calculation No. —1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[16]Ibid.

[17]Ibid.

The staff concludes that the annual frequency of occurrence of a parked vehicle fuel tank explosion was estimated in a conservative manner.

3,028-L [800-Gal] Tanker Truck While Transporting Fuel Near the Storage Pad

PG&E performed a probabilistic risk analysis[18] to estimate the annual frequency of the potential explosion hazard from the 3,028-L- [800-gal-] gasoline tanker truck while using the transport route near the proposed storage pads. Based on the U.S. Department of Transportation (2003a,b) statistics for large trucks, 429,000 crashes took place in 2001 with approximately 334,721 million km [207,686 million mi] of travel. Therefore, the involvement rate for large trucks would be 207 per 161 million km [100 million mi].

Single-vehicle accident data compiled by the U.S. Department of Transportation (2003a) show that a total of 96,000 of the crashes involved a single vehicle, which is approximately 22 percent of all large truck crashes. Additionally, approximately 31 percent of these crashes took place at a speed below 48 km/hr [30 mph]. Moreover, approximately 0.5 percent of all large truck crashes resulted in fires (U.S. Department of Transportation, 2003a).

PG&E committed preventing any vehicle to pass the tanker truck within 180 m [600 ft] of the proposed facility when the tanker truck is in motion (Assumption 8)[19], so that only single vehicle crashes need to be considered in the analysis. The setback distance is calculated using the methodology given in Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) so that the air overpressure experienced by any safety-related structures, SSC from an accidental explosion of the gasoline tanker truck would be a maximum of 6.9 kPa [1 psi]. Additionally, administrative controls would prevent any vehicle movement at a speed greater than 40 km/hr [25 mph] within the setback region from the proposed facility (Assumption 7).[20]

Assuming that the gasoline tanker will explode if caught on fire, PG&E estimated that the frequency of tanker explosion would be

$$207 \times 0.22 \times \frac{0.31}{100 \times 10^6} \times 0.005 = 7.06 \times 10^{-10} \text{ per mile} \qquad (15\text{-}4)$$

The exposure distance, *s,* is estimated to be 690 m [2,300 ft] based on a 180-m [600-ft] exclusion area from the near cask in the proposed facility. Assumption 5[21] states that the tanker truck would pass by the proposed facility six times in each week. Therefore, the annual frequency of shipment, *f,* is 312. Using Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978), the estimated exposure rate, *r,* is

[18]Womack, L.F. Letter (March 28) DIL–03–010, Enclosure 2, Calculation No. M-1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[19]Ibid.

[20]Ibid.

[21]Ibid

$$r = 7.06 \times 10^{-10} \times 312 \times \frac{2{,}300}{5{,}280} = 9.59 \times 10^{-8} \text{ per year} \qquad (15\text{-}5)$$

The staff concludes that the annual frequency of occurrence of an explosion of the 3,028-L [800-gal] gasoline tanker truck while using the transport route near the proposed storage pads was estimated in a conservative manner.

Propane and Acetylene Bottles Transported Past the Storage Pad

The maintenance facility east of the proposed facility uses acetylene with the cutting torch and propane to run forklifts. One acetylene bottle is the maximum required in 1 year. Additionally, 25.5 L [7 gal] of liquefied propane bottles would be transported past the proposed facility at a rate of one bottle per week. Using administrative procedures, PG&E will ensure that all compressed gas bottles transported past the proposed facility are appropriately secured in the transporting vehicle in the upright position (Assumption 19).[22]

A bottle containing 25.5 L [7 gal] of liquefied propane may rupture while being transported past the proposed facility area releasing the compressed gas. The propane will subsequently mix with air, and the resulting vapor cloud will detonate, which may generate damaging air overpressure levels. Holtec International (2001b) and PG&E (2002) stated that the minimum distance between the point of explosion and the storage casks would be the distance between the storage pads and the security fence because no combustible materials would be permitted inside the proposed facility. The detonation of 26.5 L [7 gal] of propane is equivalent to 4.7 kg [10.37 lb] of trinitrotolune (TNT). At a distance of 15 m [50 ft], the resulting air overpressure would be 16.9 kPa [2.45 psi] (Holtec International, 2001b).

Acetylene is stored and transported in bottles as a gas dissolved in acetone under pressure. While being transported past the proposed facility, acetylene bottles may be ruptured, resulting in release of the confined gas. The acetylene gas, after being released, mixes with air, and the resulting vapor cloud subsequently may detonate. Holtec International (2001b) and PG&E (2002) argued that the minimum distance between the point of explosion and the storage casks would be the distance between the storage pads and the security fence because no combustible materials would be permitted inside the proposed facility. Assuming that a standard bottle contains 2.7 kg [6 lb] of acetylene, Holtec International (2001b) estimated that the explosive potential of the released gas will be equivalent to 0.91 kg [2.015 lb] of TNT. At a distance of 15 m [50 ft], the resulting air overpressure has been estimated to be 8.2 kPa [1.19 psi] (Holtec International, 2001b). Because the HI-STORM 100SA storage casks are designed to perform satisfactorily under 68.9 kPa [10 psi] of air overpressure for a duration of 1 second, accidental detonation of a propane or an acetylene tank while being transported past the proposed facility would not damage the storage casks placed on the pad (Pacific Gas and Electric Company, 2002; Holtec International, 2001b); however, this overpressure level is larger than the allowable air overpressure limit of 6.9 kPa [1 psi]. Therefore, PG&E conducted a

[22]Womack, L.F. Letter (March 28) DIL–03–010, Enclosure 2, Calculation No. —-1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

probabilistic risk analysis[23] to estimate the annual exposure frequency of SSC important to safety to this air-overpressure level.

PG&E[24] postulated that the motive force required for a compressed-gas bottle to fail or explode would be from a vehicle crash. Because the crashes near the proposed facility are limited to single-vehicle incidents, PG&E used an explosion rate, *n*, of 7.06×10^{-10} per mile, estimated for large truck crashes. Additionally, the frequency of bottle shipment, *f*, is assumed to be four times a week or 208 times a year to be conservative. The exposure distance, *s*, is assumed to be 690 m [2,300 ft], the same as with the tanker truck crash. Therefore, the estimated exposure frequency, *r*, is

$$r = 7.06 \times 10^{-10} \times 208 \times \frac{2{,}300}{5{,}280} = 6.39 \times 10^{-8} \text{ per year} \qquad (15\text{-}6)$$

Although pressurized gas bottles may also fail along the welded seam, the bottles are required to meet the current industry standards. Therefore, this mode of failure of gas bottles was not considered credible.

The staff concludes that the annual frequency of occurrence of an explosion of the propane and acetylene bottles transported past the storage pads was estimated in a conservative manner.

Compressed Gas Cylinders

Cylinders containing compressed acetylene, air, argon, helium, nitrogen, oxygen, and propane gases are stored inside the reactor-controlled area and near its south gate (Holtec International, 2002). Internal pressure of the compressed gas cylinders can be in excess of 13.8 MPa [2,000 psi]. Potential energy of the cylinders at such high pressures could have catastrophic effects during a rupture because this potential energy would be released as kinetic energy that could potentially damage SSC important to safety. PG&E postulated that these compressed gas cylinders may be damaged so that the valve assembly at the top of the cylinders is broken. This failure would create a hole, approximately 5 cm [2 in] in diameter, at one end of the cylinder. Gases escaping through this hole would impart a large acceleration to the cylinder body and/or the valve assembly. The cylinders and/or the valve assemblies could accelerate toward the cask systems resulting in impacts (Holtec International, 2001b).

One function of both HI-TRAC 125 Transfer Cask and HI-STORM 100SA storage casks is to prevent any missiles (e.g., gas cylinder body and valve assembly) from affecting the MPC. Based on the calculations performing by Holtec International (2001b), any missile impacting the HI-TRAC 125 Transfer Cask must penetrate a minimum of 3.8 cm [1.5 in] of steel before impacting the confinement boundary of the MPC. Similarly, any missile has to penetrate at least 5 cm [2 in] of steel before impacting the MPC for the HI-STORM 100SA storage cask neglecting the presence of the concrete overpack. Holtec International (2001b) estimated the maximum velocity of all ruptured gas cylinders using the bounding discharge coefficient so that the

[23]Womack, L.F. Letter (March 28) DIL–03–010, Enclosure 2, Calculation No. M-1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[24]Ibid.

estimated acceleration and the resulting force are maximum, and, therefore, the depth of penetration in a steel plate would be maximum.

The maximum depth of penetration by the gas cylinder body occurs with propane gas and is equal to 0.59 cm [0.232 in]. Valve assembly produces a penetration of 0.61 cm [0.241 in]. Therefore, the maximum depth of penetration for all types of cylinders and the valve assemblies is substantially less than the steel thickness available to resist penetration. Consequently, there is reasonable assurance that no SSCs important to safety will be damaged from accidental rupture of compressed gas cylinders.

Stationary Fuel Tanks Near the Transport Route

Three large stationary fuel tanks are located approximately 360 m [1,200 ft] from the transport route at the closest point to the proposed facility. These tanks include a 946-L [250-gal] propane tank, a 7,571-L [2,000-gal] diesel fuel tank, and an 11,356-L [3,000-gal] gasoline tank. These three fuel tanks are located close enough to each other so that an explosion of one tank could cause potential rupture of the other two tanks. Diesel does not present an explosion hazard because of its high flash point. While a rupture and subsequent detonation of either the propane tank or the gasoline tank could potentially rupture the diesel fuel tank, the spilled diesel could burn without exploding. Consequently, the stored diesel would not contribute to the explosion overpressure. Therefore, this event is limited to near-simultaneous explosion of both the propane and gasoline tanks to generate any incident air overpressure. An explosion of these tanks may potentially affect the canister transfer operations at the CTF, the storage casks placed on pads, and the transfer casks being transported on the transport route.

Holtec International (2001b) estimated the air overpressure from a simultaneous explosion of 946 L [250 gal] of propane and 11,356 L [3,000 gal] of gasoline. Combined fuel is equivalent to 53.27 kg [117.33 lb] of TNT, which generates an air overpressure of 5.79 kPa [0.84 psi] at a distance of 366 m [1,200 ft], the minimum distance between the stationary fuel tanks and the transport route (Pacific Gas and Electric Company, 2002). Based on Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978), an air overpressure of 5.79 kPa [0.84 psi] would not cause damage to any safety-related structures (Pacific Gas and Electric Company, 2002). The storage area security fence and the CTF are further away from the storage tanks. Therefore, it is expected that the air overpressure at these locations will be lower than 5.79 kPa [0.84 psi].

The stationary fuel tanks are more than 805 m [0.5 mi] from the proposed storage pad location and at an elevation of approximately 61 m [200 ft] below. These tanks are located southwest of the proposed facility with prevailing southeastern wind directions. Therefore, the winds would normally take the vapor cloud south of the proposed facility. Additionally, the vapor cloud generated at the fuel tank location needs to climb the 61-m [200-ft] hill to reach the proposed facility. Moreover, there is a major cut in the hillside directly above and east of the tanks. This cut would likely channel the vapor cloud away from the proposed facility. Therefore, there is reasonable assurance that any vapor cloud generated at these stationary tanks would not pose any undue hazard to the proposed facility.

The stationary fuel tanks will be periodically filled by standard fuel tankers with a capacity between 11,356 to 15,142 L [3,000 to 4,000 gal]. During any spent fuel transport operation, the filling of these tanks would be suspended and all vehicle movements will be administratively

controlled (Pacific Gas and Electric Company, 2002). Additionally, Section 8.2.6 of the SAR states that administrative controls will be used to ensure that the air overpressure received by any safety-related structures from an explosion of a tanker truck would be less than the 6.9-kPa [1-psi] limit.

Mineral Oil from Diablo Canyon Power Plant Unit 2 Main Bank Transformers

There are six transformers on the Unit 2 side of DCPP: three single-phase 500-kV, two three-phase 25-kV, and one three-phase 12-kV. Additionally, two spare transformers are stored adjacent to the active transformers. The three single-phase 500-kV transformers are located closest to the transport route. Other transformers are mostly shielded from the transport route by these 500-kV transformers because of the layout with respect to the transport route. Each active transformer has a fire-suppression system that will activate in case of a fire.

The mineral oil in the transformers acts as a coolant. It has a flash point of 135 °C [275 °F]. Therefore, an explosion of mineral oil does not pose a significant hazard (Holtec International, 2001b) because this is not a flammable liquid. To be classified as a flammable liquid, the flash point of the liquid should be less than 37.8 °C [100 °F] (National Fire Protection Association, 1997). Although an electrical fault may occur within one of the transformers, the resulting rupture of the transformer case may ignite and burn the mineral oil, but the mineral oil would not explode. Therefore, a potential explosion of the mineral oil at Unit 2 of DCPP was not considered a credible hazard to the operation of the proposed storage facility.

Bulk Hydrogen Storage Facility

A bulk hydrogen facility is located approximately 4.5 m [15 ft] from the transport route from where the loaded transfer casks enter and leave the FHB/AB of Unit 1 of DCPP. This facility contains 6 hydrogen tanks with a total capacity of approximately 8,495 l [300 ft^3]. These tanks are refilled approximately twice a month and are kept in a seismic-qualified rack enclosed in a seismic-qualified vault. The vault has a 0.3- [12-in-] diameter top vent to ensure that no leaked gas builds up. The vault only opens toward the FHB/AB. The hydrogen facility is designed against excessive flow, over pressurization, and vehicle damage during refilling. Therefore, it is extremely difficult to accumulate significant quantity of loaded gas leading to an explosion.

The Electric Power Research Institute Fire Events database considers hydrogen fire to be a credible event and provides a frequency of 3.2×10^{-3} per year.[25] Therefore, the hourly frequency of fire at the bulk hydrogen facility is estimated to be $3.2 \times 10^{-3}/8760$, or 3.7×10^{-7}. Because the design of the facility prevents accumulation of leaked hydrogen gas in confined spaces, it is extremely difficult to have an explosion even in the case of a hydrogen fire. PG&E[26] assumed that in 10 percent of the cases, a hydrogen fire would lead to an explosion in the bulk hydrogen facility and, therefore, the estimated hourly frequency of hydrogen explosion would be $3.7 \times 10^{-7} \times 0.1$, or 3.7×10^{-8}.

---

[25]Womack, L.F. Letter (March 28) DIL–03–010, Enclosure 2, Calculation No. —1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[26]Ibid.

PG&E states that the loaded cask transporter would be in the vicinity of the hydrogen tanks for less than 1 hour during each shipment (Assumption 14), and there will be eight shipments each year (Assumption 1)[27]. To be conservative, PG&E[28] assumed yearly exposure of 10 hours. Therefore, the annual exposure frequency of the transfer cask to a potential hydrogen tank explosion would be

$$3.7 \times 10^{-8} \times 10 = 3.7 \times 10^{-7} \text{ per year} \qquad (15\text{-}7)$$

The staff concludes that the annual frequency of occurrence of an explosion of the bulk hydrogen storage facility was estimated in a conservative manner.

Acetylene Bottles Stored on the East Side of the Cold Machine Shop

A maximum of 10 acetylene bottles are stored on the east side of the cold machine shop near the DCPP. This facility is more than 7.5 m [25 ft] from the transporter route and is protected by concrete block walls on two sides. The third side is protected by a building. Administrative procedures ensure that these bottles are restrained in an upright position because of seismic considerations. This restraint ensures that no potential missiles, originated from an exploding bottle, would be aimed at the transporter route. Furthermore, the cold machine shop facility location allows limited access of vehicles. Additionally, administrative procedures would control any vehicle movement within 52.5 m [175 ft] of the transporter route when the transporter is hauling a loaded transfer cask. Therefore, there is no motive force to initiate damage to the gas bottles leading to an explosion. Consequently, PG&E[29] concluded that accidental detonation of acetylene bottles stored on the east side of the cold machine shop would not be a credible hazard to any safety-related SSC at the proposed facility.

Summary of Review

The potential explosion hazards that may affect the storage casks and the canister transfer operation are (1) detonation of a transporter and onsite vehicle fuel tank, (2) detonation of 3,028-L [800-gal] tanker truck while transporting gasoline past the ISFSI storage pads, (3) detonation of a propane bottle transported past the ISFSI storage pads, (4) detonation of an acetylene bottle transported past the ISFSI storage pads, (5) detonation of large stationary fuel tanks, and (6) an explosive decompression of a compressed gas cylinder. Other sources are far away from the proposed storage site and contain sufficiently small amounts of explosive materials to not pose a credible hazard to the storage casks and canister transfer operations. A transfer cask loaded on a transporter could be affected by (1) detonation of the fuel tank of a transporter or an onsite vehicle (including the potential explosion of a parked vehicle fuel tank), (2) explosion of large stationary fuel tanks in the vicinity of the transport route, (3) explosion of mineral oil from the DCPP Unit 2 main bank of transformers, (4) explosion of the Bulk Hydrogen

---

[27]Womack, L.F. Letter (March 28) DIL–03–010, Enclosure 2, Calculation No. M-1052 to the U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[28]Ibid.

[29]Womack, L.F. Letter (July 28) DIL–03–010, Enclosure 2 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

Storage Facility, and (5) explosion of acetylene bottles stored on the east side of the cold machine shop. Administrative controls will prevent movement of the tanker truck and any onsite vehicles during transporter operation. Similarly, no acetylene or propane bottles will be transported during transporter operations. Decompression of compressed gas cylinders does not pose an air overpressure hazard; missiles generated by the decompression of the cylinders are the primary concern in this situation.

PG&E conducted a probabilistic risk analysis[30] of the remaining explosion hazards that have a potential to cause damage to safety-related structures at the proposed facility. Based on the previous discussion, the annual frequency of exposure to explosion hazards by the storage casks placed on the storage pads at the proposed facility and the canister transfer operation is:

$$P_1 = P_{onsite\,vehicle} + P_{propane/acetylene} + P_{tanker\,truck} + P_{stationary\,tanks}$$
$$or,\ P_1 = 5.92 \times 10^{-7} + 6.39 \times 10^{-8} + 9.59 \times 10^{-8} + 0 = 7.52 \times 10^{-7} \text{ per year} \qquad (15\text{-}8)$$

Similarly, the annual frequency of exposure to explosion hazards by the transfer cask while being transported by the transporter is

$$P_2 = P_{onsite\,vehicle} + P_{parked\,car} + P_{stationare\,tanks} + P_{transformers} + P_{hydrogen} + P_{acetylene}$$
$$or,\ P_2 = 0 + 6.18 \times 10^{-9} + 0 + 0 + 3.7 \times 10^{-7} + 0 = 3.76 \times 10^{-7} \text{ per year} \qquad (15\text{-}9)$$

Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) provides an acceptable methodology to evaluate the potential hazards by an explosion on safety-related SSCs. Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) also states that potential explosion hazards can be screened out if the annual exposure frequency is less than 10^{-6} and conservative estimates are used. PG&E made conservative estimates of the potential explosion hazards, so an annual frequency limit of 10^{-6} is applicable here. Therefore, the staff concludes, based on the review of information and analyses presented by PG&E, that no safety-related SSCs at the proposed facility will be subjected to explosion overpressures that exceed the 6.9 Kpa [1 psi] threshold.

The staff reviewed the information provided by the applicant regarding potential hazards from an accidental onsite explosion at the proposed facility. The staff found the analysis acceptable because

- Applicant has identified the potential sources of hazard
- Applicant used 6.9 kPa [1 psi] as the limiting air overpressure for all safety-related structures
- Applicant developed probabilistic hazard analysis to estimate the annual frequency of exposure of safety-related structures from each potential source of

[30]Womack, L.F. Letter (July 28) DIL–03–010, Enclosure 2 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

explosion that does not meet stand-off zone criterion based on 6.9-kPa [1-psi] air overpressure limit

- Applicant summed the annual frequency of explosion hazard from each individual source to estimate the total hazard to the proposed facility as recommended in Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978)

- Applicant used conservative assumptions to estimate the annual frequency of exposure from each source of the explosion hazard

Based on the foregoing evaluation, the staff finds the ability of the SSCs to maintain subcriticality, confinement, and sufficient shielding of the stored fuel in case of potential onsite explosions has been adequately evaluated.

15.1.2.5 Electrical Accident

Section 8.2.8 of the SAR (Pacific Gas and Electric Company, 2002) evaluates the potential consequences of lightning strikes and a 500-kV transmission line drop on the HI-STORM 100SA storage and HI-TRAC 125 Transfer Casks. Of the different 500-kV transmission line drop scenarios that were considered, the worst-case condition is defined by a line drop of a single conductor of one phase, which causes a single line-to-ground fault current and a voltage-induced arc at the point of contact. Both electrical events (i.e., lightning strike and 500-kV transmission line drop) manifest themselves as electrical discharges that travel along the least resistive path through the cask to ground. Because these events originate from sources that are outside the confines of the cask, the path of least electrical resistance for the HI-STORM 100SA storage cask is the overpack and, for the HI-TRAC 125 Transfer Cask, the enclosure shell. As a result, the MPC will not be susceptible to any electrically induced damage.

In the case of a lightning strike, it was satisfactorily demonstrated that the temperature increase of the HI-STORM 100SA storage cask overpack and HI-TRAC 125 Transfer Cask enclosure shell will be less than 0.6 °C [1 °F].

For the case of the 500-kV transmission line drop, it was determined that holes would be created in both the HI-STORM 100SA storage and HI-TRAC 125 Transfer Cask outer shells by way of material sublimation. Behind the steel outer shell of the HI-STORM 100SA storage cask is a thick concrete layer that would exhibit only localized spalling and crystallization in the immediate region where the steel outer shell sublimation occurred. The staff determined that the resulting effects on the HI-STORM 100SA storage cask decay heat removal and radiation shielding capabilities would be minimal. A hole created in the HI-TRAC 125 Transfer Cask outer shell could cause a loss of the water jacket used to provide neutron shielding and facilitate removal of the spent nuclear fuel decay heat. As discussed in Section 8.2.11 of the SAR, a loss of the water jacket does not cause the radiation dose to exceed the limits of 10 CFR §72.106, and the increase in fuel cladding and component material temperatures will not exceed their short-term accident temperature limits. Moreover, the MPC internal pressure will remain below the accident design limit gauge pressure of 1.38 MPa [200 psi].

In the event that an electrical accident should occur, the previously reviewed and accepted recovery operation procedures will be implemented (U.S. Nuclear Regulatory Commission, 2002a,b).

15.1.2.6 Earthquake

The staff has reviewed the information presented in the following SAR sections: 8.2.1, "Earthquake"; 2.6, "Geology and Seismology"; and 3.2.3, "Seismic Design." Section 4.5 of the SAR classifies the SSC important to safety based on a Quality Assurance (QA) Program described in Chapter 11 of the SAR. The importance to safety for each SSC important to safety is further refined into three QA classification categories based on the guidance contained in NUREG/CR–6407 (McConnell, et al., 1996) (i.e., Categories A, B, and C). The Category A SSCs important to safety include the (1) MPC; (2) fuel basket; (3) damaged fuel container; (4) transfer cask; (5) MPC lift cleats and downloader slings; (6) transfer cask impact limiters and lift links; (7) HI-STORM 100 System lifting brackets, mating device bolts, and shielding frame, and lift links; (8) cask transporter; and (9) lateral restraints (HI-TRAC 125 Transfer Cask and transporter at the CTF). The classification Category B SSC important to safety include (1) HI-STORM 100SA storage cask overpack; (2) storage pads; (3) overpack anchorage hardware; (4) CTF; (5) transfer cask horizontal lift rig and lift slings; (6) upper and lower fuel spacer columns and end plates; (7) transporter connector pins; and (8) helium fill gas. The classification Category C SSCs important to safety include the HI-STORM 100 System cask mating devices (except bolts and shielding frame).

A seismic event can occur at any time during any stage of a transfer or storage operation involving a cask or a canister. At a specific site, earthquake potential is often described by the annual probability of exceeding certain ground motion levels or seismic hazard curves. The design earthquake, double-design earthquake, Hosgri earthquake (HE), and long term seismic program earthquakes are the seismic licensing basis for the DCPP. The applicant indicated that, because both DCPP and the ISFSI sites are classified as rock and they have similar ranges of shear-wave velocities within the rock classification, and because the distance to the controlling seismic source is essentially the same, the DCPP ground motions are judged to be applicable to ISFSI design. Section 2.1.6 of this SER provides additional information about the seismic ground motion hazard and the staff review of the information.

In conducting analyses of transporter stability, slope stability, and ISFSI storage pad sliding, the applicant developed the ISFSI long-period (ILP) earthquake spectra. The ILP are 84th percentile spectra at damping values of 2, 4, 5, and 7 percent for the horizontal and vertical components that extended out 10 s and that include near-fault effects of directivity and fling. The applicant indicates that the ILP spectra envelop the double-design earthquake spectra at 2- and 5-percent damping; the HE spectra at 4-, 5-, and 7-percent damping; and the long term seismic program earthquake spectra at 5-percent damping. The applicant further indicates that the use of ILP earthquake spectra for transporter stability, slope stability, and ISFSI storage pad sliding would provide an extra design margin by considering long-period energy. Five sets of ILP spectra-compatible time histories generated from large-magnitude earthquakes (M > 6.7) recorded at short distances {< 15 km [9.3 mi] from the fault) were used as input for the analyses. Based on the statement provided in the SAR, the staff concluded that the use of ILP spectra-compatible time histories to assess transporter stability, slope stability, and ISFSI storage pad sliding potential is acceptable.

Seismic Analysis of Cask Transportation on Transport Route

The transport route from the FHB/AB at the DCPP to the ISFSI storage pad is approximately 1.93-km [1.2-mi] long. Approximately, one-third of the route is on bedrock, and the rest is on surficial deposits over bedrock. The route is made up of slopes with an 8.5-percent nominal grade decline and a 6-percent nominal grade incline and a 2-percent grade perpendicular to the roadway with a decline toward the hill side. The minimum roadway width is 7.92 m [26 ft]. The cask transporter carries a HI-TRAC 125 Transfer Cask in a horizontal position from the FHB/AB to the CTF for MPC transfer operation. After the MPC transfer operation is executed, the cask transporter carries the loaded overpack in a vertical orientation to the final position on the ISFSI storage pad. The cask transporter is 5.37-m [17.625-ft] wide and 7.47-m [24.5-ft] long. The applicant states that the maximum acceptable sliding movement along the roadway is limited to the cask transporter track length to ensure that the transporter will remain on the roadway after exiting a turn in the roadway. Assuming that the cask transporter travels along the middle of the roadway, the allowable lateral sliding distance is the distance between the edge of the transporter and the edge of the roadway, which is approximately 1.28 m [4.19 ft].

During transport to the ISFSI storage pad, the cask transporter protects the MPC from the effects of earthquake ground motions. The transporter stability assessment discussed in the SAR was analyzed three dimensionally. The cask transporter, the HI-STORM 100SA storage cask, the HI-TRAC 125 Transfer Cask, the MPC (including the fuel basket, fuel, and lid), and the cask lids were modeled as rigid bodies. The mass of the MPC and the contained spent nuclear fuel is lumped in a free-standing rigid cylinder. Three cases of roadway conditions were modeled: flat surface, 6-percent grade, and 8.5-percent grade. For all cases, the ground surface was treated as a nondeformable boundary. The SAR states that a transporter stability analysis was performed for a potential transporter overturning or sliding off the roadway using only the bedrock ground acceleration associated with the ILP earthquake time histories. The maximum sliding along the roadway axis of approximately 0.77 m [2.52 ft] occurred on the portion of 8.5-percent grade roadway, and the maximum sliding transverse to the roadway axis of approximately 0.27 m [0.89 ft] occurred on the portion of the roadway with a 6-percent grade. These sliding distances are small compared to the corresponding allowable sliding distance. The analysis also demonstrated that overturning is not credible under the ILP seismic events.

The applicant indicates that peak ground accelerations at certain points along the surface of surficial deposits over bedrock of the transport route can be 1.5 to 2.0 times the amplitude of the peak ground acceleration on bedrock. PG&E did not specifically analyze the potential for overturning and sliding of the transporter on surficial deposits. The SAR points out that a significant safety margin exists to prevent a transporter from overturning or sliding off the roadway while traveling on the surficial deposits even through the ground acceleration would be amplified.

PG&E provided two analyses to address the potential accident scenario, in which an earthquake occurs while the cask is being transported on a portion of the roadway underlain by soil to the CTF or ISFSI pad. The first is a risk assessment to show that this scenario is not credible. The second is a calculation to show that transporter and cask will remain stable during an earthquake. Staff review of these two analyses is discussed in detail in the following paragraphs. In summary staff agree with the PG&E assessment that this is not a credible scenario. The staff conclude that the regulatory of 10 CFR §72.90, §72.90(a), §72.98, §72.102, and §72.122 have been satisfied.

PG&E conducted a probabilistic risk assessment calculation and concluded that the annual probability of damage to the transporter, while transporting material from the power plant to the CTF is 2.1×10^{-10}, which is substantially less than the 1×10^{-6} threshold criteria recommended for credible events.[31] The PG&E probabilistic risk assessment calculation includes the 1.4×10^{-3} annual exposure probability for transport casks on the transport route (12 hr/yr) and the 1.2×10^{-7} annual exceedence probability for two times the ILP earthquake ground motions.

Staff reviewed the PG&E probabilistic risk calculation and agree with their conclusion. Specifically, the annual exceedence probability for earthquake-induced damage of the casks while in transit from the power plant to the CTF is less than 1×10^{-6} and is, therefore, not a credible hazard. The use, however, of the annual exceedence probability associated with twice the ILP earthquake ground motions is not considered to be appropriate. Twice the ILP earthquake ground motions was used by PG&E to account for possible site response amplification on those portions of the transport route underlain by soil, not as an added factor in the probability calculation. For this reason, the annual probabilities of the ILP earthquake ground motions, not the annual probabilities for twice the ILP earthquake ground motions, should be used in the probabilistic risk calculation.

Staff independently estimated the upper bound annual exceedence probability for earthquake-induced damage of the casks while in transit from the power plant to the CTF. The estimated probability would be no more than 1.4×10^{-7} per year. The calculation performed by the staff assumed a maximum annual probability for the ILP earthquake ground motions to be less than 1×10^{-4}. When combined with the 1.4×10^{-3} probability of annual exposure of the transfer casks being in transit, an upper bound value of 1.4×10^{-7} probability of annual exposure is calculated. The exact probability depends on a number of factors, including the spectral frequency of interest and the statistical measure used (mean, median, or 84th percentile). Based on this calculation, staff concludes, with reasonable assurance, that earthquake-induced damage of the casks while in transit from the power plant to the CTF is not a credible hazard to the proposed facility.

In response to NRC staff RAIs, the applicant conducted additional analyses to demonstrate the transporter stability during an earthquake event while traveling on surficial deposits. The first analysis involved an assessment of the ILP spectra used for evaluation of transporter stability on the transport route over bedrock[32]. This analysis examined the thickness and shear-wave velocity profiles of the surficial deposits at the site and compared the shear-wave velocity profiles in the surficial deposits with those for the design-basis long-term seismic program ground motions and concluded that the design-basis long-term seismic program ground motions are applicable to sites on surficial deposits. Because the ILP ground motions envelop the design-basis long-term seismic program ground motions, the ILP ground motions are applicable to the portions of the route with surficial deposits as well.

---

[31]U.S. Nuclear Regulatory Commission. In the Matter of Private Fuel Storage LLC (Independent Spent Fuel Storage Installation). Memorandum Order (November 14) CLI–01–22. NRC Docket No. 72-22-ISFSI. Washington, DC: U.S. Nuclear Regulatory Commission. 2001.

[32]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

The second analysis includes an assessment of transporter stability subjected to the five ground accelerations twice those of the ILP earthquake accelerations.[33] The transporter and the HI-TRAC 125 Transfer Cask were modeled as a lumped mass, using the SAP2000 nonlinear computer code. Two bounding coefficients of friction for the transporter-ground interface were analyzed: (1) coefficient of friction equal to 0.4, emphasizing sliding, and (2) coefficient of friction equal to 0.8, emphasizing overturning. The ground surface was treated as a nondeformable boundary for all the cases analyzed. For the case with a coefficient of friction equal to 0.4, the maximum amount of transport sliding along the roadway axis will be approximately 4.13 m [13.56 ft], and the maximum amount of transverse movement is approximately 1.43 m [4.68 ft] for a 6-percent slope and is approximately 5.28 m [17.33 ft] for the maximum sliding along the roadway axis and 1.48 m [4.85 ft] for the maximum transverse movement on a slope with a 8.5-percent grade. The best estimate, which is the average of the maximum sliding distances obtained using the five time-history sets, on a 6-percent grade slope is 2.48 m [8.15 ft] and 0.87 m [2.86 ft] for sliding along the roadway axis and transverse direction. Analysis of transporter sliding on an 8.5-percent grade roadway was performed using one time history only. No analyses were performed using the other four time histories. Therefore, the best estimates for sliding on an 8.5-percent grade slope are not available. For the case with a coefficient of friction equal to 0.8, the transporter is not found to be susceptible to rigid body rocking and, consequently, overturning. The maximum lateral sliding of 1.48 m [4.85 ft] on an 8.5-percent grade roadway appears to exceed the allowable sliding distance of 1.28 m [4.19 ft]. The staff recognizes that the best estimate for sliding may be more representative to assess the transporter stability instead of the maximum value. The best estimate is likely to be smaller than 1.48 m [4.85 ft]. Because the best estimate was not provided in the analysis report, the staff cannot conclude with reasonable assurance that the transporter will remain on the roadway during a seismic event. However, as stated previously, staff agree with the PG&E conclusion that this scenario is not credible event.

Seismic Analysis of the Canister Transfer Facility

The staff reviewed Section 4.2.1.2 of the SAR and found that structural analysis of the CTF to mitigate effects of seismic loading has been demonstrated as documented in Section 5.1.4.4 of this SER.

The steel structures were analyzed to demonstrate compliance with the material allowables (Holtec International, 2001c). This analysis addressed the following major structural elements: main shell, lifting jacks, jack support platform, CTF base support block, and lifting platform. The appropriate spectral values are used to account for possible amplification of the horizontal accelerations of the stacked components. It was demonstrated that the factors of safety for all components and all load conditions are greater than 1.0. The adequacy of the structures has been demonstrated by the analysis results given in the SAR, as designed to satisfy the requirements of ASME Section II, Subsection NF (ASME International, 1995a).

Loads from the Holtec International structural evaluation were also used in the calculation of the necessary thickness and reinforcement for the CTF concrete (ENERCON Services Inc., 2001a).

---

[33]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

The analysis determined the required size and general reinforcing requirements to resist the loads applied to the concrete structure. The concrete structure is designed to withstand loads from both the CTF and the transporter. Using the controlling load combinations, an analysis identified shear and axial forces and moments in the reinforced concrete structural elements of the CTF. Steel reinforcement size and placement for the pad and wall were established based on these demands. The design of the concrete structure and its reinforcement are based on the requirements in American Concrete Institute (ACI) 349-97 (American Concrete Institute, 1998). Results of the analysis indicate that the available design strength of the CTF exceeds that required for the factored design loads.

Seismic Analysis of HI-STORM 100SA Anchored on the ISFSI Storage Pad

Structural analyses of the HI-STORM 100SA storage cask are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). The Diablo Canyon ISFSI SAR Section 4.2.3 provides a summary of the analyses performed in the HI-STORM 100 System FSAR. The loading conditions at the Diablo Canyon ISFSI are enveloped by the loading conditions considered in the HI-STORM 100 System FSAR (Holtec International, 2002). As documented in the HI-STORM 100 System SER, the structural analysis shows that the structural integrity of the HI-STORM 100 System cask system is maintained during all credible loads. Based on the results presented in the HI-STORM 100 System FSAR, the stresses in the overpack structures during the most critical load combinations are less than the allowable stresses of ASME Boiler and Pressure Vessel Code, Section III (ASME International, 1995b) for the structures materials.

Seismic Analysis of the ISFSI Storage Pad

SAR Section 8.2.1.2.3.1, "Cask and Anchorage Seismic Analysis," summarizes seismic analyses of the cask and anchorage system performed by Holtec International. Although the Diablo Canyon site-specific seismic zero period accelerations for all events are lower than those identified in Appendix B of the Holtec CoC (U.S. Nuclear Regulatory Commission, 2002a), Holtec International performed an analysis of the anchored HI-STORM100 System casks at the Diablo Canyon ISFSI (Holtec International, 2001d). The primary reason for this analysis was a difference in the number of anchor rods identified for the Diablo Canyon ISFSI with respect to the design basis given for the HI-STORM 100 System storage system (Holtec International, 2002). The staff review is summarized in Section 5.1.3.4 of this SER. The results indicate that the casks do not develop body decelerations that exceed the cask design basis of 45 g. The seismic events do not induce stress in the preloaded anchor studs, cask flange, and shell that exceed the design-basis ASME Code limits. The interface loads transferred to the ISFSI pad embedment were established.

SAR Section 8.2.1.2.3.2, "Storage Pad Seismic Analyses," identifies the analysis performed to ensure that the reinforced concrete pads and the anchored casks remain functional during all seismic conditions. Two analyses are covered in this section, a static analysis (ENERCON Services Inc., 2001b) and a nonlinear pad sliding analysis (Pacific Gas and Electric, 2001a). This analysis (ENERCON Services Inc., 2001b) included a static analysis to determine the storage pad size and thickness required to resist the loads resulting from seismic accelerations applied to the pad and resultant loads from the cask dynamic analysis (Holtec International, 2001d). In addition to the cask loads, an inertia force was applied to the pad with reference to

the zero period acceleration of the seismic event. The pad and cask vertical displacements are small and within acceptable limits. These maximum tensile stresses in the concrete are less than the tensile stress that will cause cracking in the 34.5-MPa [5,000-psi] concrete. The maximum compressive stress is significantly less that the 34.5-MPa [5,000-psi] design value. Sections throughout the pad were isolated for the HE seismic event calculations and the internal forces acting upon them were computed. The resulting internal forces for design purposes are given in Table 11 of the ENERCON calculation package (ENERCON Services Inc., 2001b). The results of the analysis will be used in Calculation No. PGE–009–CALC–007 to evaluate the concrete according to the design codes and to determine the size of the steel reinforcement in compliance with the requirements of ACI 349-97 (American Concrete Institute, 1998). Since this is a preliminary design, this analysis (PGE–009–CALC–007) has not been submitted and was not reviewed as part of this SER. A nonlinear analysis was performed to determine the extent of sliding that occurs at the pad/rock interface (Pacific Gas and Electric Company, 2001a).

The anchorage system was designed to meet the ductile anchorage provision of the proposed Draft Appendix B for ACI 349-97 (American Concrete Institute, 1998). To satisfy the requirements of Appendix B of ACI 349-97 the diagonal tension shear capacity must exceed the anchor bar ductile design strength of 1.05 MN [235.63 kips]. The applicant has provided sufficient reinforcing steel to ensure the failure cone for concrete pullout intersects sufficient rebar to prevent brittle failure (ENERCON Services Inc., 2003a). The reinforcing steel in the storage pad (ENERCON Services Inc., 2003b) has been sized in accordance with the requirements of ACI 349-97 (American Concrete Institute, 1998).

15.1.2.7 Loss of Shielding

Section 8.2.11 of the SAR (Pacific Gas and Electric Company, 2002) evaluates the potential consequences of a loss-of-neutron shielding for the HI-TRAC 125 Transfer Cask. The potential consequences of this postulated accident were determined by assuming a loss of the water jacket and Holtite-A solid neutron shielding. The staff previously determined that the methodology used to assess this postulated accident is acceptable and the short-term fuel cladding and other component temperature limits, the MPC accident internal pressure, and the accident dose limits defined by 10 CFR §72.106 are not exceeded so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to.

In the event that the HI-TRAC 125 Transfer Cask loses its neutron shielding, the previously reviewed and accepted recovery operation procedures will be implemented (U.S. Nuclear Regulatory Commission, 2002a,b).

Section 8.2.6.3 of the SAR specifies that consequences of the Diablo Canyon site explosion events involving detonation are enveloped by the design-basis accident conditions in the HI-STORM 100 System FSAR (Holtec International, 2002). Additionally, there is no effect on shielding, criticality, thermal, or confinement capabilities of the HI-STORM 100 System as a result of the explosion pressure load. Based on the structural and radiological evaluations presented in Chapters 3 and 11 of the HI-STORM 100 System FSAR, the applicant concludes that the MPC confinement boundary will remain intact and the shielding effectiveness of the storage and transfer casks will not be significantly affected by any potential onsite explosion.

Considering the results of the onsite explosion accident analysis evaluation presented in Section 15.1.2.4, "Fire and Explosion," of this SER, the staff finds that, based on submitted information and analyses, the maximum reduction in ISFSI radiation shielding thickness, material shielding effectiveness, or loss of temporary shielding in all possible shielding areas caused by postulated onsite explosion events, has been adequately evaluated by the applicant. Therefore, information and analysis presented by the applicant provide reasonable assurance that the dose to any individual beyond the owner-controlled area will not exceed the limits specified in 10 CFR §72.106(b) and the occupational exposures from accident recovery operations will not exceed the limits specified in 10 CFR Part 20.

15.1.2.8 Adiabatic Heatup

The staff has previously determined that the methodology used to estimate the time required to reach the short-term, fuel-cladding temperature limit of spent nuclear fuel stored in the HI-STORM 100 System storage cask under adiabatic conditions is acceptable (U.S. Nuclear Regulatory Commission, 2002a,b). The HI-STORM 100 System FSAR (Holtec International, 2002, Figure 11.2.6) indicates that a total cask decay heat load of 30 kW [102,360 BTU/hr], which bounds the cask decay heat load specified for the Diablo Canyon ISFSI, will not cause the short-term cladding temperature limit for the spent nuclear fuel to be exceeded for 45 hours under adiabatic conditions. Moreover, the internal pressure limit for the MPC is not exceeded within the 45-hour timeframe for this condition.

In the event that the HI-STORM 100 System storage cask is subjected to conditions that thermally insulate its exterior (e.g., encased within soil as the result of a landslide), the previously reviewed and accepted recovery operation procedures will be implemented (U.S. Nuclear Regulatory Commission, 2002a,b).

15.1.2.9 Full Blockage of Air Inlets and Outlets

The staff previously determined that the methodology used to estimate the time required to reach the short-term, fuel-cladding temperature limit of spent nuclear fuel stored in the HI-STORM 100SA storage cask subjected to 100-percent blockage of the air inlet ducts is acceptable (U.S. Nuclear Regulatory Commission, 2002a,b). For the bounding values of decay heat load of 30 kW [102,360 BTU/hr] and insolation of 834 w/m^2 [800 g-cal/cm^2] per day {387 W/m^2 [123 BTU/hr-ft^2]}, the short-term cladding temperature limit for the spent nuclear fuel will not be exceeded for 72 hr when the HI-STORM 100SA storage cask air inlet ducts are 100-percent blocked. Moreover, the internal pressure limit for the MPC is not exceeded within the 72-hour timeframe for this condition. Furthermore, the HI-STORM 100 System CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix A) includes surveillance requirements for ensuring that the cask heat removal system is operational during storage (i.e., the air ducts are inspected or the temperature differential between the convected cooling air exiting the outlet vents and ambient air is measured every 24 hours to ensure that the ducts are free of blockages). In the event that the HI-STORM 100SA storage cask air inlet ducts are found to be partially obstructed or blocked, the previously reviewed and accepted recovery operation procedures will be implemented (U.S. Nuclear Regulatory Commission, 2002a,b).

15.1.2.10 Tornadoes and Missiles Generated by Natural Phenomena

The staff reviewed the information presented in SAR Sections 3.2.8, "Tornado and Wind Loadings;" 3.3.2.3.3, "Maximum Permissible Tornado Wind and Missile Load;" 4.2.3.3.2.6, "Tornado Winds and Missiles;" and 8.2.2, "Tornado." In addition, responses to RAIs 4-3, 15-18, 15-19, 15-20, and 15-21, including Holtec International (2001e, Attachment 4-1, report "Design Basis Wind and Tornado Evaluation for DCPP") and Section 3.3, "Wind and Tornado Loadings of DCPP" Units 1 and 2 FSAR of, Pacific Gas and Electric Company (2001b), have been reviewed. This evaluation assumed that site personnel would not have any prior warning before the proposed facility SSC are impacted by a potential design-basis tornado and a tornado missile.

The annual mean number of days with tornadoes is zero at the proposed site. Characteristics of the design-basis tornado and tornado missile are given in Section 3.2.1 of the SAR (Pacific Gas and Electric Company, 2002). The SAR developed the characteristics of the design-basis tornado in accordance with the DCPP licensing-basis wind speed of 89 m/s [200 mph]. The proposed site is located in Region II as defined in Regulatory Guide 1.76 (U.S. Nuclear Regulatory Commission, 1974). The characteristics of the design-basis tornado for the proposed site are defined as a tornado with a maximum wind speed of 89 m/s [200 mph], a rotational speed of 70 m/s [157 mph], a translational speed of 19 m/s [43 mph], and a 5.9-kPa [0.86-psi] pressure drop at a rate of 2.5 KPa/s [0.36 psi/s].

The design-basis tornado missiles considered in the SAR of Diablo Canyon ISFSI are based on Spectrum II missiles of Section 3.5.1.4, "Missiles Generated by Natural Phenomena," of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981a), Diablo Canyon FSAR Update (Revision 14) (Pacific Gas and Electric Company, 2001b), and three 500-kV tower missiles specific to the ISFSI (Pacific Gas and Electric Company, 2002). These objects are postulated to be picked up and transported by the winds of a design-basis tornado. A list of these missiles is provided in Table 15.1.

Important to safety SSCs that may be affected by design-basis tornado missiles are: (1) CTF, (2) site transporters, (3) transfer casks, and (4) storage casks. These SSC are required to function during this design-basis event.

Based on kinetic energy, PG&E (2002) assumed that an automobile at 203 km/hr [126 mph], a 500-kV insulator string at 253 km/hr [157 mph], and a 2.5-cm-[1-in-] diameter steel rod at 144 km/hr [89.5 mph] are the bounding missiles for large, intermediate, and small missiles categories. PG&E assumed that the impact velocity of an automobile is consistent with that suggested in NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981b). PG&E (2002) developed an equation to estimate the maximum horizontal missile velocity for a 322-km/h [200-mph] tornado from a 386-km/h [240-mph] Type III tornado curve using Figure 16.3.1 of Simiu and Scanlan (1986)[34]. However, the basis for the equation is not clear. This formula will produce a different result for the correlation power factor if tornados other than Type III are used.

---

[34]Womack, L.F. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon Independent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company. 2002.

The staff confirmatory calculation indicated that energy imparted by the automobile is significantly larger than that of the utility pole. Therefore, any impact of a utility pole would be bounded by the automobile impact for assessing transporter stability. Holtec International (2001e) studied the effects of transporter stability while transporting a loaded transfer cask to the storage area at the proposed facility. This analysis included a large missile represented by a 1,800-kg [4,000-lb] car traveling at a speed of 56 m/s [126 mph]. The impact analysis result indicates that a loaded transporter would be displaced laterally by a distance of only 1.65 cm [0.65 in]. However, the transporter remains stable and does not tipover as a result of this impact.

The staff reviewed the information provided by the applicant, evaluated the analyses of potential hazards from design-basis tornadoes and tornado missiles at the proposed facility, and conducted a confirmatory analysis. The staff concludes that a tornado or tornado-generated missile would not impair the ability of the SSCs to maintain subcriticality, confinement, and sufficient shielding of the stored fuel.

15.1.2.11 Accidents at Nearby Sites—Aircraft Crash Hazards

The staff reviewed the information presented in the SAR Section 2.2 (Pacific Gas and Electric Company, 2002). In addition, the staff reviewed information presented by PG&E in response to the staff RAIs[35]. The purpose of this review is to ensure that the risk to the proposed facility caused by aircraft hazards has been appropriately estimated and is acceptable.

The staff reviewed the aircraft crash hazard analysis in accordance with NUREG–0800, Section 3.5.1.6, Aircraft Hazards (U.S. Nuclear Regulatory Commission, 1981b). The staff accepts the methodology in NUREG–0800, as applicable, for reviewing the aircraft crash probability for the proposed facility site. NUREG–0800, Section 3.5.1.6, "Aircraft Hazards," (U.S. Nuclear Regulatory Commission, 1981b) provides three screening criteria that must be satisfied to conclude, by inspection, that the aircraft hazards at a nuclear power plant are less than 1×10^{-7} per year for accidents that could result in radiological consequences greater than 10 CFR Part 100 exposure guidelines. The staff review indicates the proposed facility site does not satisfy screening Criterion II.1(a), which states "The plant-to-airport distance, D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance, D, is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2." Based on the information given in the SAR and projected air traffic increase in the next 25 years by the Federal Aviation Administration (FAA), the projected annual number of operations may not satisfy Criterion II.1(a). Additionally, screening Criterion II.1c) states "The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern." As stated in Womack,[36] air traffic to San Luis Obispo County Regional Airport passes the proposed site at a distance of 1.6 km [1 mi]. Therefore,

---

[35]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[36]Ibid.

Table 15-1. Tornado missiles considered in Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI)

Missile	Mass kg [lb]	Velocity Considered m/s [mph]	
		Diablo Canyon ISFSI Safety Assessment Report	Holtec International (Region I)
Automobile	1,800 to 1,814 [3,968 to 4,000]	56 [126]	56 [126]
Utility Pole	510 [1,124] 33-cm- [13.5-in-] diameter, 10.7 m [35 ft] long, density of 688.8 kg/m^3 [43 lb/ft^3] in Diablo Canyon Power Plant Units 1 & 2	16 [35]	48 [107.4]
30-cm- [12-in-] diameter Schedule 40 pipe	340 [744] 4.5 m [15 ft] long, density of 7,849 kg/m^3 [490 lb/ft^3] in DCPP Units 1 and 2	2.2 [5]	28 [62.6]
15-cm- [6-in-] diameter Schedule 40 pipe	130 [285] 4.5 m [15 ft] long, density of 7,849 kg/m^3 [490 lb/ft^3] in DCPP Units 1 and 2	3 [7]	42 [93.9]
20-cm- [8-in-] diameter solid steel cylinder	125 [276]	56 [126]	56 [126]
10 cm × 30 cm × 3.3 m [4 in × 12 in × 10 ft] board	49 [108] In DCPP Units 1 & 2, 91 kg [200 lb], density of 801 kg/m^3 [50 lb/ft^3]	89 [200]	Not Applicable
7.5-cm- [3 in] diameter, 3.3— [10-ft-] long Schedule 40 pipe	34.5 [76] In DCPP Units 1and 2, 4.5— [15-ft-] long pipe with density of 7,849 kg/m^3 [490 lb/ft^3]	29.8 [66.7]	Not Applicable
500-kV insulator string	344.7 [760]	70 [157]	Not Applicable
5 cm × 5 cm × 0.32 cm [2 in × 2 in × 1/8 in] steel angle {1.5 m [5 ft] long}	3.9 [8.6]	70 [157]	Not Applicable
2.5-cm- [1-in-] diameter steel rod	4 [8] 0.9— [3-ft-] long, density of 7,849 kg/m^3 [490 lb/ft^3] in DCPP Units 1 and 2	2.2 [5]	40 [89.5]
2.5-cm- [1-in-] diameter solid steel sphere	0.22 [0.5]	56 [126]	56 [126]

screening Criterion II.1.(c) is also not satisfied. According to NUREG–0800 review guidance, a detailed review is, therefore, needed to assess the aircraft crash hazards to the proposed site. PG&E conducted a detailed analysis[37] to estimate the annual frequency of a potential aircraft crash on the proposed facility. Additionally, the staff conducted its own confirmatory analysis. These analyses are discussed in the following.

Estimating the total probability of an aircraft crash onto the proposed facility site requires an evaluation of crash probabilities from several sources:

- Aircraft taking off and landing at San Luis Obispo County Regional Airport
- Aircraft taking off and landing at other municipal airports located close to the site, such as Oceano County Airport, Camp San Luis Obispo Heliport, and Vandenberg Air Force Base
- Aircraft flying the low-altitude Victor Airway 27 (V–27) (commercial airway)
  —Landing at or departing from San Luis Obispo County Regional Airport
  —Not landing at or departing from San Luis Obispo County Regional Airport
- Aircraft flying military training route VR–249.

Aircraft Taking Off and Landing at San Luis Obispo County Regional Airport

San Luis Obispo County Regional Airport is approximately 19.3 km [12 mi] east of the proposed site. This airport has four runways. Only Runway 11 is equipped for instrument landing approach. The other three runways are used for visual landing. Aircraft use Airway V–27 to align for instrument landing at San Luis Obispo County Regional Airport. These aircraft come to approximately 1.6 km [1 mi] of the proposed site at an elevation of 914 m [3,000 ft]. Based on NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981b), aircraft, while flying Airway V–27 for instrument landing at San Luis Obispo County Regional Airport, will be in-flight mode. Their contribution to the crash hazard has been accounted for in the analysis of V-27. The commonly used approach route for visual landing at San Luis Obispo County Regional Airport passes approximately 12.8 km [8 mi] from the proposed site.

PG&E[38] in Section 2.2.1.3, "Hazards from Air Crashes," stated that approximately 92,300 operations (taking off or landing) occur annually at San Luis Obispo Regional County Airport. However, while discussing local traffic on Airway V–27, PG&E[39] stated approximately 16,100 takeoffs and landings occur annually at San Luis Obispo County Regional Airport, based on an average of data from 1998–2001, by commercial or air-taxi aircraft. Primarily turboprop aircraft with a gross weight of not more than 13,608 kg [30,000 lb] are used in these commercial flights. Additionally, private aircraft (i.e., general aviation aircraft) landed at or took off from San Luis Obispo County Regional Airport approximately 7,560 times monthly, based on the

---

[37]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[38]Ibid.

[39]Ibid.

average of data from 1998–2001. These aircraft have gross weight of less than 5,670 kg [12,500 lb]. Consequently, at least a total of approximately 106,720 landings and departures took place annually at the San Luis Obispo County Regional Airport without counting the operations by military aircraft.

PG&E concluded that no analysis would be necessary as the number of annual operations at San Luis Obispo County Regional Airport is below the number needed to have an annual crash frequency of 10^{-7} based on Criterion II.1(a) of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981b).

Staff Analysis: Staff independently verified the number of annual operations at this airport from the FAA database <http://www.gcr1.com/5010WEB/default.htm> and an other source <http://www.airnav.com>. Based on this information, approximately 72,000 annual operations take place at the San Luis Obispo County Regional Airport out of which approximately 16,500 operations are by commuter aircraft and approximately 55,000 operations are by General Aviation aircraft in addition to approximately 900 operations by military aircraft. Because the number of annual operations given by PG&E (i.e., 106,720 with 900 additional operations by military aircraft) is bounding, the staff used that value for further review.

Because the airport is approximately 19.3 km [12 mi] away from the proposed site, the estimated annual frequency of crash onto the proposed facility is insignificant using the methodology, given in NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981b), to analyze the crash potential of aircraft landing at or taking off from an airport.

The staff reviewed the information and the analysis provided by the applicant with respect to the potential hazard of aircraft taking off and landing at San Luis Obispo County Airport. The staff found the hazards acceptable because

- Adequate information has been presented to describe the potential hazard.
- Acceptable methodology has been used to screen the potential hazard.

Based on the this information, the staff has concluded that aircraft taking off and landing at San Luis Obispo County Airport would not pose any undue hazard to the proposed facility.

Aircraft Taking Off and Landing at Oceano County Airport, Camp San Luis Obispo Heliport, and Vandenberg Air Force Base

There are several smaller municipal airports in the vicinity of the proposed site. Oceano County Airport is located 24 km [15 mi] away from the proposed site. Only General Aviation aircraft with weight not more than 5,670 kg [12,500 lb] use this airport. Estimated annual traffic at this airport is approximately 26,400 (Pacific Gas and Electric Company, 2002). Both the FAA database <http://www.gcr1.com/5010WEB/default.htm> and another source <http://www.airnav.com> give the estimated annual number of flight operations at this airport to be approximately 10,000. Therefore, the estimated number for annual flights used in the SAR is conservative. Again, PG&E concluded that no analysis would be necessary as the number of annual operations at San Luis Obispo County Regional Airport is below the number needed to have an annual crash frequency of 10^{-7} based on Criterion II.1(a) of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981b).

Based on the analysis methodology given in NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981b), staff estimates that the frequency of aircraft crashing onto the proposed facility while taking off or landing at Oceano County Airport is insignificant.

Camp San Luis Obispo airfield, located approximately 13 km [8 mi] northeast of the proposed site, is a heliport owned by the U.S. Army. Therefore, the staff concludes that landings and takeoffs by helicopters at this heliport do not pose a credible hazard to the proposed facility because of long distance, based on the U.S. Department of Energy (DOE) Standard (U.S. Department of Energy, 1996).

Vandenberg Air Force Base is 56 km [35 mi] away from the proposed site. Therefore, any landing or takeoff operations at Vandenberg Air Force Base will pose a negligible hazard to the proposed facility.

Aircraft Flying Low-Altitude Airway Victor 27 (V-27)

A low-altitude Victor Airway 27 (V–27) passes approximately 8 km [5 mi] east of the proposed facility. Aircraft use this airway to fly between the Santa Barbara area and the Big Sur area. Aircraft using V–27 can either land at San Luis Obispo County Regional Airport or fly to the destination without landing. The majority of the aircraft using airway V-27 fly at an en route altitude of 3,333 m [10,000 ft] above mean sea level (MSL). Occasionally, V-27 is also used by traffic approaching San Luis Obispo County Regional Airport from the south for instrument landings and instrument departures to the south from Runway 11 or they circle to land on Runway 29, and instrument departures to the south from Runway 29 at San Luis Obispo County Regional Airport. Aircraft using this approach or departure pattern pass as close as 1.6 km [1 mi] from the proposed site at an elevation of 914 m [3,000 ft].

Aircraft Landing or Departing San Luis Obispo County Regional Airport

PG&E used the FAA database <http://www.apo.data.faa.gov/faaatadsall.htm> to obtain information about commercial or air taxi and general aviation operations at San Luis Obispo County Regional Airport. An average of 16,100 operations (i.e., takeoffs or landings) took place annually during 1998–2001.[40] Additionally, an average of 1,781 landings took place annually at this airport under instrument meteorological conditions[41].

Based on the scheduled airline flight information at San Luis Obispo County Regional Airport, PG&E estimated approximately 65 percent of the commercial traffic is departing to or approaching from the south.[42] Therefore, approximately 1,781 × 0.65, or, 1,157 instrument landings may use the V–27 airway annually. Assuming a similar number of takeoffs during

[40]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[41]Ibid.

[42]Ibid.

instrument conditions, approximately 2,314 flights will use Airway V–27 annually that may pose a hazard to the proposed facility.[43]

The FAA database shows that approximately 7,560 landings and takeoffs by general aviation aircraft took place monthly at San Luis Obispo County Regional Airport over the 4-year period of 1998–2001.[44] During the same period, an average of 1,430 flights, which includes local and itinerant general aviation and military flights, landed at San Luis Obispo County Regional Airport annually under instrument conditions.[45] PG&E again assumed that approximately 65 percent of the general aviation traffic is departing to or approaching from the south. Therefore, PG&E considered that approximately 2 × (1,430 × 0.65), or 1,860 operations (landings and takeoffs) took place annually under instrument conditions.[46]

The CREPE and CADAB intersections are approximately 18 km [11 mi] and 34 km [21 mi] from the proposed site. Holding patterns at both these intersections place the aircraft further away from the proposed site.[47]

Since the Morro Bay Very-High Frequency Omnidirectional Range Navigation System is used for missed approaches to San Luis Obispo County Regional Airport, PG&E estimated that 5 percent of all instrument landing approaches are missed, and each airport remains in the holding pattern for 10 passes.[48] Therefore, commercial aircraft traffic increases by an additional 579 (2,314/2 × 0.05 × 10) annual flights and general aviation aircraft traffic by 465 [1,860/2 × 0.05 × 10] additional annual flights. In response to an RAI, PG&E[49] stated that the assumption of 5 percent of all instrument landing approaches are missed is conservative based on discussions with the personnel at the control tower of San Luis Obispo County Regional Airport regarding the specific approaches available to the airport. Additionally, discussions with pilots of commercial and private aircraft support this conclusion. San Luis Obispo County Regional Airport has limited landing facilities. Most instrument approaches are near minimum weather requirements for using the visual flying rule and result in a visual landing under an instrument flying rule approach. Essentially, zero landing misses take place under this type of approach to the airport. Runway 11 is the only runway available with a precision instrument landing system. If wind and fog results in downwind landing on Runway 11, commercial aircraft will not depart San Luis Obispo County Regional Airport.[50]

---

[43]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[44]Ibid.

[45]Ibid.

[46]Ibid.

[47]Ibid.

[48]Ibid.

[49]Ibid.

[50]Ibid.

PG&E[51] states that aircraft approaching from the south and not during weather classified as instrument meteorological conditions will fly to the CADAB intersection and will land on Runway 29 under visual control. These aircraft do not generally use Airway V–27 while landing at San Luis Obispo County Regional Airport. However, when San Luis Obispo County Regional Airport is under instrument meteorological conditions, all aircraft arriving from the south will use Runway 11 approach, if the ceiling is below 270 to 330 m [900 to 1,100 ft] depending on the aircraft type. This approach uses V-27. However, if the ceiling is above 270 to 330 m [900 to 1,100 ft], the pilot may also use the Runway 90 approach, which does not use V–27. Consequently, a major portion of the aircraft approaching San Luis Obispo County Regional Airport from the south (PG&E has estimated it to be approximately 65 percent) do not use V–27 to land; however, PG&E has conservatively assumed that all aircraft approaching from the south use Airway V–27.

V–27 has a width of 12.8 km [8 statute mi] with a center approximately 8 km [5 mi] from the proposed site. Consequently, the proposed facility site is 1.6 km [1 statute mi] from the edge of V–27 airway with effective width equal to 16 km [10 statute mi].

PG&E assumed that the wingspan of commercial aircraft is 29.9 m [98 ft] with a skid distance of 213 m [700 ft] and cotangent of the impact angle, cot ϕ, equal to 10.2[52]. Using length, width, and height of the facility as 152, 32, and 6.1 m [500, 105, and 20 ft] PG&E estimated the effective area of the facility to be 0.0580 km^2 [0.0224 mi^2] for commercial aircraft, using the formula given in DOE Standard DOE–STD–3014–96 (U.S. Department of Energy, 1996). Using a wingspan of 22.3 m [73 ft], skid distance of 213 m [700 ft], and cot ϕ of 10.2, PG&E estimated the effective area of the facility to be 0.0554 km^2 [0.0214 mi^2] for general aviation aircraft.[53]

Use of 213 m [700 ft] as the skid distance by PG&E is based on the layout of the proposed facility[54]. The proposed facility is surrounded by hills on three sides, which limits the potential skid distance by a crashing aircraft to reach to the items important to safety. The fourth side is protected by a drop in the terrain with a slope greater than 1:1 (PG&E SAR, 2002, Figure 2.2-1).

PG&E[55] assumed a crash rate of 2.5×10^{-10} per km [4×10^{-10} per mi] for commercial aircraft and 0.97×10^{-7} per km [1.55×10^{-7} per mi] for general aviation aircraft flying in this corridor. The crash rate for commercial aircraft is based on the suggested value in Section 3.5.1.6 of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981b). Additionally, PG&E used Kimura, et. al (1996) to select the crash rate for General Aviation aircraft.

Based on the above parameters and using the formula given in NUREG–0800, Section 3.5.1.6 (U.S. Nuclear Regulatory Commission, 1981b), PG&E estimated the annual crash frequency

---

[51]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[52]Ibid.

[53]Ibid.

[54]Ibid.

[55]Ibid.

onto the proposed facility by the commercial aircraft to be 2.59×10^{-9}.[56] Similarly, the annual crash frequency of general aviation aircraft is estimated by PG&E to be 7.7×10^{-7}. Therefore, the total crash frequency by aircraft flying Airway V–27 is 7.72×10^{-7} per year[57].

Staff Analysis: Staff consulted the FAA database <http://www.apo.data.faa.gov/faaatadsall.htm> to verify, independently, the number of annual flights by both commercial and General Aviation aircraft approaching San Luis Obispo County Regional Airport during instrument meteorological conditions. Staff confirms that the 4-year (1998–2001) average of flights during instrument meteorological conditions for both types of aircraft are acceptable. Additionally, inclusion of 2002 data would somewhat decrease the annual average for both commercial and General Aviation aircraft. Therefore, consideration of information from 1998–2001 is reasonable.

Commercial Aviation

San Luis Obispo County Regional Airport is served primarily by turboprop or smaller aircraft for the commercial or air taxi traffic.[58] The maximum capacity of these aircraft is 41 people with a maximum gross weight of 13,608 kg [30,000 lb]. Although PG&E used the crash rate of commercial aircraft equal to 4×10^{-10} per mile, as suggested in Section 3.5.1.6 of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981b), the staff used a crash rate of 9.28×10^{-10} per km [5.801×10^{-10} per mi], as given in Table 2.13 of Kimura, et al. (1996) for off-airport crashes with destroyed aircraft or aircraft that sustained substantial damage to the airframe as a result of the crash. Staff considers this crash rate to be more appropriate for the type of aircraft under consideration. A search of the website <http://www.sloairport.com/flightinfo.html> shows that certified air carriers operate at this airport. A certified air carrier is an air carrier possessing a Certificate of Public Convenience and Necessity issued by the U.S. Department of Transportation in accordance with 14 CFR Part 121 to operate scheduled air services (Kimura et al., 1996). The information obtained by the staff independently from the websites <http://airnav.com> and <http://www.gcr1.com/5010WEB/default.htm> indicates that the traffic at San Luis Obispo County Regional Airport does not have any air-taxi operations, rather it has commercial operations. Therefore, the staff used the crash rate of 9.28×10^{-10} per km [5.801×10^{-10} per mi] as the crash rate appropriate for commercial aircraft operating at San Luis Obispo County Regional Airport.

Although PG&E[59] stated that San Luis Obispo County Regional Airport is primarily serviced by turboprop or smaller aircraft for commercial traffic, PG&E used a wingspan of 29.4 m [98 ft], as suggested for air carriers in commercial aviation in Table B–16 of DOE Standard DOE–STD–3014–96 (U.S. Department of Energy, 1996). This table suggests that the wingspan for turboprop aircraft, classified as a General Aviation aircraft, is 22.3 m [73 ft]. Therefore, use

---

[56]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[57]Ibid.

[58]Ibid.

[59]Ibid.

of a higher value for wingspan will produce a larger estimate of the effective area, and therefore, it is conservative. Staff used a wingspan of 29.9 m [98 ft] for commercial aviation. Additionally, staff considered all General Aviation aircraft to be the turboprop type, which has the largest wingspan of all General Aviation aircraft types (U.S. Department of Energy, 1996, Table B–16).

Table B–17 of the DOE standard (U.S. Department of Energy, 1996) provides the suggested values for the mean of the cotangent of the impact angle (cot ϕ). For commercial aviation aircraft, the suggested value is 10.2. DOE (1996) recommends cot ϕ equal to 8.2 for General Aviation aircraft. Mean skid distances for commercial and General Aviation aircraft are 439 and 18 m [1,440 and 60 ft], as per Table B–18 of DOE Standard (U.S. Department of Energy, 1996). As argued by PG&E, a commercial aircraft does not have enough space around the proposed facility to skid for a distance of 439 m [1,440 ft] because of the topography surrounding the proposed facility. Staff agrees with this conclusion and has used a skid distance of 213 m [700 ft] as appropriate skid distance for commercial aircraft in the calculation. Nevertheless, staff also used 439 m [1,440 ft] in the calculation to test the sensitivity of the skid distance parameter.

Using a wingspan of 29.9 m [98 ft], cot ϕ of 10.2, and a skid distance of 213 m [700 ft], the staff estimates the effective area of the proposed facility to be 0.058 km^2 [0.0224 mi^2]. Using a skid distance of 439 m [1,440 ft], however, the estimated area increases to 0.0997 km^2 [0.0385 mi^2] for commercial aircraft. Using a wingspan of 22.3 m [73 ft], cot ϕ of 8.2, and skid distance of 18 m [60 ft], the effective area is 0.01844 km^2 [0.00712 mi^2]. As discussed before, the width of the airway is 16 km [10 mi]. Based on this information, staff estimates that the annual frequency of a crash of a commercial aircraft onto the proposed facility, using the formula given in NUREG–0800 Section 3.5.1.6 (U.S. Nuclear Regulatory Commission, 1981b), is approximately 3.8×10^{-9} with a skid distance of 213 m [700 ft]; however, assuming a skid distance of 439 m [1,440 ft], which is not considered realistic given the topography surrounding the proposed facility, the probability of a commercial aircraft crash increases to approximately 6.5×10^{-9} per year.

General Aviation

PG&E stated that the General Aviation aircraft using the airport and airways near the proposed facility includes small single- and dual-engine aircraft, and small corporate aircraft powered by either propeller or jet. These aircraft with an average gross weight of less than 5,670 kg [12,500 lb] have a capacity of maximum eight people. Kimura, et al. (1996) provide crash rate per flight mile for single- and multi-engine reciprocating, turboprop and turbojet, rotary wing with either reciprocating or turbine engine aircraft. Because the proportion of these aircraft is not known, the staff considers use of the crash rate of 2.48×10^{-7} per km [1.550×10^{-7} per mi] for all powered aircraft (total powered aircraft in Kimura, et al., 1996) is appropriate. As a part of the sensitivity analysis, the staff also used the crash rates equal to 2.416×10^{-7} per km [1.510×10^{-7} per mi] for all fixed-wing (single- and multi-engine reciprocating, turboprop and turbojet) aircraft and 5.669×10^{-7} per km [3.543×10^{-7} per mile] for all rotary-wing (reciprocating or turbine engine) aircraft. Additionally, a wingspan of 22.3 m [73 ft] has been used. As discussed before, this is a conservative estimate of the actual wingspan as the typical wingspan of a general aviation aircraft is given as 15.2 m [50 ft], except for a turboprop aircraft, which has a wingspan of 22.3 m [73 ft] (U.S. Department of Energy, 1996). Additionally, a skid distance of 18 m [60 ft] and cot ϕ of 8.2 have been used to estimate the effective area of the proposed facility. The effective area of the proposed facility has been estimated to be

0.01844 km^2 [0.00712 mi^2]. Therefore, staff estimates the annual frequency of crash of a General Aviation aircraft onto the proposed facility, using the formula given in NUREG–0800, Section 3.5.1.6 (U.S. Nuclear Regulatory Commission, 1981b), is approximately 2.6×10^{-7} assuming the crash rate for total powered General Aviation aircraft. Assuming the crash rates for all fixed-wing and all rotary-wing aircraft, the estimated annual frequencies of crash of a General Aviation aircraft onto the proposed facility are approximately 2.5×10^{-7} and 5.9×10^{-7} per year.

The staff reviewed the information and analysis presented by the applicant with respect to potential hazards of aircraft flying Victor Airway V-27 to land at or depart from San Luis Obispo County Regional Airport. The staff found them acceptable because

- Adequate information has been presented to describe the potential hazards.
- Appropriate basis has been provided for the assumed crash rates for both commercial and General Aviation aircraft.
- Appropriate bases have been provided for the assumed number of flights of each type of aircraft in the vicinity of the proposed facility using this flying corridor.
- Conservative values of crash parameters have been used to estimate the annual crash frequencies of different types of aircraft.

Aircraft Not Landing or Departing San Luis Obispo County Regional Airport

As discussed before, V–27 is a federal airway also used by aircraft flying between the Santa Barbara and Big Sur areas. These aircraft do not land at San Luis Obispo County Regional Airport. The majority of the aircraft in V–27 fly at an altitude of 3,333 m [10,000 ft] above MSL; however, some smaller aircraft may fly at elevations as low as 1,050 m [3,500 ft]. Based on information from the FAA, PG&E[60] estimates that mostly commercial aircraft fly in this airway at a rate of approximately 20 per day or, 7,300 flights per year[61]. Using a crash rate of 6.4×10^{-10} per flight km [4.0×10^{-10} per flight mi] and an effective area of 0.058 km^2 [0.0224 mi^2], PG&E[62] estimated the annual frequency of aircraft flying in this part of V–27 crashing onto the proposed facility would be 6.53×10^{-9}.

Staff Analysis: Staff only estimated the annual crash frequency of aircraft with a skid distance of 213 m [700 ft], and a crash rate of 9.282×10^{-10} per flight km [5.801×10^{-10} per flight mi] as the contribution of this activity to the overall crash frequency is relatively minor. Using the methodology given in NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981b), staff estimated crash frequency to be approximately 9.5×10^{-9} per year.

---

[60]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[61]Ibid.

[62]Ibid.

The staff reviewed the information and analysis presented by the applicant with respect to potential hazards of aircraft using Victor Airway V–27 to transit between Santa Barbara and Big Sur areas. The staff found them acceptable because

- Information presented to describe the potential hazards is adequate.
- Appropriate basis has been provided for the assumed crash rate.
- Appropriate basis has been provided for the assumed number of flights of each type of aircraft in the vicinity of the proposed facility.

Aircraft Flying in Military Training Route VR–249

VR–249 is a military training route. Aircraft can fly at any elevation up to 3,333 m [10,000 ft]. Flight through this route requires at least 8 km [5 mi] of visibility with a ceiling at 900 m [3,000 ft]. Aircraft using this route usually remain offshore. They do not fly directly over the proposed facility and the DCPP (Pacific Gas and Electric Company, 2002).

A majority of the aircraft that flew through VR–249 in the period of September 2001 to September 2002 were F/A–18s.[63] Additionally, a limited number of F-16s, C-130s, and EA6B aircraft and some helicopters used this route. Based on the information obtained by PG&E from the Naval Air Station at Lenore (Pacific Gas and Electric Company, 2002), bombs are not carried onboard a majority of the aircraft that fly VR–249, although air-to-air missiles and cannon/machine guns may be carried. The amount of explosive charges in these armaments is too small to pose a hazard to the proposed facility (Pacific Gas and Electric Company, 2002).

The route VR–249 is used by military aircraft quite infrequently, approximately 50 flights annually[64]. Additionally, aircraft fly near the proposed facility area in normal flight mode. To be conservative, PG&E assumed that approximately 75 flights use this route in a year.

PG&E assumed a wingspan of 33.5 m [110 ft] of F/A–18 aircraft[65]. Additionally, a skid distance of 213 m [700 ft] and cot ϕ of 10.2 have been assumed by PG&E. The calculated effective area of the proposed facility is 0.059 km^2 [0.0228 mi^2][66].

PG&E was not able to obtain specific crash information for F/A–18 aircraft to develop a crash rate in normal inflight mode[67]. As a result, PG&E assumed a crash rate of 4.378×10^{-8} per km [2.736×10^{-8} per mi] for all types of aircraft flying in this corridor based on the crash rate of

---

[63]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[64]Ibid.

[65]Ibid.

[66]Ibid.

[67]Ibid.

F–16 developed for the license application for the Private Fuel Storage Facility (Private Fuel Storage Limited Liability Company, 2002?). The implied assumption is that the crash rate of Fw 16 aircraft would be similar to F/A–18 aircraft.

The centerline of the route VR–249 is approximately 3.2 km [2 mi] offshore. PG&E[68] assumed the width of the route for estimating the crash hazard as 1.6 km [1 mi]. The crash hazard has been estimated using these parameters as 4.68×10^{-8} per year.

Staff Analysis: The majority of the aircraft flying through the route, VR–249, are F/A–18s[69]. Staff used the information given in Tables B–16 through B–18 of the DOE Standard (U.S. Department of Energy, 1996) to estimate the effective area of the proposed facility assuming all aircraft are either F–16s, F/A–18s, C–130, or EA–6B. Both F–16s and F/A–18s are high-performance small aircraft with wingspans of 10.0 m [32 ft 10 in] and 13.62 m [45 ft]. Suggested value for cot ϕ is 10.4 and the skid distance is 136 m [447 ft] for both these aircraft (U.S. Department of Energy, 1996). It should be noted that C–130s are transport aircraft and should be categorized as large military aircraft. The EA–6B is a twin-engine aircraft used for electronic countermeasures and is based on the airframe of A-6 aircraft. It has been categorized as a small military aircraft for estimating the effective area of the proposed facility. C–130s have a wingspan of 40.4 m [132 ft 7 in]. EA–6Bs have a wingspan of 16.2 m [53 ft]. A skid distance of 112 m [368 ft] and cot ϕ of 9.7 have been used for C–130 aircraft. Similarly, a skid distance of 136 m [447 ft] and a cot ϕ of 10.4 have been used for EA–6B aircraft (U.S. Department of Energy, 1996). The estimated effective area of the proposed facility are 0.0386 km^2 [0.0149 mi^2] for F–16s, 0.0396 km^2 [0.0153 mi^2] for F/A–18s, 0.0401 km^2 [0.0155 mi^2] for EA–6Bs, and 0.0409 km^2 [0.0158 mi^2] for C–130s. Therefore, a value of 0.0396 km^2 [0.0153 mi^2], appropriate for F/A–18s, has been used by the staff. Use of the effective area for any other aircraft would make an insignificant difference in the estimated annual frequency of aircraft crash onto the proposed facility.

Staff searched the website <http://www.chinfo.navy.mil/navpalib/factfile/aircraft/air-fa18.html> of the U.S. Navy and found that the F/A–18 is a twin-engine aircraft. It is expected that the crash rate of a twin-engine, high-performance aircraft would be less than a single-engine aircraft, such as an F–16. The crash rate given in Table 4.8 of Kimura, et al. (1996) for F–16 and F–15 (a twin-engine aircraft) indicates that the F–15 has a smaller crash rate than F–16. Therefore, it is expected that the crash rate of an F/A–18 would be at most equal to F–16 aircraft. Therefore, a crash rate of 4.378×10^{-8} per km [2.736×10^{-8} per mi] is acceptable because any other specific information is not available. Additionally, a crash of a military aircraft traversing route VR–249 would produce a small contribution to the overall aircraft crash hazard.

Although the centerline of the route VR–249 is approximately 3.2 km [2 mi] offshore, PG&E assumed the width of the route for estimating the crash hazard as 1.6 km [1 mi]. This assumption is conservative because it places all 75 flights in a 1.6-km- [1-mi-] wide corridor centered over the proposed site for crash hazard estimation purpose. In reality, some of the aircraft would fly further away from the proposed facility site. Staff also used a width of the route

---

[68]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[69]Ibid.

equal to 1.6 km [1 mi]. Using these parameters, the estimated crash hazard of aircraft flying route VR–249 is approximately at 3.1×10^{-8} per year.

The staff reviewed the data and analysis presented by the applicant with respect to the potential hazards of aircraft flights in military training route VR–249. The staff found them to be acceptable because

- Adequate information has been presented to describe the potential hazard.
- Acceptable methodology has been used to estimate the crash potential.
- Although PG&E could not obtain reliable information to estimate the crash rate of F/A–18 aircraft, it used the crash rate of another similar high-performance aircraft, the F–16. Being a single-engine aircraft, the F–16 is expected to have a higher crash rate than the F/A–18, which is a twin-engine aircraft. Nevertheless, PG&E conducted a sensitivity analysis by doubling or tripling the crash rate of the F/A–18 used. Because the number of aircraft flying annually through this route is quite small, the crash rate has only a minor effect on the cumulative aircraft crash hazard of the proposed facility.

Probability Acceptance Criterion for Aircraft Crash Hazards for Diablo Canyon Independent Spent Fuel Storage Facility

NUREG–0800 Section 3.5.1.6, "Aircraft Hazards" (U.S. Nuclear Regulatory Commission, 1981b) provides the methodology to estimate the annual frequency of a crash of an aircraft onto a nuclear power plant. An operating nuclear power plant requires active systems to control the dynamic nuclear and thermal processes that occur in the conversion of nuclear reactions into thermal power. In the event of a mishap, large amounts of thermal energy within the reactor core can be affected. Emergency cooling systems are provided as part of a reactor facility design to avoid core damage or meltdown and the release of radioactive material into the environment.

Compared to a nuclear reactor facility, an ISFSI is a relatively passive system that does not have complex control requirements and that has contents with relatively low thermal energy. Therefore, potential fuel damage and the associated radioactive source terms from a potential accident are significantly less than those expected from a potential accident at a nuclear reactor facility. As a result, the estimated consequences from a potential accident at an ISFSI are less severe than from a potential accident at a nuclear reactor facility. Therefore, the staff concludes that a frequency of 1×10^{-6} crashes per year is an appropriate acceptance criterion for evaluating aircraft crash hazards at the proposed facility.

Summary of Review and Discussion

PG&E examined past and present activities in connection with potential hazards from the crash of both civilian and military aircraft flying in the vicinity of the proposed facility. The activities examined include aircraft taking off and landing at San Luis Obispo County Regional Airport, Oceano County Airport, Camp San Luis Obispo Heliport, and Vandenberg Air Force Base; aircraft flying Airway V–27; and military aircraft flying in route VR–247. The applicant provided sufficient information and used acceptable methods to evaluate the potential hazard to the

proposed facility from an aircraft crash. The staff reviewed the scenarios, data, information, and analyses presented by PG&E in connection with the proposed facility and also carried out independent confirmatory analyses in selected cases, as presented in the previous section of this SER. The confirmatory analyses relied on assumptions different from those applied by PG&E.

Summarizing the staff review, the crash frequencies for aircraft are given in Table 15-2. As indicated in the discussion of aircraft hazards within this section, these frequencies are estimated on the basis of several elements that determine the overall likelihood that each specific type of aircraft operation may lead to an impact at the proposed facility. Typically, these elements include measures that reflect traffic density (e.g., flights per year), a crash rate (e.g., crashes per mile), effective target area, as well as width of the flying corridor. Other factors, such as human errors in aircraft design, fabrication, or maintenance, also influence the estimated probabilities but have not been addressed explicitly since their effects are inherently taken into account through the use of historically established crash rate data.

Table 15-2. Estimated annual frequency of crashes at Diablo Canyon Independent Spent Fuel Storage Installation

Source	Estimated Annual Frequency (Crashes/Year)	
	Pacific Gas and Electric*	U.S. Nuclear Regulatory Commission
Aircraft taking off and landing at San Luis Obispo County Regional Airport	0	~0
Aircraft taking off and landing at other nearby airports	0	~0
Aircraft flying Airway V–27 and landing at or departing from San Luis Obispo County Regional Airport •Commercial Aviation •General Aviation	 2.59×10^{-9} 7.7×10^{7}	 3.8×10^{-9} 2.5×10^{-7} to 5.9×10^{7}
Aircraft flying Airway V–27 and not landing at or departing from San Luis Obispo County Regional Airport	6.53×10^{-9}	9.5×10^{-9}
Aircraft flying military training route VR–249	4.68×10^{-8}	3.1×10^{-8}
Cumulative Aircraft Crash Hazard	8.25×10^{-7}	2.9×10^{-7} to 6.3×10^{-7}

*Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

The estimated crash frequency values determined by the staff, as listed in Table 15-2, may be different from those determined by PG&E[70] because of the sensitivity or confirmatory calculations performed by the staff. PG&E has used more conservative values than suggested in the DOE Standard (U.S. Department of Energy, 1996) for skid distance of a crashing general aviation aircraft. Consequently, the calculated effective area and, in turn, the estimated annual crash frequency are higher and more conservative. The values determined by PG&E have been accepted by the staff as reasonable. Based on the information presented in Table 15-2 and the threshold probability criterion of 1×10^{-6} crashes per year, the staff concludes that the probability of crash for both civilian and military aircraft and ordnance at the PG&E site is acceptable.

Future Developments

PG&E estimated the projected growth of civilian flights based on the FAA long-range forecast (Federal Aviation Administration, 1999). Commercial aircraft operations include air carrier and commuter/air taxi takeoffs and landings at all United States towered and nontowered airports. Based on the FAA forecasts, the commercial aircraft operations are projected to increase from 28.6 million in 1998, to 36.6 million in 2010, and to 47.6 million in 2025. Therefore, commercial aviation operations in the United States are projected to increase by 66 percent by 2025.

The annual general aviation operations (takeoffs and landings) at all towered and nontowered airports in the United States are projected to increase from 87.4 million in 1998 to 92.8 million in 2010 and to 99.2 million in 2025 (Federal Aviation Administration, 1999). Therefore, the FAA projects an increase of general aviation traffic by 14 percent by 2025.

PG&E has discussed the long-term trend of military aviation to project the estimated aircraft crash probability onto the proposed facility. The FAA predicts that the military air traffic would not increase appreciably, if at all, in the foreseeable future. Based on the projection of the FAA (Federal Aviation Administration, 1999), the number of military aircraft handled at the FAA en route to traffic control centers would remain constant at 4.2 million in 1998–2025.

Based on the estimated annual frequencies listed in Table 15-2 and increase of commercial and general aviation traffic projected by the FAA, the annual frequency of aircraft crash onto the proposed facility would increase in 2025 to 9.40×10^{-7} using the frequencies estimated by PG&E. Applying these growth factors to the estimated crash probability of commercial and General Aviation aircraft, the staff estimated the crash frequency by 2025 would be 3.4×10^{-7} to 7.2×10^{-7} per year from 2.9×10^{-7} to 6.3×10^{-7} as shown in Table 15-1.

Conclusion

Based on the information and analysis provided by PG&E, the staff concludes that the cumulative probability of a civilian or military aircraft crashing at or affecting the proposed facility is below the threshold probability criterion of 10^{-6} per year. Therefore, there is reasonable assurance that civilian or military air crashes would not pose a hazard to the proposed facility.

---

[70]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

15.1.2.12 Accidents at Nearby Sites—Missile Testing at Vandenberg Air Force Base

Vandenberg Air Force Base is located approximately 56 km [35 mi] south-southeast of the proposed facility. Approximately 15 to 20 missiles are tested in a year at this Air Force Base. Missiles are fired in directions ranging from due west to southeast. Therefore, the flight paths of these missiles do not come near the proposed facility[71]. Additionally, intercontinental ballistic missiles are tested at Vandenberg Air Force Base. They are launched from sites at the northern part of the base and typically fly due west. Typical launches for spacelift missions are carried out at sites on the southern part of the base and fly in a southerly direction. Polar orbit launches at this base are also carried out in a southerly direction. Based on the information from the Base Chief Safety Officer, the most northerly missile launch site is approximately 40 km [25 mi] south of the proposed facility. Vandenberg Air Force Base is also a designated alternate landing site for space shuttles, although the base has not been used yet for that purpose. The landing approach is west to east and does not bring the shuttle within 48 km [30 mi] of the proposed facility (Pacific Gas and Electric Company, 2002). Therefore, the planned flight paths of tests from different missiles launched from Vandenberg Air Force Base and space shuttles are always away from the proposed facility site.

Only a small fraction of the missiles deviate from the intended trajectories[72]. If a missile after launch deviates from its planned flight path, the missile is destroyed before the debris path exceeds a narrow preplanned window[73]. Therefore, the probability of missiles launched from Vandenberg Air Force Base striking any safety-related SSCs is negligibly small.

The staff reviewed the information with respect to potential hazards of missile testing at Vandenberg Air Force Base. The staff found the information acceptable because

- Verifiable information from the U.S. Air Force was used to determine the number of missile tests carried out annually and their intended flight paths.
- Intended flights paths are always away from the proposed facility site.
- The U.S. Air Force uses avoidance as one of the primary safety measures to protect facilities.

Based on the foregoing information, there is reasonable assurance that different missile tests and potential space shuttle landings at Vandenberg Air Force Base will not pose a hazard to the proposed facility because (1) the selected flight paths are away from the proposed facility site and (2) several low-probability events need to take place before a missile or the space shuttle would hit the proposed facility.

---

[71]Womack, L.F. Letter (February 14) DIL–03–002, Enclosures 1–4 to U.S. Nuclear Regulatory Commission. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

[72]Ibid.

[73]Ibid.

15.1.2.13 Leakage Through Confinement Boundary

Section 8.2.7 of the SAR (Pacific Gas and Electric Company, 2002) evaluates the potential consequences of a leakage through a confinement-boundary accident. The potential consequences of this postulated accident are determined by assuming that 100 percent of the cladding for the fuel rods have ruptured and the MPC pressure boundary has been breached. The staff has previously determined that the methodology used to assess this postulated accident is acceptable and that there are no consequences that affect the public health and safety so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to. Moreover, NUREG–1536 (U.S. Nuclear Regulatory Commission, 1997, Chapter 7, Section V.2) indicates that casks closed entirely by welding do not require seal monitoring. The MPC, which is the confinement system for the HI-STORM 100 System, is closed using a welded seal. As a result, the staff finds the applicant proposal not to provide monitoring of the confinement barrier for the HI-STORM 100 System acceptable because the casks will be loaded, welded, inspected, and tested in accordance with appropriate procedures.

15.1.2.14 Loading of an Unauthorized Fuel Assembly

Section 8.2.9 of the SAR (Pacific Gas and Electric Company, 2002) indicates that loading of an unauthorized fuel assembly into the MPC will not occur because of the technical specifications and administrative procedures that will be implemented during loading operations. The review of these technical specifications and administrative procedures is provided in Chapters 10 and 16 of this SER.

15.1.2.15 Partial Blockage of Multi-Purpose Canister Vent Holes

Section 8.2.13 of the SAR (Pacific Gas and Electric Company, 2002) evaluates the potential consequences of the partial blockage of the MPC vent holes. The potential consequences of this postulated accident were determined by assuming that only the minimum semicircular area of the vents are credited in the thermal models. The staff previously determined that the methodology used to assess this postulated accident is acceptable and partial blockage of the MPC vent holes has no effect on the structural, confinement, and thermal analyses of the MPC so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to.

15.1.2.16 100-Percent Fuel Rod Rupture

Section 8.2.14 of the SAR (Pacific Gas and Electric Company, 2002) evaluates the potential consequences of 100-percent fuel rod rupture within the MPC. The potential consequences of this postulated accident were determined by assuming that the fission-product gases and fill gas are released from the fuel rods into the MPC cavity. The staff has previously determined that the methodology used to assess this postulated accident is acceptable and 100-percent fuel rod rupture within the MPC has no effect on the shielding, criticality, and thermal analyses of the MPC so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to. The internal MPC gauge pressure of 1.18 MPa [170.8 psi], however, does approach the accident-design pressure of 1.38 MPa [200 psi].

15.1.2.17 Transmission Tower Collapse

Section 8.2.16 of the SAR addresses the potential collapse of two 500-kV transmission towers that are in the vicinity of the dry storage area and the CTF. The transporter will be designed to protect the transfer cask from direct impact of a collapsing tower. As a result, an analysis of the collapsing tower with the HI-TRAC 125 Transfer Cask is not necessary.

There were two tower-collapse scenarios evaluated (Holtec International, 2001f). The first scenario was a tower collapse onto the CTF, with the tower directly impacting the lid of the MPC, which has been lowered into the HI-STORM 100 System storage overpack located within the confines of the CTF shell. The second scenario was a tower collapse onto the HI-STORM 100SA storage cask anchored to the dry storage area pad.

Using an explicit finite element modeling method, it was determined that the maximum impact force on the MPC lid was 1.9 MN [427 kips] and, for the anchored HI-STORM 100SA storage cask, 2.4 MN [534 kips]. In the case of the MPC lid, this impact force is bounded by previously evaluated tornado missile impact loads (U.S. Nuclear Regulatory Commission, 2002a,b). For the anchored HI-STORM 100SA storage cask, the impact force is predominantly oriented in the vertical direction. The horizontal component of the tower collapse impact force on the anchored HI-STORM 100SA storage cask, 0.4 MN [93 kips], is bounded by previously evaluated tornado missile impact loads (U.S. Nuclear Regulatory Commission, 2002a,b). The vertical component of the tower collapse impact force on the anchored HI-STORM 100 SA storage cask, when converted into an equivalent gravity load, is bounded by the previously reviewed and accepted equivalent gravity load for a cask drop (i.e., 45 g) (U.S. Nuclear Regulatory Commission, 2002a,b).

Even though an analysis of a collapsing tower impact with the HI-TRAC 125 Transfer Cask was not performed, the potential impact forces would be similar to those calculated for the MPC and anchored HI-STORM 100SA storage cask. Because these impact loads are bounded by previously reviewed and accepted loading conditions (U.S. Nuclear Regulatory Commission, 2002a,b), the staff has determined that a separate analysis is not needed.

15.1.2.18 Nonstructural Failure of a Cask Transfer Facility Lift Jack

Section 8.2.17 of the SAR evaluates a postulated failure of a CTF lift jack. The CTF lifting mechanism is configured with three lifting jacks, and the postulated lift jack failure evaluation assumes that only one of these will fail at any given time. If the failed mechanism cannot be repaired within 22 hours, which corresponds to the time for the short-term fuel cladding temperature limits to be reached because of the diminished convective cooling efficiency of the HI-STORM 100SA storage cask when located within the CTF, the MPC will be returned to the HI-TRAC 125 Transfer Cask and the storage cask removed from the CTF so the necessary repairs can be made.

The design of the CTF lifting mechanism is such that the three lifting jack power screws are always loaded in tension. Because of this tension loading-only design feature, buckling failure of the lifting jack power screws, either singly or in combination, is unlikely.

15.1.2.19 Accidents Associated with Pool Facilities

The proposed facility will use dry storage technology, and there will be no pool at the proposed facility. Therefore, accidents associated with pool facilities is not applicable.

15.1.2.20 Building Structural Failure and Collapse onto Structures, Systems, and Components

Section 4.4.5 of the SAR evaluates the CTF for response to the design criteria identified in Chapter 3, "Principal Design Criteria" of the SAR. The CTF is designed to survive these events. (See also SER Section 15.2.2.18, "Transmission Tower Collapse.") Therefore, an accident involving structural failure of the facility is not applicable.

15.1.2.21 Hypothetical Failure of the Confinement Boundary

The HI-STORM 100 System MPC is a seal-welded pressure vessel, designed, fabricated, and tested in accordance with the ASME. The MPCs have redundant welds to ensure that radioactive fuel is confined. The proposed facility SAR and HI-STORM 100 System FSAR have demonstrated that the MPC would maintain its integrity and the fuel would be adequately protected under site-specific and generic design-basis normal, off-normal, and accident conditions. As discussed in Chapter 9 of this SER, the dose (at the owner-controlled area boundary) calculated from a hypothetical failure of the confinement boundary is below the dose limits specified in 10 CFR §72.106(b).

15.2 Evaluation Findings

The applicant has provided acceptable analyses of the design and performance of SSCs important to safety under credible off-normal events and accident scenarios. The following summarizes the findings of the staff that pertain to the off-normal event and accident review.

Off-Normal Events

PG&E has committed to design the cask transporter so it will have redundant drop protection features and will conform to the requirements of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980), American National Standards Institute (ANSI) N14.6 (American National Standards Institute, 1993), and ASME B30.9-1996 (ASME International, 1996). The staff previously determined that the casks can be lifted to any height necessary during transportation between the FHB/AB and CTF (U.S. Nuclear Regulatory Commission, 2002a) if these cask transporter design requirements are met. As a result, an evaluation of a cask drop less than the design allowable height is not required.

The staff has previously determined (U.S. Nuclear Regulatory Commission, 2002b) that the HI-STORM 100 System storage cask provides adequate heat removal capacity under partial vent blockage conditions so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to and the environmental characteristics of the site are bounded by the corresponding design criteria (see Section 6.1.3 of this SER). In addition, the HI-STORM 100 System CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix A) includes surveillance requirements for ensuring that the cask heat removal system is operational during storage (i.e., the air ducts

are inspected or the temperature differential between the convected cooling air exiting the outlet vents and ambient air is measured every 24 hours to ensure that the ducts are free of blockages). In the event that the HI-STORM 100SA storage cask air inlet ducts are found to be partially blocked, the blockage will be removed within one operating shift.

The staff find that the information provided regarding failure of instrumentation gave reasonable assurance that important to safety functions will not be affected for the proposed cask system or the proposed facility.

The staff find that potential vehicular impact will not impair the ability of the SSC to maintain subcriticality, confinement, and sufficient shielding of the stored fuel.

The staff find that the applicant evaluation of loss of electrical power as an off-normal event is adequate in providing assurance that Diablo Canyon ISFSI operations can be conducted without endangering the health and safety of the public.

The staff find that the applicant assessment of cask transporter off-normal operation is adequate in providing assurance that Diablo Canyon ISFSI operations can be conducted without endangering the health and safety of the public.

The staff previously determined that the HI-STORM 100SA storage and HI-TRAC 125 Transfer Casks provide adequate heat removal capacity during off-normal ambient temperature conditions so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to.

The staff previously determined that the methodology used to assess off-normal pressure within the MPC is acceptable and that there are no consequences that affect the public health and safety so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to.

Accidents

The staff has previously determined that cask tip-over events need not be considered if the preapproved HI-STORM 100SA storage cask anchorage system is used and the storage pad design specifications are met (U.S. Nuclear Regulatory Commission, 2002a,b).

PG&E has committed to design the cask transporter such that it will have redundant drop protection features and conform to the requirements of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980), ANSI N14.6 (American National Standards Institute, 1993), and ASME B30.9-1996 (ASME International, 1996). The staff has previously determined that the casks can be lifted to any height necessary during transportation between the FHB/AB and CTF (U.S. Nuclear Regulatory Commission, 2002a), if these cask transporter design parameters are met. As a result, an evaluation of a cask drop is not required.

The staff find that the information provided by the applicant is sufficient to characterize flooding as a noncredible accident at the Diablo Canyon ISFSI. As discussed in Section 2.1.4, "Surface Hydrology," of this SER, PG&E has adequately demonstrated that local natural and man-made drainage systems are sufficient to prevent flooding of the ISFSI pad site and CTF.

The staff reviewed the information provided by the applicant regarding potential fire and explosion hazards at the proposed facility. The staff find that the design bases for the SSCs needed to meet subcriticality, confinement, and shielding requirements for the stored fuel for all credible on-site fire and explosion hazards will not be exceeded.

In the event that an electrical accident should occur, the previously reviewed and accepted recovery operation procedures will be implemented (U.S. Nuclear Regulatory Commission, 2002a,b).

Staff concludes that earthquake-induced damage of the casks while in transit from the power plant to the CTF is not a credible hazard.

The staff find that the design of the CTF concrete structure and its reinforcement satisfies the applicable codes and standards for all design basis accident loads.

The staff find that the design basis loading conditions for the Diablo Canyon ISFSI are enveloped by the loading conditions considered in the HI-STORM 100 System FSAR (Holtec International, 2002). As documented in the HI-STORM 100 System SER, the structural analysis shows that the structural integrity of the HI-STORM 100 System is maintained during all credible loads. Based on the results presented in the HI-STORM 100 System FSAR, the stresses in the storage cask and anchorage structures during the most critical load combinations are less than the allowable stresses of ASME Boiler and Pressure Vessel Code, Section III (ASME International, 1995b) for the materials to be used.

The staff find that the HI-STORM 100SA storage cask anchorage system was designed to meet the ductile anchorage provision of the proposed Draft Appendix B for ACI 349-97 for the most critical load combinations.

The staff previously determined that the methodology used to assess the loss of neutron shielding accident is acceptable and the short-term fuel cladding and other component temperature limits, the MPC accident internal pressure, and the accident dose limits defined by 10 CFR §72.106 are not exceeded so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to. In the event that the HI-TRAC 125 Transfer Cask loses its neutron shielding, the previously reviewed and accepted recovery operation procedures will be implemented (U.S. Nuclear Regulatory Commission, 2002a,b).

The staff finds that the maximum reduction in ISFSI radiation shielding thickness, material shielding effectiveness, or loss of temporary shielding in all possible shielding areas caused by postulated on-site explosion events has been adequately evaluated by the applicant. Therefore, information and analysis presented by the applicant provide reasonable assurance that the dose to any individual beyond the owner-controlled area will not exceed the limits specified in 10 CFR §72.106(b) and the occupational exposures from accident recovery operations will not exceed the limits specified in 10 CFR Part 20.

The HI-STORM 100 System FSAR (Holtec International, 2002, Figure 11.2.6) indicates that a total cask decay heat load of 30 kW [102,360 BTU/hr], which bounds the cask decay heat load specified for the Diablo Canyon ISFSI, will not cause the short-term cladding temperature limit for the spent nuclear fuel to be exceeded for 45 hours under adiabatic conditions. Moreover,

the internal pressure limit for the MPC is not exceeded within the 45-hour timeframe for this condition. In the event that the HI-STORM 100 System storage cask is subjected to conditions that thermally insulate its exterior (e.g., encased within soil as the result of a landslide), the previously reviewed and accepted recovery operation procedures will be implemented (U.S. Nuclear Regulatory Commission, 2002a,b).

The staff previously determined that the methodology used to estimate the time required to reach the short-term, fuel-cladding temperature limit of spent nuclear fuel stored in the HI-STORM 100SA storage cask subjected to 100-percent blockage of the air inlet ducts is acceptable (U.S. Nuclear Regulatory Commission, 2002a,b). For the bounding values of decay heat load of 30 kW [102,360 BTU/hr] and insolation of 834 w/m^2 [800 g-cal/cm^2] per day {387 W/m^2 [123 BTU/hr-ft^2]}, the short-term cladding temperature limit for the spent nuclear fuel will not be exceeded for 72 hr when the HI-STORM 100SA storage cask air inlet ducts are 100-percent blocked. Moreover, the internal pressure limit for the MPC is not exceeded within the 72-hour time frame for this condition. Furthermore, the HI-STORM 100 System CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix A) includes surveillance requirements for ensuring that the cask heat removal system is operational during storage (i.e., the air ducts are inspected or the temperature differential between the convected cooling air exiting the outlet vents and ambient air is measured every 24 hours to ensure that the ducts are free of blockages). In the event that the HI-STORM 100SA storage cask air inlet ducts are found to be 100-percent blocked, the previously reviewed and accepted recovery operation procedures will be implemented (U.S. Nuclear Regulatory Commission, 2002a,b).

The staff reviewed the information provided by the applicant, evaluated the analyses of potential hazards from design-basis tornadoes and tornado missiles at the proposed facility, and conducted a confirmatory analysis. The staff concludes that a tornado or tornado-generated missile would not impair the ability of the SSCs to maintain subcriticality, confinement, and sufficient shielding of the stored fuel.

The staff finds that the applicant has adequately demonstrated that the cumulative probability of occurrence of civilian and military aircraft crashes, and ordnance accidents is below the threshold probability criterion of 1×10^{-6} crashes per year. As a result, the staff concludes that civilian and military aircraft crashes, and ordnance accidents at the PG&E site are noncredible events.

The staff finds with is reasonable assurance that different missile tests and potential space shuttle landings at Vandenberg Air Force Base will not pose a hazard to the proposed facility because (1) the selected flight paths are away from the proposed facility site and (2) several low-probability events need to take place before a missile or the space shuttle would hit the proposed facility.

The staff has previously determined that the methodology used to assess leakage through the confinement boundary is acceptable and that there are no consequences that affect the public health and safety so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to. Moreover, NUREG–1536 (U.S. Nuclear Regulatory Commission, 1997, Chapter 7, Section V.2) indicates that casks closed entirely by welding do not require seal monitoring. The MPC, which is the confinement system for the HI-STORM 100 System, is closed using a welded seal. As a result, the staff finds the applicant proposal not to provide monitoring of the

confinement barrier for the HI-STORM 100 System acceptable because the casks will be loaded, welded, inspected, and tested in accordance with appropriate procedures.

Section 8.2.9 of the SAR (Pacific Gas and Electric Company, 2002) indicates that an unauthorized fuel assembly will not be loaded into the MPC because of the technical specifications and administrative procedures that will be implemented during loading operations. The review of these technical specifications and administrative procedures is provided in Chapters 10 and 16 of this SER.

The staff previously determined that the methodology used to assess partial blockage of the MPC vent holes is acceptable and this postulated accident has no effect on the structural, confinement, and thermal analyses of the MPC so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to.

The staff previously determined that the methodology used to assess the potential consequences of a 100-percent fuel rod rupture within the MPC is acceptable and this postulated accident has no effect on the shielding, criticality, and thermal analyses of the MPC so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to.

The staff find that the impact loads associated with the two postulated tower-collapse scenarios are bounded by previously reviewed and accepted loading conditions (U.S. Nuclear Regulatory Commission, 2002a,b).

The design of the CTF lifting mechanism is such that the three lifting jack power screws are always loaded in tension. Because of this tension loading-only design feature, the staff find that a buckling failure of the lifting jack power screws, either singly or in combination, would not occur.

The proposed facility will use dry storage technology, and there will be no pool at the proposed facility. Therefore, accidents associated with pool facilities are not applicable.

The staff find, based on information provided by the applicant, that an accident involving structural failure of the facility is not applicable.

Based on the information provided, the staff find that a postulated failure of the confinement boundary is below the dose limits specified in 10 CFR §72.106(b) because the HI-STORM 100 System MPC is a seal-welded pressure vessel, designed, fabricated, and tested in accordance with the applicable codes and standards.

In summary, the PG&E analyses of off-normal and accident events demonstrate that the proposed facility will be sited, designed, constructed, and operated so that during all credible off-normal and accident events, public health and safety will be adequately protected. Based on analyses submitted by the applicant and independent confirmatory analyses performed by the staff, the staff finds that the proposed facility will maintain subcriticality, maintain confinement, and provide sufficient shielding for all credible off-normal events and accident scenarios consistent with the requirements of 10 CFR §72.92, §72.94, §72.98(a), §72.98(b), §72.98(c)

§72.102(f), §72.106(b), §72.122(b), §72.122(c), §72.122(h), §72.122(i), §72.122(l), §72.124(a), and §72.128(a)(2).

15.3 References

AirNav, LLC. Morganville, NJ:. http://www.airnav.com. 2003.

American Concrete Institute. *Code Requirements for Nuclear Safety Related Concrete Structures.* ACI 349-97. Detroit, MI: American Concrete Institute. 1998

Afzali, A. *Risk Assessment of Dry Cask/Spent Fuel Transportation within the DCPP Owner Controlled Area.* Calculation File No. PRA01-01, Revision 1. Avila Beach, CA: Pacific Gas and Electric Company. 2002.

American National Standards Institute. *Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More.* ANSI N14.6–1993. Washington, DC: American National Standards Institute. 1993.

ASME International. *ASME Boiler and Pressure Vessel Code, Section III, Division 1.* New York, NY: ASME. 1995b.

ASME International. *ASME Boiler and Pressure Vessel Code, Section II.* New York, NY: ASME. 1995a.

ASME International. *SLINGS Safety Standard for Cable-Ways, Cranes, Derricks, Hoists, Jacks, and Slings.* ASME B30.9–1996. New York, NY: ASME. 1996.

ENERCON Services Inc. *Cask Transfer Facility (Reinforced Concrete).* PGE–009–CALC–002, Rev 1. Tulsa, OK: ENERCON Services Inc. December 17, 2001a.

ENERCON Services Inc. *ISFSI Cask Storage Pad Seismic Analysis.* PGE–009–CALC–003, Rev 2. Tulsa, OK: ENERCON Services Inc. December 14, 2001b.

Federal Aviation Administration. *FAA Long-Range Aerospace Forecasts Fiscal Years 2015, 2020 and 2025.* FAA–APO–99–5. Washington, DC: Federal Aviation Administration. Office of Aviation Policy and Plans. 1999.

GCR & Associates, Inc. New Orleans, LA: http://www.gcr1.com/5010WEB/default.htm. 2003.

Holtec International. *Evaluation of Site-Specific Wild Fires for the Diablo Canyon ISFSI.* HI–2012615, Marlton, NJ: Holtec International. 2001a.

Holtec International. *Evaluation of Site-Specific Blasts and Explosions for the Diablo Canyon ISFSI.* HI–2002512, Rev 2. Marlton, NJ: Holtec International. 2001b.

Holtec International. *Structural Evaluation of Diablo Canyon Cask Transfer Facility.* HI–2012626, Rev 7. Marlton, NJ: Holtec International. 2001c.

Holtec International. *Analysis of Anchored HI-STORM Casks at the Diablo Canyon ISFSI.* HI–2012618, Rev 5. Marlton, NJ: Holtec International. 2001d.

Holtec International. *Design Basis Wind and Tornado Evaluation for DCPP.* HI–2012497, Rev. 1. Marlton, NJ: Holtec International. 2001e.

Holtec International. *Evaluation of Effects of Lightning and a 500-kV Line Break on Holtec Casks.* HI–2002559. Marlton, NJ: Holtec International. 2001f.

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System Revision 1 (HI-STORM 100 Cask System).* Volumes I and II, Revision 1. HI–2002444. Docket 72-1014. Marlton, NJ: Holtec International. 2002.

Kimura, C.Y. R.E. Glaser, R.W. Mensing, T. Lin, T.A. Haley, A.B. Barto, and M.A. Stutzke. *Data Development Technical Support Document for the Aircraft Crash Risk Analysis Methodology (ACRAM) Standard.* UCRL–ID–124837. Livermore, CA: Lawrence Livermore National Laboratory. 1996.

McConnell, J.W., Jr., A.L. Ayers, Jr., and M.J. Tyacke. *Classification of Transportation Packing and Dry Spent Fuel Storage System Components According to Importance to Safety.* NUREG/CR–6407. Idaho Falls, ID: Idaho National Engineering Laboratory. 1996.

National Fire Protection Association. *Fire Protection Handbook.* 18th Edition. Quincy, MA: National Fire Protection Association. 1997.

Pacific Gas and Electric Company. *Non-Linear Seismic Sliding Analysis of the ISFSI Pad.* PG&E Calculation 52.27.100.704, Rev. 0. Avila Beach, CA: Pacific Gas and Electric Company 2001a.

Pacific Gas and Electric Company. *Diablo Canyon Power Plant Final Safety Analysis Report Update.* Rev. 14. Avila Beach, CA: Pacific Gas and Electric Company. 2001b.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Safety Analysis Report.* Rev. 1. Avila Beach, CA: Pacific Gas and Electric Company. 2002.

Private Fuel Storage Limited Liability Company. Aircraft Crash Impact Crash Impact Hazard at the Private Fuel Storage Facility. Rev. 4. La Crosse, WI: Private Fuel Storage Limited Liability Company. 2000.

San Luis Obispo County Regional Airport. San Luis Obispo, CA: http://www.sloairport.com/flightinfor.html. 2003

Simiu, E. and R.H. Scanlan. *Wind Effects on Structures: An Introduction to Wind Engineersing.* Second Edition. New York, NY: John Wiley and Sons. 1986.

U.S. Department of Energy. *DOE Standard: Accident Analysis for Aircraft Crash Into Hazardous Facilities.* DOE–STD–3014–96. Washington, DC: U.S. Department of Energy. 1996.

U.S. Department of Transportation. *Traffic Safety Facts 2001: 2001 Motor Vehicle Crash Data from FARS and GES.* http://www-nrd/nhtsa.dot.gov/pdf/nrd-30/NCSA/TSFAnn/TSF2001.pdf. National Highway Traffic Safety Administration. 2003a

U.S. Department of Transportation. *Large Truck Crash Facts 2001.* http://ai.volpe.dot.gov/Carrier Research Results/PDFs/Large Truck Crash Facts 2001.pdf. Federal Motor Carrier Safety Administration. 2003b.

U.S. Navy. United States Navy Fact File: F/18-18 Hornet. Washington, DC: http://www.chinfo.navy.mil/navpalib/factfile/aircraft/air-fa18.html. 2003.

U.S. Federal Aviation Administration. Washington, DC: http://apo.data.faa.gov/faaatadsall.htm. 2003.

U.S. Nuclear Regulatory Commission. *Design Basis Tornados for Nuclear Power Plant.* Regulatory Guide 1.76. Washington, DC: U.S. Nuclear Regulatory Commission. 1974.

U.S. Nuclear Regulatory Commission. *Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants.* Regulatory Guide 1.91. Rev. 1. Washington, DC: U.S. Nuclear Regulatory Commission. 1978.

U.S. Nuclear Regulatory Commission. *Control of Heavy Loads at Nuclear Power Plants.* NUREG–0612. Washington, DC: U.S. Nuclear Regulatory Commission. 1980.

U.S. Nuclear Regulatory Commission. "Standard Review Plan Section 3.5.1.4: Missiles Generated by Natural Phenomena." Rev. 2. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.* NUREG–0800. Washington, DC: U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. 1981a.

U.S. Nuclear Regulatory Commission. "Standard Review Plan Section 3.5.1.6 Aircraft Hazards." *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.* NUREG–0800. Washington, DC: U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. 1981b.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Dry Cask Storage Systems,* NUREG–1536. Washington, DC: U.S. Nuclear Regulatory Commission. 1997.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Spent Fuel Dry Storage Facilities—Final Report.* NUREG–1567. Washington, DC: U.S. Nuclear Regulatory Commission. 2000.

U.S. Nuclear Regulatory Commission. *10 CFR Part 72 Certificate of Compliance No. 1014, Amendment 1, for the HI-STORM 100 Cask System.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. July 15. 2002a.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report.* Amendment 1. Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2002b.

16 EMERGENCY PLAN

To be provided by NRC

17 FINANCIAL QUALIFICATIONS AND DECOMMISSIONING FUNDING ASSURANCE

To be provided by NRC

18 PHYSICAL PROTECTION PLAN

To be provided by NRC

19 TECHNICAL SPECIFICATIONS

To be provided by NRC

20 CONCLUSIONS

To be provided by NRC