Diablo Canyon Independent Spent Fuel Storage Installation
Updated Final Safety Analysis Report - Revision 9
December 2021

Diablo Canyon Independent Spent Fuel Storage Installation

Final Safety Analysis Report Update

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CONTENTS

CONTENTS

CONTENTS

LIST OF CURRENT PAGES

Page No.	Revision
CONTENTS	
i thru iii	4
GLOSSARY	
1 thru 8	7
CHAPTER 1	
i thru ii	4
1.1-1 thru 1.1-5	9
1.2-1 thru 1.2-2	9
1.3-1 thru 1.3-2	9
1.4-1	0
1.5-1 thru 1.5-2	4
Table 1.1-1	0
CHAPTER 2	
i thru vii	6
2.1-1 thru 2.1-7	9
2.2-1 thru 2.2-15	7
2.3-1 thru 2.3-4	6
2.4-1 thru 2.4-4	4
2.5-1 thru 2.5-3	4
2.6-1 thru 2.6-68	4
Table 2.1-1	0
Table 2.1-2	0
Table 2.1-3	0
Table 2.1-4	0
Table 2.3-1	0
Table 2.6-1	0
Table 2.6-2	0
Table 2.6-3	0
Table 2.6-4	0
Table 2.6-5	0
Table 2.6-6, 2 Sheets	0
Table 2.6-7	0
Table 2.6-8, 2 Sheets	1
Figure 2.1-1	0
Figure 2.1-2	2
Figure 2.1-3	0
Figure 2.1-4	0
Figure 2.1-5	0
Figure 2.1-6	0
Figure 2.1-7	0
Figure 2.1-8	0
Figure 2.2-1	2
Figure 2.4-1	0
Figure 2.4-2	0
Figure 2.4-3	2
Figure 2.6-1	0
Figure 2.6-2	0
Figure 2.6-3	0
Figure 2.6-4	0
Figure 2.6-5	0
Figure 2.6-6	0
Figure 2.6-7	0
Figure 2.6-8	0
Figure 2.6-9	0
Figure 2.6-10	0
Figure 2.6-11	0
Figure 2.6-12	0
Figure 2.6-13	0
Figure 2.6-14	0
Figure 2.6-15	0
Figure 2.6-16a	0
Figure 2.6-16b	0
Figure 2.6-17a	0
Figure 2.6-17b	0
Figure 2.6-18	0
Figure 2.6-19	0
Figure 2.6-20	0
Figure 2.6-21	0
Figure 2.6-22	0
Figure 2.6-23	0
Figure 2.6-24	0
Figure 2.6-25	0
Figure 2.6-26	0
Figure 2.6-27	0
Figure 2.6-28	0
Figure 2.6-29	0
Figure 2.6-30	0
Figure 2.6-31	0
Figure 2.6-32	0
Figure 2.6-33	0
Figure 2.6-34	0
Figure 2.6-35	0
Figure 2.6-36	0
Figure 2.6-37	0
Figure 2.6-38	0
Figure 2.6-39	0
Figure 2.6-40	0
Figure 2.6-41	0
Figure 2.6-42	0
Figure 2.6-43	0
Figure 2.6-44	0
Figure 2.6-45	0
Figure 2.6-46	0
Figure 2.6-47	0
Figure 2.6-48	0
Figure 2.6-49	0
Figure 2.6-50	0
Figure 2.6-51	0
Figure 2.6-52	0
Figure 2.6-53	0
Figure 2.6-54	0
Figure 2.6-55	0
Figure 2.6-56	0
Figure 2.6-57	0
Figure 2.6-58	0
Figure 2.6-59	0
Figure 2.6-60	0
Figure 2.6-61	1
CHAPTER 3	
i thru iii	6
3.1-1 thru 3.1-6	6
3.2-1 thru 3.2-9	6
3.3-1 thru 3.3-17	6
3.4-1	4
Table 3.1-1	2
Table 3.1-2	4
Table 3.2-1	0
Table 3.2-2	0
Table 3.4-1, 2 Sheets	1
Table 3.4-2, 4 Sheets	4
Table 3.4-3	2
Table 3.4-4	2
Table 3.4-5	2
Table 3.4-6, 11 Sheets	4
Figure 3.3-1	0
CHAPTER 4	
i thru v	8
4.1-1 thru 4.1-2	6
4.2-1 thru 4.2-33	9
4.3-1 thru 4.3-9	5

LIST OF CURRENT PAGES

Page No.	Revision
4.4-1 thru 4.4-15	7
4.5-1 thru 4.5-4	2
4.6-1 thru 4.6-3	8
4.7-1 thru 4.7-6	9
Table 4.2-1	9
Table 4.2-2	4
Table 4.2-3	2
Table 4.2-4	6
Table 4.2-5, 7 Sheets	2
Table 4.3-1	2
Table 4.5-1	9
Table 4.7-1, 4 Sheets	4
Figure 4.1-1	1
Figure 4.2-1	0
Figure 4.2-2	0
Figure 4.2-3	1
Figure 4.2-5	6
Figure 4.2-7, Sheets 1-8 and 10-14	8
Figure 4.2-7, Sheet 9	9
Figure 4.2-8, 10 Sheets	2
Figure 4.2-13, 7 Sheets	8
Figure 4.2-14, 3 Sheets	8
Figure 4.4-3, 6 Sheets	2
CHAPTER 5	
i thru iii	6
5.1-1 thru 5.1-12	6
5.2-1	0
5.3-1	0
5.4-1	0
Table 5.1-1	0
CHAPTER 6	
i	0
6.1-1	0
6.2-1	0
6.3-1	0
CHAPTER 7	
i thru v	6
7.1-1 thru 7.1-6	6
7.2-1 thru 7.2-9	6
7.3-1 thru 7.3-6	6
7.4-1 thru 7.4-3	9
7.5-1 thru 7.5-4	6
7.6-1	0
7.7-1	0
Table 7.2-1A	4
Table 7.2-1B	4
Table 7.2-2A	4
Table 7.2.2B	4
Table 7.2-3A	4
Table 7.2-3B	4
Table 7.2-4A	4
Table 7.2-4B	4
Table 7.2-5A	4
Table 7.2-5B	4
Table 7.2-6A	4
Table 7.2-6B	4
Table 7.2-7	0
Table 7.2-8, 2 Sheets	0
Table 7.3-1A	6
Table 7.3-1B	6
Table 7.3-2A	6
Table 7.3-2B	6
Table 7.3-3	6
Table 7.3-4A	4
Table 7.3-4B	4
Table 7.5-1	4
Table 7.5-2	4
Table 7.5-3	4
Table 7.5-4	4
Figure 7.3-1	0
Figure 7.3-2	0
Figure 7.3-3	0
CHAPTER 8	
i thru iii	6
8.1-1 thru 8.1-15	6
8.2-1 thru 8.2-72	6
8.3-1	0
Table 8.1-1	2
Table 8.2-1	0
Table 8.2-2	2
Table 8.2-3	0
Table 8.2-4	2
Table 8.2-5	2
Table 8.2-6	2
Table 8.2-7	2
Table 8.2-8	8
Table 8.2-9	0
Table 8.2-10	2
Table 8.2-12	0
Table 8.2-13	2
Table 8.2-14	2
Table 8.3-1	0
CHAPTER 9	
i thru ii	5
9.1-1 thru 9.1-7	9
9.2-1 thru 9.2-4	4
9.3-1	0
9.4-1 thru 9.4-3	0
9.5-1	0
9.6-1	0
Figure 9.1-1	9
Figure 9.1-2	9
CHAPTER 10	
i thru iii	6
10.1-1	0
10.2-1 thru 10.2-17	7
Table 10.1-1	4
Table 10.2-1	4
Table 10.2-2, 2 Sheets	4
Table 10.2-3, 2 Sheets	4
Table 10.2-4	4
Table 10.2-5	4
Table 10.2-6	7
Table 10.2-7	6
Table 10.2-8	6
Table 10.2-9	7
Table 10.2-10	7
Table 10.2-11, 2 Sheets	4
Figure 10.2-1	0
Figure 10.2-2	0
Figure 10.2-3	0
Figure 10.2-4	0
CHAPTER 11	
i	0
11-1	0

LIST OF CURRENT PAGES

GLOSSARY

A glossary of most of the terms and acronyms used in this final safety analysis report, including their frequently used variations, is presented in this section as an aid to readers and reviewers.

Accident Events means events that are considered to occur infrequently, if ever, during the lifetime of the facility. Natural phenomena, such as earthquakes, tornadoes, floods, and tsunami, are considered to be accident events.

ALARA means as low as is reasonably achievable.

Boral is a generic term to denote an aluminum-boron carbide cermet manufactured in accordance with U.S. Patent No. 4027377. The individual material supplier may use another trade name to refer to the same product.

BPRA means burnable poison rod assembly.

Cask Transporter (or Transporter) is a U-shaped tracked vehicle used for lifting, handling, and onsite transport of loaded overpacks and the transfer cask.

CEDE means committed effective dose equivalent.

CFR means Code of Federal Regulations.

CIMIS means the California Irrigation Management Information System.

CoC means a certificate of compliance issued by the NRC that approves the design of a spent fuel storage cask design in accordance with Subpart L of 10 CFR 72.

Confinement Boundary means the outline formed by the sealed, cylindrical enclosure of the multi-purpose canister (MPC) shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell providing the redundant sealing.

Confinement System means the MPC that encloses and confines the spent nuclear fuel and associated nonfuel hardware during storage.

Consolidated Fuel means a fuel assembly that contains more than 264 fuel rods.

Controlled Area (or Owner-Controlled Area) means the area, outside the restricted area but inside the site boundary, for which access can be limited by PG&E.

Cooling Time is the time between discharging a spent fuel assembly and associated nonfuel hardware from the reactor (reactor shutdown) and the time the spent fuel assembly and associated nonfuel hardware are loaded into the MPC.

GLOSSARY

CTF means the cask transfer facility. The CTF is used to transfer an MPC from the transfer cask to an overpack, following transport from the FHB/AB and prior to moving the loaded overpack to the storage pad. The CTF can also be used to transfer an MPC from a loaded overpack to the transfer cask for transport back to the FHB/AB.

Damaged Fuel Assembly is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks; empty fuel rod locations that are not replaced with dummy fuel rods; or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Damaged Fuel Container (or DFC) means a specially designed enclosure for damaged fuel or fuel debris that permits gaseous and liquid media to escape while minimizing dispersal of gross particulates. The DFC features a lifting location that is suitable for remote handling of a loaded or unloaded DFC.

DBE means design-basis earthquake.

DCPP means Diablo Canyon Power Plant.

DCPP FSAR Update means the FSAR for DCPP that is maintained up-to-date in accordance with 10 CFR 51.71(e).

DCSS means dry cask storage system.

DDE means double design earthquake or deep dose equivalent.

DE means design earthquake.

Design Life is the minimum duration for which the component is engineered to perform its intended function as set forth in this SAR, if operated and maintained in accordance with this SAR.

Diablo Canyon ISFSI (or ISFSI) means the total Diablo Canyon storage system and includes the HI-STORM 100 System, transporter, CTF, storage pads, and ancillary equipment.

Diablo Canyon ISFSI Technical Specifications (or Diablo Canyon ISFSI TS or DC ISFSI TS) means the Technical Specifications issued as part of the license for PG&E to operate the Diablo Canyon ISFSI.

DOE means the US Department of Energy.

FHB/AB means the DCPP fuel handling building/auxiliary building.

GLOSSARY

Fracture Toughness is a property that is a measure of the ability of a material to limit crack propagation under a suddenly applied load.

FSAR means final safety analysis report.

Fuel Basket means a honeycombed structural weldment with square openings that can accept a fuel assembly of the type for which it is designed.

Fuel Debris refers to ruptured fuel rods, severed rods, loose fuel pellets, or fuel assemblies with known or suspected defects that cannot be handled by normal means.

HE means Hosgri earthquake.

High Burnup Fuel is a spent fuel assembly with an average burnup greater than 45,000 MWD/MTU.

HI-STORM 100 Overpack (or Loaded Overpack or Storage Cask) means the cask that receives and contains the sealed MPCs (containing spent nuclear fuel and nonfuel hardware) for final storage on the storage pads. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the MPC.

HI-STORM 100SA Overpack is a variant of the HI-STORM 100 overpack that is shorter, has a Version B top lid, and is outfitted with an extended baseplate and gussets to enable the overpack to be anchored to the storage pad. The HI-STORM 100SA overpack is designed for high-seismic applications and is used at the Diablo Canyon ISFSI.

HI-STORM 100 System consists of the Holtec International MPC, HI-STORM 100SA overpack, and HI-TRAC transfer cask. For Diablo Canyon, the HI-STORM 100SA overpack replaces the HI-STORM 100 overpack.

HI-TRAC 125 Transfer Cask (or HI-TRAC 125D Transfer Cask or HI-TRAC Transfer Cask or HI TRAC or Transfer Cask) means the cask used to house the MPC during MPC fuel loading, unloading, drying, sealing, and onsite transfer operations to an overpack. The HI-TRAC shields the loaded MPC allowing loading operations to be performed while limiting radiation exposure to personnel. The HI-TRAC is equipped with a pair of lifting trunnions to lift the HI-TRAC with a loaded MPC. HI-TRAC is an acronym for **H**oltec **I**nternational **Tra**nsfer **C**ask. The transfer cask used at the Diablo Canyon ISFSI is the HI-TRAC 125D design, which has been modified specifically for the Diablo Canyon ISFSI to allow vertical movement of the fuel throughout the loading, transport and storage processes.

Holtite is a trademarked Holtec International neutron shield material.

GLOSSARY

IFBA means integral fuel burnable absorber.

Important to Safety (ITS) means a function or condition required to store spent nuclear fuel safely; to prevent damage to spent nuclear fuel during handling and storage; and to provide reasonable assurance that spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public. This definition is used to classify structures, systems, and components of the ISFSI as important to safety (ITS) or not important to safety (NITS).

Independent Spent Fuel Storage Installation (ISFSI) means a facility designed, constructed, and licensed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage in accordance with 10 CFR 72. For Diablo Canyon, this term is clarified to mean the total storage system and includes the HI-STORM 100 System, transporter, CTF, storage pads, and ancillary equipment.

Insolation means incident solar radiation.

Intact Fuel Assembly means a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. Partial fuel assemblies, that is fuel assemblies from which fuel rods are missing, shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).

ISFSI Site means the ISFSI storage site and CTF.

ISFSI Storage Site (or Storage Site) means the area contained within the nuisance fence that circumscribes the ISFSI security fence and storage pads.

LAR means license amendment request.

LCO means limiting condition for operation.

LDE means lens dose equivalent.

License Life means the duration that the HI-STORM 100 System and the Diablo Canyon ISFSI are authorized by virtue of certification by the US NRC.

Low or Moderate Burnup Fuel is a spent fuel assembly with an average burnup less than or equal to 45,000 MWD/MTU.

Lowest Service Temperature (LST) is the minimum metal temperature of a part for the specified service condition.

GLOSSARY

LPT means the low profile transporter used to move the HI-TRAC transfer cask in a vertical configuration from the FHB/AB through the access door to the cask transporter.

LPZ means low population zone.

LTSP means long-term seismic program.

Maximum Reactivity means the highest possible k-effective including bias, uncertainties, and calculational statistics evaluated for the worst-case combination of fuel basket manufacturing tolerances.

MCNP means Monte Carlo N-Particle transport computer code.

METAMIC® is a trade name for an aluminum/boron carbide composite neutron absorber material qualified for use in the MPCs.

MPC-24 means the Holtec MPC designed to store up to 24 intact PWR fuel assemblies and associated nonfuel hardware. The originally-licensed MPC-24s will require modifications and analyses similar to the MPC-32 prior to their use.

MPC-24E means the Holtec MPC designed to store up to 24 PWR fuel assemblies and associated nonfuel hardware, 4 of which can be DFCs containing damaged fuel assemblies in designated fuel basket locations, and the balance being intact fuel assemblies. The originally-licensed MPC-24Es will require modifications and analyses similar to the MPC-32 prior to their use.

MPC-24EF means the Holtec MPC designed to store up to 24 PWR fuel assemblies and associated nonfuel hardware, 4 of which can be DFCs containing damaged fuel assemblies or fuel debris in designated fuel basket locations, and the balance being intact fuel assemblies. The originally-licensed MPC-24EFs will require modifications and analyses similar to the MPC-32 prior to their use.

MPC-32 means the Holtec MPC designed to store up to 32 intact PWR fuel assemblies and associated nonfuel hardware.

MSL means mean sea level.

MTU means metric tons of uranium.

Multi-Purpose Canister (MPC) means the sealed canister that consists of a honeycombed fuel basket contained in a cylindrical canister shell that is welded to a baseplate, lid with welded port cover plates, and closure ring. The MPC is the confinement boundary for storage conditions.

MWD/MTU means megawatt–days per metric ton of uranium.

GLOSSARY

Neutron Shielding means a material used to thermalize and capture neutrons emanating from the radioactive spent nuclear fuel.

NFPA means National Fire Protection Association.

Nonfuel Hardware is defined as burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), and other similarly designed devices with different names.

NRC means the US Nuclear Regulatory Commission.

NSOC means the DCPP Nuclear Safety Oversight Committee.

Nuisance Fence means the fence located outside the security fence, and is primarily intended to deter personnel from entering. This fence is capable of being utilized as a restricted area fence.

NWPA means the Nuclear Waste Policy Act of 1982 and any amendments thereto.

OBE means operating basis earthquake.

PMF means probable maximum flood.

Post-Core Decay Time (PCDT) is synonymous with cooling time.

Protected Area (or ISFSI Protected Area) means the area within the security fence that circumscribes the storage pads.

Protected Area Boundary means the security fence that circumscribes the storage pads.

PSRC means the DCPP Plant Staff Review Committee.

PWR means pressurized water reactor.

RCCA means rod cluster control assembly.

Reactivity is used synonymously with effective neutron multiplication factor or k-effective.

Regionalized Fuel Loading is a term used to describe an optional fuel loading strategy used in lieu of uniform fuel loading. Regionalized fuel loading allows high-heat-emitting fuel assemblies to be stored in fuel storage locations in the center of the fuel basket provided lower-heat-emitting fuel assemblies are stored in the peripheral fuel storage

GLOSSARY

locations. When choosing regionalized fuel loading, other restrictions in the Diablo Canyon ISFSI Technical Specifications must be considered also, such as those for nonfuel hardware and damaged fuel containers.

Restricted Area means the Radiological Controls Area (RCA) within the fence circumscribing the storage pads, access to which is limited by PG&E for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials.

Restricted Area Fence means the fence posted with RCA signage that circumscribes the storage pads. It is located to ensure the dose rate at this boundary will be less than 2 mrem/hr in compliance with 10 CFR 20 requirements for a restricted area boundary. This fence may be the same as the security fence.

SAT means systematic approach to training.

Security Fence is the first fence circumscribing the storage pads.

SDE means shallow dose equivalent.

Service Life means the duration for which the component is reasonably expected to perform its intended function, if operated and maintained in accordance with the provisions of the CoC. Service life may be much longer than the design life because of the conservatism inherent in the codes, standards, and procedures used to design, fabricate, operate, and maintain the component.

SFP means spent fuel pool.

Single Failure Proof Handling System means that the handling system is designed so that all directly-loaded tension and compression members are engineered to satisfy the enhanced safety criteria of paragraphs 5.1.6(1)(a) and (b) of NUREG-0612.

SNF means spent nuclear fuel.

SR means surveillance requirement.

SSC means structures, systems, and components.

SSE means safe shutdown earthquake.

STP means standard temperature and pressure conditions.

TEDE means total effective dose equivalent.

GLOSSARY

Thermosiphon is the term used to describe the buoyancy-driven natural convection circulation of helium within the MPC.

TLD means thermoluminescent dosimeter.

TODE means total organ dose equivalent.

TPD means thimble plug device.

Transport Route means the route used by the transporter for onsite movement of the loaded transfer cask from the FHB/AB to the CTF and from the CTF to the ISFSI pad.

Uniform Fuel Loading is a fuel loading strategy where any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions in the Diablo Canyon ISFSI Technical Specifications, such as those restrictions applicable to nonfuel hardware and damaged fuel containers.

USGS means the US Geological Survey.

UTM means Universal Transverse Mecator and is used to define topographic locations in metric coordinates.

Westinghouse LOPAR fuel assemblies have been used at DCPP and are one of the types of spent fuel assemblies that will be stored at the ISFSI.

Westinghouse VANTAGE 5 fuel assemblies have been used at DCPP and are one of the types of spent fuel assemblies that will be stored at the ISFSI.

χ/Q means site-specific atmospheric dispersion factors used in radiological dose calculations for routine and accidental releases.

ZPA means zero period acceleration.

Zr means fuel cladding material with the trade names Zircaloy-2, Zircaloy-4, or ZIRLO, unless otherwise specified. Any discussion of Zircaloy fuel cladding material in this SAR applies to any of these variants of zirconium-based fuel cladding material for low burnup fuel. High burnup fuel is limited to Zircaloy-2 or Zircaloy-4 cladding material.

CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION

CONTENTS

CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION

TABLES

CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION

Pursuant to 10 CFR 72, the Nuclear Regulatory Commission (NRC) issued Materials License SNM-2511 to PG&E on March 22, 2004, authorizing PG&E to build and operate the Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI). The license was issued for a period of 20 years in accordance with 10 CFR 72.42. This Final Safety Analysis Report (FSAR) Update is issued by PG&E and will be updated periodically in accordance with the provisions of 10 CFR 72.70.

This FSAR chapter explains the need for the Diablo Canyon ISFSI, and provides general descriptions of the co-located Diablo Canyon Power Plant (DCPP) and the ISFSI. Also, agents and contractors are identified, as well as material incorporated by reference. Some of the information pertaining to DCPP and the ISFSI site was taken from Chapters 1 and 2 of the DCPP FSAR Update (Reference 1 in Section 1.5 of this FSAR). Information pertaining to the Diablo Canyon ISFSI and its dry cask storage system was taken from the storage system vendor documents cited in FSAR Section 1.5.

In February 2010, the NRC approved a license amendment (LA) 1 to allow the use of Metamic as an alternative neutron absorber in the Multi-Purpose Canister (MPC); revise the MPC boron verification requirements by reducing the time of sampling prior to loading or unloading an MPC, allowing a boron concentration based on the maximum initial fuel assembly enrichment, and limiting the need to sample for boron concentration when out of the Spent Fuel Pool; eliminate the time limit for a loaded MPC in the CTF; revise the helium leak testing requirements for the MPC. Additionally, some clarifications were made, and Administrative requirements for combustible gas monitoring during welding and cutting, cladding oxidizing atmosphere control, and boron dilution control, were added.

In January 2012, the NRC approved license amendment LA 2 to allow the loading of high-burnup fuel (>45,000 MWD/MTU) in the MPC-32, add allowance for loading of neutron source assemblies (NSAs) and instrument tube tie rods (ITTRs) as non-fuel hardware, add allowance for loading of fuel with different names provided the critical characteristics are met, eliminate the restriction on loading high-burnup Zirlo clad fuel, delete the option for vacuum drying of fuel, specify the reference temperature for the helium backfill pressure range, identify in the Technical Specification that the HI-STORM can be considered operable with up to 50% vent blockage, and add requirements for use of a Supplemental Cooling System when loading high-burnup fuel. Additionally, changes to Administrative requirements were made to support the changes.

In February 2014, the NRC approved license amendment LA 3 to allow the increase in the allowable heat load to 28.74 kW for high burnup fuel in the MPC-32, clarified how to calculate heat load for regionalized loading of high burn-up fuel (HBF), revised the helium backfill range for certain MPCs, provided clarification that the use of the supplemental cooling system is only applicable to SNF previously transferred, and revised the DC ISFSI maximum average storage and maximum transfer temperatures to reflect more accurate site data.

In January 2016, the NRC approved license amendment LA 4 to allow changes in the security force weapons, pursuant to NRC's Order EA-13-092.

In April 2016, the NRC approved license amendment LA 5 to amend the Technical Specifications (TS) of SNM License No. SNM-2511, to remove preferential loading references from the TS and improve the readability and human factors usage of the TS.

1.1 INTRODUCTION

DCPP consists of two nuclear generation units located on the California coast approximately 6 miles northwest of Avila Beach, California. The two units are essentially identical pressurized water reactors (PWRs), each rated at a nominal 1,100 megawatts-electric (MWe). The two units share a fuel handling building/auxiliary building (FHB/AB) as well as certain components of auxiliary systems. The reactors, including their nuclear steam supply systems, were furnished by Westinghouse Electric Corporation. Each reactor has a dedicated fuel handling system and spent fuel pool (SFP). Both SFPs share a single 125-ton capacity crane for fuel handling activities. Both units and the plant site are owned and operated by PG&E.

Unit 1 began commercial operation in May 1985 and Unit 2 in March 1986. The operating licenses expire in November 2, 2024 for Unit 1 and August 26, 2025 for Unit 2. In general, the operating and spent fuel storage histories of DCPP Unit 1 and Unit 2 are similar to those of other PWRs. The spent fuel storage racks were initially of low-density design, capable of accommodating only one and one-third cores of spent fuel assemblies. These low-density racks were replaced in the late 1980s with high-density racks that are currently in use.

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 fuel assemblies are permanently discharged from each unit after a refueling. The SFP for each unit presently has sufficient capacity for the storage of 1,324 fuel assemblies, excluding the temporary cask pit racks.

The Diablo Canyon ISFSI consists of the storage pads, a cask transfer facility (CTF), an onsite cask transporter, and the dry cask storage system. The dry cask storage system that has been selected by PG&E for the Diablo Canyon ISFSI is the Holtec International (Holtec) HI-STORM 100 System. The HI-STORM 100 System is comprised of a multi-purpose canister (MPC), the HI-STORM 100SA storage overpack, and the HI-TRAC transfer cask. The HI-STORM 100 System is certified by the Nuclear Regulatory Commission (NRC) for use by general licensees as well as site-specific licensees (see NRC 10 CFR 72 Certificate of Compliance [CoC] No. 1014, Amendment 1) (Reference 2, Section 1.5).

The Holtec CoC No. 1014, Amendment 1 (Reference 2), includes a HI-STORM 100SA storage overpack, an MPC-32 design (for storage of 32 PWR spent fuel assemblies), and additional 24 PWR assembly capacity MPC designs with different fuel storage (for example, high burnup fuel and certain damaged fuel). As discussed below, Holtec CoC No. 1014, Amendment 1, supports the Diablo Canyon ISFSI. PG&E understands, however, that some of the features in Holtec CoC No. 1014, Amendment 1, are not currently applicable to the Diablo Canyon ISFSI.

Later Holtec CoC No. 1014 Amendments have been used to support changes made in amendments to the Diablo Canyon ISFSI License. Specifically, CoC Amendment 2 was used to support the use of a variable dissolved boron concentration in the MPC based on maximum fuel assembly initial enrichment loaded (LA 1). CoC Amendment 3 was used as the basis for the selection criteria for high-burnup fuel (LA 2), allow loading of NSAs as non-fuel hardware (LA 2), change the MPC leakage criteria from measuring leak rate from the lid-to-shell weld and the port cover plates to only verifying the port cover plates to leak-tight criteria (LA 1). CoC Amendment 5 was used to support the thermal evaluation methodology, 3-D Computational Fluid Dynamics (CFD) model, change (LA 2), allow for the HI-STORM to be considered OPERABLE with up to 50% vent blockage, and decoupling of the 100% rod rupture and 100% vent blockage accidents (LA 2). CoC Amendment 6 was used to support the addition of ITTRs as non-fuel hardware (LA 2).

Revision 2 of this FSAR incorporates site-specific modifications, which pertain only to the MPC-32 and related components. These modifications, which will facilitate the fuel-loading campaigns, include: (1) use of a single-failure proof fuel handling building crane; (2) shortening the MPC-32 and transfer cask to allow vertical handling of the transfer cask throughout each load campaign; (3) use of a low profile transporter to transport a loaded transfer cask from the FHB/AB to the cask transporter; (4) elimination of the CTF lifting platform; (5) use of a single-failure proof transporter for heavy load handling outside the FHB/AB; and (6) modifying the overpack lid. The MPC-32 can store up to 32 intact fuel assemblies that meet the approved content requirements of the Diablo Canyon ISFSI Technical Specifications (TS). The MPC-24, MPC-24E, and MPC-24EF were originally licensed to store up to 24 fuel assemblies that meet the approved content requirements of the Diablo Canyon ISFSI TS, including limited storage of damaged fuel assemblies and fuel debris. The originally-licensed MPC-24s will require modifications and analyses similar to the MPC-32 prior to their use.

The Diablo Canyon ISFSI is designed to hold up to 140 storage casks (138 casks plus 2 spare locations). The physical characteristics of the spent fuel assemblies to be stored are described in Section 3.1. Based on the current fuel strategy and the principal use of the MPC-32, the ISFSI with a storage pad capacity of 140 casks will be capable of storing the spent fuel generated by DCPP Units 1 and 2 through 2024 and 2025, respectively.

The Diablo Canyon ISFSI incorporates these designs in a preferred cask system licensing approach as follows:

(1) The Diablo Canyon ISFSI site-specific license incorporates the MPC capabilities as specified in the CoC No. 1014, Amendment 1. The NRC issued a Safety Evaluation Report (SER) in July 2002. While the MPC capabilities covered by the Holtec CoC No. 1014, Amendment 1, does not completely envelope all of the spent fuel characteristics eventually needed for DCPP fuel, it covers most of the current SFP inventory and will permit the storage of nearly all spent fuel and associated nonfuel hardware generated during the license term.

(2) MPC designs that may be needed for the balance of the DCPP spent fuel characteristics will be addressed in future revisions to the CoC. As these changes are submitted by Holtec and approved by the NRC, PG&E will amend the Diablo Canyon ISFSI site-specific license to incorporate these changes. The resulting capability will provide PG&E with the flexibility to store onsite all the spent fuel and nonfuel hardware from DCPP Units 1 and 2 generated during the term of its operating licenses.

(3) In a Federal Register Notice dated October 11, 2001 (66 FR 51823), the NRC issued the final rule change regarding greater than class C (GTCC) waste (for example, split pins and thimble tubes). The rule change applies only to the interim storage of GTCC waste generated or used by commercial nuclear power plants. The rule change allows interim storage of reactor-related GTCC wastes under a 10 CFR 72 site-specific license. In accordance with the guidance contained in Interim Staff Guidance Document 17 (ISG 17), PG&E plans to request a modification to its proposed site-specific license at a future date to allow interim storage of GTCC wastes at the Diablo Canyon ISFSI.

(4) Exemptions pertaining to the Diablo Canyon ISFSI license are identified in Table 1.1-1.

Licensing of the Diablo Canyon ISFSI also involved NRC review of a number of site-specific issues. They included the site-specific environmental review, geotechnical issues related to the site and natural phenomena, and other site-specific matters.

Although the Holtec CoC No. 1014, Amendment 1 includes a high-seismic capability for the storage overpack (the HI-STORM 100SA), it did not incorporate some Diablo Canyon specific information (for example, the pad design, the overpack seismic anchorage design, the cask transporter seismic design, and the CTF design). PG&E submitted information on these items as part of its site-specific application and these issues were reviewed and licensed as part of the PG&E site-specific 10 CFR 72 license.

This FSAR refers to a number of dry storage and ancillary components licensed under the HI-STORM 100 System CoC, Amendment 1 and Holtec FSAR, Revision 1A (Section 1.5). Some of these components were modified by Holtec International under the provisions of 10 CFR 72.48. Wherever necessary, these changes are discussed in the text, tables, and figures in this FSAR.

In accordance with 10 CFR 72.42, the Diablo Canyon ISFSI license was issued for a term of 20 years. If near the end of the initial license, permanent or interim DOE High Level Waste (HLW) facilities are unavailable for acceptance of commercial nuclear spent fuel, PG&E expects to submit an application for ISFSI license renewal pursuant to 10 CFR 72.42(b).

The Diablo Canyon ISFSI is designed to protect the stored fuel and prevent release of radioactive material under all normal, off-normal, and accident conditions of storage in accordance with all applicable regulatory requirements contained in 10 CFR 72. This FSAR was prepared in compliance with the requirements of 10 CFR 72 and using the guidance contained in Regulatory Guide 3.62, “Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks,” (February 1989); and NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities,” (March 2000).

Additionally, the NRC has issued license amendments allowing PG&E to take credit for soluble boron in the spent fuel pools (Reference 11) and permit cask handling activities in the DCPP fuel handling building/auxiliary building (Reference 12). Also, PG&E applied for and received an exemption from the criticality requirements of 10 CFR 68(b)(1) during loading, unloading, and handling of the MPC in the DCPP SFP (References 13 and 14, respectively).

1.2 GENERAL DESCRIPTION OF LOCATION

The DCPP site consists of approximately 750 acres of land located in San Luis Obispo County, California, adjacent to the Pacific Ocean and roughly equidistant from San Francisco and Los Angeles. The site is located directly southeast of Montana de Oro State park, which is located along the coast of California in San Luis Obispo County. This site area is approximately 12 miles west-southwest of the city of San Luis Obispo, the county seat and nearest significant population center.

The nearest residential community is Los Osos, approximately 8 miles north of the plant site. The township of Avila Beach is located along the coast at a distance of approximately 6 miles southeast of the plant site. The city of Morro Bay is located along the coast approximately 10 miles northwest of the plant site. A number of other cities, as well as some unincorporated residential areas, exist along the coast and inland. However, these are at distances greater than 8 miles from the plant site. Only a few individuals reside within 5 miles of the plant site.

Access to the plant site is controlled by security fencing that defines the plant-protected area within the owner-controlled area, which is surrounded by a farm-type fence. The plant site is located near the mouth of Diablo Creek, and a portion of the site is bounded by the Pacific Ocean. All coastal properties located north of Diablo Creek, extending north to the southerly boundary of Montana de Oro State Park and reaching inland approximately 0.5 miles are owned by PG&E. Coastal properties located south of Diablo Creek and reaching inland approximately 0.5 miles are owned by Eureka Energy Company, a wholly owned subsidiary of PG&E. Except for the DCPP site, the 4,500 acres of this area are encumbered by two grazing licenses.

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 plant operation within the owner-controlled area. The plant site is not traversed by public highway or railroad. Normal access to the site is from the south by private road, which is fenced and posted by PG&E. The offshore area is not under PG&E control and is at times entered by commercial or sports fishing boats.

The plant site occupies a coastal terrace that ranges in elevation from 60 to 150 ft above mean sea level (MSL) and is approximately 1,000 ft wide. Plant grade, determined at the turbine building main floor, is at elevation 85 ft above MSL. The seaward edge of the terrace is a near-vertical cliff. Back from the terrace and extending for several miles inland are the rugged Irish hills, an area of steep, brush-covered hillsides and deep canyons that are part of the San Luis Mountains, which attain an elevation of 1,500 ft within about a mile of the site.

The reactors and ancillary structures are situated on top of bedrock. The coastal areas surrounding the plant are well drained, primarily via Diablo Creek, and groundwater is at a depth of at least 170 ft below the surface of the ISFSI pad. The climate of the site area is typical of that along the central California coast. The winter comprises the rainy season, with more than 80 percent of the average annual rainfall of approximately 16 inches. The average annual temperature of the site area is about 55°F, with a variation between approximately 32°F minimum and 97°F maximum, which reflects the strong marine influence.

The ISFSI is located within the PG&E owner-controlled area at DCPP. Figure 2.1-1 shows the location of the plant and ISFSI sites on a map of western San Luis Obispo County. Figure 2.1-2 shows a plan drawing of the ISFSI site. There are no important to safety structures, systems or components that are shared between the ISFSI and DCPP. A more detailed description of the ISFSI site is provided in FSAR Section 2.1.

1.3 GENERAL STORAGE SYSTEM DESCRIPTION

The Diablo Canyon ISFSI includes the following major structures, systems, and components (SSCs): the storage pads, CTF, onsite transporter, and dry cask storage system. The dry cask storage system selected by PG&E is the Holtec International HI-STORM 100 System, which has been certified by the NRC for use by general licensees as well as site-specific licensees. The HI-STORM 100 System is comprised of the MPC, the HI-STORM 100 storage overpack, and the HI-TRAC transfer cask; the design and operation of these components are described in detail in the HI-STORM 100 System FSAR. A general description of major SSCs is provided herein. More detailed descriptions of the HI-STORM 100 System are contained in Section 4.2 of this FSAR and in the Holtec International documents cited in References 2 and 4, Section 1.5 of this FSAR. Likewise, more details on the storage pads, CTF, and transporter are provided in Sections 4.2 through 4.4 of this FSAR.

The MPC provides the confinement boundary for the spent fuel and associated nonfuel hardware. It is an integrally-welded pressure vessel that holds up to 24 or 32 DCPP spent fuel assemblies and meets the stress limits of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. The MPCs are welded cylindrical structures consisting of a honeycomb fuel basket, a baseplate, canister shell, a lid, and a closure ring. The honeycomb fuel basket uses geometric spacing and Boral or Metamic neutron absorbers for criticality control. The MPC is made entirely of stainless steel, except for the neutron absorbers, and an aluminum seal washer or port plug with thread protector in both the vent and drain ports assemblies. An alternative vent and drain port plug configuration may be used, which does not contain aluminum washers.

A loaded MPC is stored within the HI-STORM 100SA overpack in an anchored vertical orientation. The overpack provides gamma and neutron shielding, ventilation passages, and protects the MPC from missiles and natural phenomena. It is a rugged, heavy-walled cylindrical container. The main structural function of the storage overpack is provided by carbon steel, and the main shielding function is provided by unreinforced concrete. The overpack concrete is enclosed by cylindrical steel shells, a thick steel baseplate, and a top plate. The overpack lid is designed as a steel-encased concrete disc to provide neutron and gamma attenuation in the vertical direction. Inlets at the bottom and corresponding outlets at the top of the overpack allow air to circulate naturally to cool the MPC. The inner shell of the overpack has guides attached to its inner diameter to guide the MPC during insertion/removal and to allow cooling airflow to circulate through the overpack.

The transfer cask provides an internal, cylindrical cavity of sufficient size to house an MPC during loading, unloading, and movement of the MPC from the SFP to the overpack. It provides gamma and neutron shielding and protects the MPC from missiles and natural phenomena. The structural function of the transfer cask is provided by the carbon steel shell, top lid, and bottom lid. Neutron and gamma shielding are provided by water and lead, respectively. Figure 4.2-8 shows the transfer cask. The MPC access hole through the transfer cask top lid allows the lowering/raising of the MPC between the transfer cask and the overpack. The bottom lid is bolted to the bottom flange of the transfer cask and is used during MPC fuel loading, sealing operations, and transport. In addition to providing shielding in the axial direction, the bottom lid incorporates a seal that is designed to hold demineralized water in the transfer cask inner cavity, thereby preventing contamination of the exterior of the MPC by contaminated SFP water.

A transporter is used to move the transfer cask/MPC assembly from outside the FHB/AB to the CTF, which is adjacent to the ISFSI storage pads. The transporter will transfer the MPC to the overpack at the CTF, and then move the loaded overpack to the storage pads. The transporter is a U-shaped tracked vehicle consisting of the vehicle main frame, hydraulic lifting towers, an overhead beam system that connects between the lifting towers, a cask restraint system, the drive and control systems, and a series of cask lifting attachments. The transporter design permits the transfer cask/MPC assembly and the loaded overpack to only be handled vertically.

As shown in Figure 4.1-1, the CTF is located about 100 ft from the storage pads. The CTF is designed to contain an overpack below grade to facilitate the transfer of a loaded MPC from the HI-TRAC transfer cask to the overpack. Figure 4.4-3 shows the CTF.

The loaded overpacks are stored on a series of concrete storage pads within a protected area separate from that of DCPP. Each storage pad is designed to accommodate up to 20 loaded overpacks in a 4-by-5 array as shown in Figure 4.1-1. Ultimately, seven such pads may be built. Each loaded overpack is approximately 11 ft in diameter, 20 ft high, and weighs about 360,000 pounds. There is approximately 6 ft, surface-to-surface distance between the overpacks. The series of 7 storage pads will cover an area approximately 500 ft by 105 ft. The protected area has applicable barrier, access, and surveillance controls meeting 10 CFR 73.55 for an ISFSI co-located with a nuclear power plant.

The preparation and loading of the MPCs take place in the FHB/AB. These activities, plus the full summary of activities culminating in the storage of MPCs on the storage pads, are described in Sections 4.4 and 5.1.

The important-to-safety SSCs of the ISFSI are identified in Section 4.5.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

PG&E is performing the engineering, site preparation, and construction of the ISFSI storage pads and CTF, using specialty contractors as necessary.

The spent fuel storage system, provided by Holtec International, consists of the HI-STORM 100SA overpack, the MPC, and the HI-TRAC transfer cask; the transporter; and design criteria for the ISFSI storage pads, and CTF.

PG&E is responsible for the operation of the ISFSI.

All of these activities involving important-to-safety structures, systems, and components are subject to NRC-approved QA programs as discussed in Chapter 11 and in the Holtec references cited in Section 1.5.

1.5 MATERIAL INCORPORATED BY REFERENCE

1. Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update.

2. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 1, July 15, 2002.

3. Deleted in Revision 2.

4. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 1A, January 2003.

5. Submittal of Holtec Proprietary Design Drawing Packages, PG&E Letter to the NRC DIL-01-008, dated December 21, 2001

6. Diablo Canyon Power Plant Units 1 & 2, Emergency Plan.

7. Diablo Canyon ISFSI Technical Specifications.

8. Diablo Canyon ISFSI Training Program.

9. Deleted in Revision 0.

10. Diablo Canyon ISFSI Preliminary Decommissioning Plan.

11. License Amendment 154, Credit for Soluble Boron in the Spent Fuel Pool Criticality Analysis, issued by the NRC, September 15, 2002.

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

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

14. Exemption From the Requirements of 10 CFR 50.68(b)(1), Issued by the NRC, January 30, 2004.

15. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 2, June 7, 2005

16. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 3, May 29, 2007.

17. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 5, July 14,2008.

18. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 6, August 17, 2009.

TABLE 1.1-1

DIABLO CANYON ISFSI LICENSE
EXEMPTIONS

Code of Federal Regulations Reference	Exemption
10 CFR 72.72(d)	As specified in License Condition 16 of the Diablo Canyon ISFSI License SNM-2511, the NRC has granted PG&E an exemption from the provisions of 10 CFR 72.72(d) with respect to maintaining a duplicate set of spent fuel storage records. PG&E may maintain records of spent fuel and high level radioactive waste in storage either in duplicate, as required by 10 CFR 72.72(d), or, alternatively, a single set of records may be maintained at a records storage facility that satisfies the standards of ANSI N45.2.9-1974. All other requirements of 10 CFR 72.72(d) must be met.
10 CFR 50.68(b)(1)	10 CFR 50.68(b)(1) prohibits the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water. Specifically, the regulation ensures a subcritical condition will be maintained without credit for soluble boron. For an MPC loaded with fuel having the highest permissible reactivity, soluble boron credit is necessary to ensure the MPC remains subcritical in the SFP. Therefore, PG&E requested an exemption from 10 CFR 50.68(b)(1) to allow MPC loading, unloading, and handling operations without meeting the requirement of being subcritical under the most adverse moderation conditions feasible by unborated water. In the exemption request (Reference 13, Section 1.5 of this FSAR), PG&E evaluated the possibility of an inadvertent criticality during MPC loading, unloading, and handling in the DCPP SFP. Based on the alarms, procedures, administrative controls, assumption of zero burnup fuel, and availability of trained operators described in Reference 13, the NRC granted an exemption from the criticality requirements of 10 CFR 50.68(b)(1) during loading, unloading, and handling of the MPC in the DCPP SFP (Reference 14 in Section 1.5 of this FSAR).

CHAPTER 2

SITE CHARACTERISTICS

CONTENTS

CHAPTER 2

SITE CHARACTERISTICS

CONTENTS (Continued)

CHAPTER 2

SITE CHARACTERISTICS

TABLES

CHAPTER 2

SITE CHARACTERISTICS

FIGURES

CHAPTER 2

SITE CHARACTERISTICS

FIGURES (Continued)

CHAPTER 2

SITE CHARACTERISTICS

FIGURES (Continued)

SITE CHARACTERISTICS

FIGURES (Continued)

CHAPTER 2

SITE CHARACTERISTICS

This chapter provides information on the location of the Diablo Canyon ISFSI and descriptions of the geographical, demographical, meteorological, hydrological, seismological, and geological characteristics of the storage site and the surrounding vicinity. Some of the provided information is taken from Chapters 1 and 2 of the DCPP FSAR Update.

2.1 GEOGRAPHY AND DEMOGRAPHY OF SITE SELECTED

A description of the geography and demography of the Diablo Canyon area is contained in the DCPP FSAR Update. This information generally applies to the Diablo Canyon ISFSI, as described below.

2.1.1 SITE LOCATION

The general area surrounding the Diablo Canyon Power Plant and the ISFSI is shown in Figure 2.1-1. The ISFSI will be located within the PG&E owner-controlled area at Diablo Canyon, which consists of approximately 750 acres of land located in San Luis Obispo County, California, adjacent to the Pacific Ocean and roughly equidistant from San Francisco and Los Angeles. The boundary of this area is used for the analyses required in accordance with 10 CFR 72.104 and 72.106. This area is located along the coast of California in San Luis Obispo County directly southeast of Montana de Oro State Park and is approximately 12 miles west-southwest of the city of San Luis Obispo, the county seat and the nearest significant population center.

The nearest residential community is Los Osos, approximately 8 miles north of the ISFSI site. This community is located in a mountainous area adjacent to Montana de Oro State Park. The township of Avila Beach is located down the coast approximately 6 miles southeast of the ISFSI site. The city of Morro Bay is located up the coast approximately 10 miles northwest of the site. A number of other cities, as well as some unincorporated residential areas, exist along the coast and inland. However, these communities are greater than 8 miles from the ISFSI site. Only a few individuals reside within 5 miles of the site.

The DCPP facilities and the ISFSI site are located near the mouth of Diablo Creek, and a portion of the power plant site is bounded by the Pacific Ocean. Approximately 165 acres of the owner-controlled area are located north of Diablo Creek. The remaining 595 acres are located adjacent to and south of Diablo Creek. The entire acreage is owned by PG&E.

The ISFSI is located at latitude 35°12'52" North and longitude 120°51'00" West. The Universal Transverse Mercator (UTM) coordinates of the ISFSI are 695,689 meters East and 3,898,723 meters North. Figure 2.1-1 shows the location of the Diablo

Canyon plant and ISFSI sites, on a map of western San Luis Obispo County. Figure 2.1-2 shows a plan drawing of the ISFSI site.

2.1.2 SITE DESCRIPTION

A security fence that defines the ISFSI protected area within the owner-controlled area, which is surrounded by a farm-type fence, controls access to the ISFSI site. PG&E owns all coastal properties north of Diablo Creek, to the southerly boundary of Montana de Oro State Park and inland a distance of 0.5 to 1.75 miles. Similarly, PG&E owns all coastal properties south of Diablo Creek for approximately 8 miles and inland approximately 1.75 miles. Except for the DCPP and ISFSI sites, all of the acreage north and south of DCPP and the ISFSI are encumbered by two grazing leases. In accordance with an agreement in principle reached in 2000 with the Central Coast Regional Water Quality Control Board, land north of DCPP, consisting of 2,013 acres of watersheds draining to approximately 5.7 miles of coastline, will be preserved by a conservation easement for ecological purposes. The primary goal is protection of marine resources from Fields Cove to Coon Creek through watershed and habitat protection of all the lands draining to that coastline. In addition, PG&E will protect 547 acres draining to Coon Creek through Best Management Practices for as long as PG&E operates the plant or holds the property, whichever is longer.

The Diablo Canyon owner-controlled area occupies a coastal terrace and adjacent uplands that range in elevation from 60 to 1,400 ft above mean sea level (MSL). The DCPP facilities, other than the intake and discharge structures, occupy an area between 60 and 150 ft MSL and approximately 1,000 ft wide. The ISFSI is located approximately 0.22 miles northeast of the Unit 1 containment (ISFSI/containment center-to-center) at an elevation of approximately 310 ft MSL (Figure 2.1-2). The seaward edge of the terrace is a near-vertical cliff. Back from the terrace and extending for several miles inland are the rugged Irish hills, an area of steep, brush-covered hillsides and deep canyons that are part of the San Luis Mountains. The coastal areas surrounding the ISFSI are well drained, primarily via Diablo Creek, and the water table is typically low.

The ISFSI is located between hillsides and is situated directly on bedrock at the site area. The topography of the site and the limited rainfall preclude any possibility of flooding. Even in the event of a probable maximum flood (PMF) and hypothetical plugging of the 10 ft diameter drainage pipe located below the two nearby switchyards, no flooding of the ISFSI is expected to occur since the roadway located north of Diablo Creek will serve as a bypass for flood waters. Water levels will flow below the elevation of the ISFSI.

The climate of the site area is typical of that along the central California coast and reflects a strong maritime influence. The rainy season is in the winter, producing more than 80 percent of the average annual rainfall of approximately 16 inches. The average annual temperature of the site area is about 55°F.

PG&E has full authority to control all activities within the ISFSI site and owner-controlled area boundaries; this authority extends to the mean high water line along the ocean coastline. The mineral rights within the 165-acre PG&E portion of the site are owned by PG&E; there is no information suggesting that the land contains commercially valuable minerals. On land, there are no activities unrelated to the ISFSI or power plant operation within the site exclusion area. The owner-controlled area is not traversed by public highway or railroad. Normal access to the ISFSI and DCPP sites is from the south by a 6.5-mile long private road, which is fenced and posted by PG&E. The private road is connected to a local public roadway, Avila Beach Drive, which runs along the shoreline of San Luis Obispo Bay. A US Coast Guard station is located adjacent to the security gate. The major access to the area is via US Highway 101, which passes about 9 miles east of the ISFSI site and is accessible at approximately 15 miles to the southeast of the site.

2.1.3 POPULATION DISTRIBUTION AND TRENDS

The population distribution and projections for areas around the ISFSI site are based on the 2000 census and on estimates prepared by the California Department of Finance. As described in Section 2.1.2, the ISFSI site is located approximately 0.22 miles northeast of the Unit 1 containment. The population data presented in this section for the ISFSI are based on distances from the Unit 1 containment rather than distances from the ISFSI site. The 0.22-mile offset to the ISFSI, however, is considered to have negligible effect on the population estimates at various distances and directions from the ISFSI.

The population data are provided for areas within a 50-mile radius of the ISFSI. Population distributions are provided for areas within specific radii and sectors, and include the 2000 census data as well as projections for the years 2010 and 2025.

The area within 50 miles of the ISFSI includes most of San Luis Obispo County, some portions of Santa Barbara County, and a small area of Monterey County. Approximately 55 percent of the area within the radius is on land, with the balance being the Pacific Ocean. In general, the portion of California that lies within 50 miles of the ISFSI is relatively sparsely populated, having approximately 424,000 residents in 2000.

The 2000 census population of this region is very close to that projected in the original FSAR for DCPP, and subsequent projections by the Department of Finance are similarly close to earlier projections. Table 2.1-1 shows population trends of the State of California and of San Luis Obispo and Santa Barbara Counties. Table 2.1-2 shows the growth since 1960 of the principal cities within 50 miles of the ISFSI site. Table 2.1-3 lists communities within 50 miles of the ISFSI site having a population of 1,000 or more, provides the distance and direction from the ISFSI site, and shows the 2000 population.

2.1.3.1 Population Within 10 Miles

In 1980, approximately 16,760 persons resided within 10 miles of the ISFSI site. The 1990 census counted approximately 22,200 residents within the same 10 miles. The 2000 census counted approximately 23,700 residents within the same 10 miles. As in 1980, the nearest residence is approximately 1.5 miles north-northwest of the ISFSI site and is occupied by two persons. There are 4 permanently inhabited dwellings, with approximately 14 residents, within 5 miles of the ISFSI. The population within a 6-mile radius, the low population zone (LPZ) as used in the emergency plan, is estimated to be 100.

Figure 2.1-3 shows the 2000 population within a 10-mile radius, wherein the area is divided into 22.5° sectors and part circles with radii of 1, 2, 3, 4, 5, and 10 miles. Figures 2.1-4 and 2.1-5 show projected population distributions for 2010 and 2025, respectively, and are based primarily on population projections published by the California Department of Finance. The distributions are based on the assumption that the land usage will not change in character during the next 25 years, and that population growth within 10 miles will be proportional to growth in San Luis Obispo County as a whole.

2.1.3.2 Population Between 10 and 50 miles

Figure 2.1-6 shows the 2000 population distribution between 10 and 50 miles within the sectors of 22.5°, with part circles of radii of 10, 20, 30, 40, and 50 miles. Figures 2.1-7 and 2.1-8 show projected distributions for 2010 and 2025, respectively, and are based primarily on population projections published by the Department of Finance and interviews with area government officials. In 2000, some 82 percent of those persons within 50 miles of the ISFSI site resided in the population centers listed in Table 2.1-3.

2.1.3.3 Transient Population

In addition to the resident population presented in the tables and population distribution charts, there is a seasonal influx of vacation and weekend visitors, especially during the summer months. This influx is heaviest south along the coast from Avila Beach to south of Oceano.

During August, the month of heaviest influx, the maximum overnight transient population in motels and state parks in this area is approximately 100,000 persons. However, there are no significant seasonal or diurnal shifts in population or population distribution within the LPZ. Table 2.1-4 lists transient population for recreation areas within 50 miles of the site for the periods of record listed.

Within the LPZ, the maximum-recorded number of persons at any single time is estimated to be 5,000. This figure is provided by the State Department of Parks and Recreation and corresponds to the maximum daytime use of Montana de Oro State Park. Overnight use is considerably less, with an estimated maximum of 400.

Evacuation of these numbers of persons from the park in the event of a radiation release could be accomplished as provided for in the emergency plan, with a reasonable probability that no injury would result.

2.1.3.4 Public Facilities and Institutions

Several elementary schools are located within 10 miles of the ISFSI site, near Los Osos and Avila Beach. These schools serve the local communities and do not draw from outlying areas. California Polytechnic State University is 12 miles north-northeast of the ISFSI site and has an enrollment of approximately 17,000. Cuesta College is located 10 miles northeast of the site and has an enrollment of approximately 10,000.

Montana de Oro State Park is located north of the site. Its area of principal use is along the beach, between 4 and 5 miles north-northwest of the site. The total number of visitor days during a 12-month period over the previous 5 years averages 600,000.

2.1.4 USE OF NEARBY LAND AND WATERS

The San Luis Range, reaching a height of 1,800 ft, dominates the region between the site and US Route 101. This upland country is used to a limited extent for grazing beef cattle and, to a very minor extent, dairy cattle. There are also wild and domestic goats, deer, and other wildlife in the vicinity of the plant site. The terrain east of US Route 101, lying in the mostly inaccessible Santa Lucia Mountains, is sparsely populated with little development. A large portion of this area is included within the Los Padres National Forest.

2.1.4.1 Agriculture

San Luis Obispo County has relatively little level land, except for a few small coastal valleys such as the Santa Maria and San Luis Valleys, and some land along the county’s northern border in the Salinas Valley and Carrizo Plain areas. Farming is a significant land use in the county. Principal crops include wine grapes, vegetables, nurseries, fruits, nuts, and grain. There are several vineyards and wineries located in the county. The county’s leading agricultural product is wine grapes, valued at approximately $74,000,000 in 1998. The total farm acreage in the county is approximately 1,200,000. The county contains a total of 2,128,640 acres.

2.1.4.2 Dairy

The only dairy activity is 12 miles northeast of the site at California Polytechnic State University, located in the city of San Luis Obispo, which produces 1,200 gallons of milk per day. Some replacement heifers and dry cows are sometimes pastured on property adjacent to the site.

2.1.4.3 Fisheries

The ISFSI site is located between two fishing harbors that support commercial and sport fishing activities. Port San Luis Harbor is located in Avila Beach, approximately 6 miles down coast of the ISFSI site. Morro Bay Harbor is located approximately 10 miles up coast of the site. In 1994 the combined sport catch totaled approximately 342,000 rockfish and 6,000 fish of other species, from a total of 16 fishing vessels.

Commercial landings are calculated by poundage of landings by port. In 1994, at Port San Luis and Morro Bay Harbors, the landings were estimated to be as follows: 2,474,000 pounds of rockfish, 5,405,000 pounds of other fish species, 1,300 pounds of abalone, 2,694,000 pounds of squid, 534,000 pounds of crab, 418,000 pounds of shrimp, and 4,400 pounds of urchins.

There has been a dramatic decrease since 1970 in the abalone catch, with approximately 621,000 pounds taken in 1966 and 200,000 pounds taken in 1970, due primarily to severe restrictions imposed by the California Department of Fish and Game. Some data suggest that the southern movement of the Southern California sea otter may have had an impact on the red abalone population.

2.1.4.4 Water use

There are two public water supply groundwater basins within 10 miles of the site. Avila Beach County Water and Sewer District and San Miguelito Mutual Water and Sewer Company provide water to the Avila Beach and Avila Valley area. The property owners to the north and south of the ISFSI site capture surface water from small intermittent streams and springs for minimal domestic use. PG&E's lessee captures water 2 to 4 miles south of the ISFSI site from streams and springs between Pecho Canyon and Rattlesnake Canyon. Property owned by PG&E captures water from Crowbar Canyon, 1 mile north of the plant site. In addition, an ocean water desalinization plant was built and has been in operation at DCPP since 1985.

2.1.4.5 Land Usage Within 5 Miles

The only agricultural activities indicated by county records are cattle grazing in much of the area surrounding the site, and a farm in the east-southeast sector, producing legumes and cereal grass such as grains. The farm is located along the site access road on the coastal plateau, starting approximately 2 miles from the plant and extending to 4.5 miles from the ISFSI. There is also a household garden greater than 500 square ft in the east sector. These activities are being conducted on land leased from PG&E.

2.1.4.6 Other Nearby Usage

The community of Avila Beach lies approximately 6 miles east-southeast of the plant. Port San Luis Harbor is located directly opposite the security entrance that controls entry into Diablo Canyon via the private access road. A small public beach is located next to the harbor area and is used frequently by the public for access to the harbor waters for recreation purposes.

A tanker-loading pier owned by UNOCAL Oil Company is located in Port San Luis Harbor directly adjacent to the small beach area. Prior to 1999, there were also several UNOCAL oil storage tanks located on the hills immediately southeast of Avila Beach. Approximately 1 to 2 local tankers per month offloaded oil for storage in these tanks until the late 1990s. The tanks were removed in 1998 as a part of an effort by UNOCAL to clean up soil contamination due to oil leaks from piping beneath Avila Beach.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

2.2.1 OFFSITE POTENTIAL HAZARDS

2.2.1.1 Description of Location and Routes

Industry in the vicinity of the Diablo Canyon ISFSI site is mainly light and of a local nature, serving the needs of agriculture in the area. Food processing and refining of crude oil are the major industries in the area, although the numbers employed are not large. Less than 8 percent of the work force in San Luis Obispo County is engaged in manufacturing. The largest industrial complex is Vandenberg Air Force Base, located approximately 35 miles south-southeast of the DCPP site in Santa Barbara County.

Port San Luis Harbor and the Point San Luis Lighthouse property are located approximately 6 miles south-southeast of the DCPP site. The Point San Luis Lighthouse is located on a 30-acre parcel of land. Until 1990, the US Coast Guard owned the lighthouse property. In 1990 the Port San Luis Harbor District, owners and operators of the Port San Luis Harbor, were granted ownership of the lighthouse and the 30 acres, except for approximately 3 acres of land, in 3 parcels, which the Coast Guard retained as owners in order to operate and maintain the modern light station and navigating equipment located on those 3 acres.

Located approximately 6 miles east-southeast of the DCPP site is the Port San Luis tanker-loading pier. The pier is located on property that is owned by the Port San Luis Harbor District and leased by UNOCAL, which built and owns the pier. However, this pier is no longer active as tanker traffic into Port San Luis has been discontinued.

US Highway 101 is the main arterial road serving the coastal region in this portion of California. It passes approximately 9 miles east of the site, separated from it by the Irish Hills. US Highway 1 passes approximately 10 miles to the north and carries moderate traffic between San Luis Obispo and the coast. The nearest public access from a US highway is by county roads in Clark Valley, 5 miles north, and See Canyon, 5 miles east. Access to the site is by Avila Beach Drive, a county road, to the entrance of the PG&E private road system.

The Southern Pacific Transportation Company provides rail service to the county by a route that essentially parallels US Highway 101. It passes approximately 9 miles east of the site, separated from it by the Irish Hills. There is no spur track into the DCPP site.

Coastal shipping lanes are approximately 20 miles offshore. Prior to 1998, there were local tankers coming into and out of Estero Bay, which is north of the DCPP site. There is no further tanker traffic in either Port San Luis or Estero Bay. The local tanker terminal at Estero Bay closed in 1994, and the Port San Luis tanker-loading pier ceased operation in 1998. Petroleum products and crude oil are no longer stored at Avila Beach since the storage tanks there were removed in 1999. However, some petroleum products and crude oil continue to be stored at Estero Bay, approximately 10 miles from the DCPP site.

The San Luis Obispo County Regional Airport is located 12 miles east of the DCPP site. The airport served, as a 4-year average between 1998 and 2001, approximately 16,000 air transport (AT) (i.e., commercial or air taxi) landing and departure operations per year. Air transport was provided primarily by turbo-prop or smaller aircraft that seat no more than 41 people with a gross weight of no more than 30,000 pounds.

The airport also served, as a 4 year average between 1998 and 2001, approximately 7,560 total landings and departures of private aircraft per month, including military operations. These consisted mostly of aircraft that seat no more than 8 people, with an average gross weight of less than 12,500 pounds. Although there are no specific air traffic restrictions over DCPP, most air traffic into and out of the San Luis Obispo County Regional Airport does not approach within 5 miles of the ISFSI site because of the mountainous terrain.

There is a federal flight corridor (V-27) approximately 5 miles east of the ISFSI that is used for aircraft flying between Santa Barbara and Big Sur areas, with an estimated 20 flights per day per year-2001 data. The majority of the aircraft using this route are above 10,000 ft. Sometimes this corridor is used also for traffic into San Luis Obispo County Regional Airport and, in this case, has traffic that passes as close as 1 mile of the ISFSI site at an elevation of 3,000 ft. However, this portion of the route is normally only used for aircraft to align for instrument landing. The more commonly used approach route for visual landings passes 8 miles from the Diablo Canyon ISFSI site on the far side of the San Luis Range.

There is also a military training route (VR-249), which runs parallel to the site and its center is approximately 2 miles off shore. This training route is not frequently used. (Estimated based on data from the period of September 2001 and September 2002 at approximately 50 flights per year). Its use requires a minimum of 5 miles visibility, and the flights are to maintain their altitude between sea level and 10,000 ft.

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

The peak Vandenberg Air Force Base employment is approximately 4,400 people, including 3,200 military and 1,200 civilian personnel. At the Vandenberg Air Force Base, there are between 15 to 20 missiles fired per year and currently, missions are flown in a range varying from due west to a southeasterly direction, depending upon launch site and mission. The Vandenberg's Intercontinental ballistic missile tests launch from sites on north base, and typically fly due west. The Vandenberg Air Force Base's spacelift missions typically launch from sites on the southern part of the base, and fly in a southerly direction. The polar orbit launches are launched in a southerly direction. As a result, none of these launches would bring missiles in the vicinity of the ISFSI facility.

There is a potential for missions in the future to fly in a northwesterly direction, but Vandenberg Air Force Base will have safeguards in place to ensure there is no potential for the missile to impact on land outside of Vandenberg Air Force Base's boundary (same techniques used to protect the cities of Lompoc, Santa Barbara, etc., and requires the immediate destruction of any missile that deviates from its intended trajectory.). Deviation from a planned trajectory and destruction of a missile is considered a low probability event by the Air Force.

Vandenberg Air Force Base's most northerly missile launch site is approximately 25 miles south of the DCPP site. Vandenberg Air Force Base is also designated as an alternate landing site for the space shuttles, but has not been used for that purpose to date. The landing approach for a space shuttle would be normally west to east, and does not bring the shuttles within 30 miles of the ISFSI site. Because of the distance to Vandenberg Air Force Base, limited flights, trajectory of the missiles and space shuttle, and the safeguards in place to protect errant launches, there is no credible hazard from this facility.

The nearest US Army installation is the Hunter-Liggett Military Reservation located in Monterey County, approximately 45 miles north of the DCPP site. The California National Guard (CNG) maintains Camp Roberts, located on the border of Monterey County and San Luis Obispo County, southeast of the Hunter-Liggett Military Reservation and approximately 30 miles north of the DCPP site. The CNG also maintains Camp San Luis Obispo, located in San Luis Obispo County, approximately 10 miles northeast of the DCPP site. In addition, as noted earlier, a US Coast Guard Light station is located in Avila Beach on property commonly known as the Point San Luis Lighthouse property.

No significant amounts of any hazardous products are commercially manufactured, stored, or transported within 5 miles of the DCPP site. Within 6 to 10 miles of the site, up to 1998, 1 to 2 local tankers per month offloaded oil for storage at Avila Beach. However, such shipments no longer occur and oil is no longer transported through or stored at Avila Beach. Due to very limited industry within San Luis Obispo County and the distances involved, any hazardous products or materials commercially manufactured, stored, or transported in the areas between 5 and 10 miles from the site are not considered to be a significant hazard to the ISFSI.

2.2.1.2 Hazards from Facilities and Ground Transportation

The ISFSI is located in a remote, sparsely populated, undeveloped area. The ISFSI site is in a canyon, which is east and above DCPP Units 1 and 2, and is directly protected on two sides by hillsides. There are no industrial facilities (other than DCPP), public transportation routes, or military bases within 5 miles of the ISFSI. Therefore, activities related to such facilities do not occur near the ISFSI and, thus, do not pose any hazard to the ISFSI.

Local shipping tankers may come within 10 miles of the DCPP site, but will remain outside of a 5-mile range. Coastal shipping lanes are approximately 20 miles offshore. Therefore, shipping does not pose a hazard to the ISFSI.

No commercial explosive or combustible materials are stored within 5 miles of the site, and no natural gas or other pipelines pass within 5 miles of the site. Therefore, there is no potential hazard to the ISFSI from any explosions or fires involving such materials.

Since there are no rail lines or public transportation routes within 5 miles of the ISFSI location, no credible explosions involving truck or rail transportation events need to be considered, pursuant to Regulatory Guide 1.91 (Reference 1). Similarly, explosions involving shipping events offshore at the DCPP site are unlikely. Although the shortest distance from the ISFSI location to the ocean is approximately 1/2-mile, there is no shipping traffic within 5 miles of this location. Therefore, consistent with the guidance of Regulatory Guide 1.91, explosions involving shipping events are not considered credible accidents for the ISFSI.

2.2.1.3 Hazards from Air Crashes

Aircraft crashes were assessed in accordance with the guidance of NUREG-0800, Section 3.5.1.6, Aircraft Hazards (Reference 2). Although this guidance applies to power reactor sites, the analysis of aircraft crash probabilities on the site is not dependent on the nature of the site other than size of the facility involved and, thus, the guidance of NUREG-0800 can be applied to the Diablo Canyon ISFSI site.

As specified in NUREG-0800, the probability of aircraft crashes is considered to be negligibly low by inspection and does not require further analysis if the three criteria specified in Item II.1 of Section 3.5.1.6 are met. In particular, Criterion 1 of Section 3.5.1.6 specifies that the plant-to-airport distance, D, must be greater than 10 statute miles, and the projected annual number of operations must be less than $1{,}000D^2$. San Luis Obispo County Regional Airport is at a distance of 12 miles, with annual flight totals of approximately 106,720, which is less than $1{,}000(12)^2$ or 144,000. The airport at Oceano is 15 miles away, with flight totals of no more than approximately 26,400 per year, which is less than $1{,}000(15)^2$ or 225,000. Vandenberg Air Force Base is 35 miles away and flight totals there are not expected to be more than $1{,}000(35)^2$ or 1,225,000 per year (or more than 3,300 each day). Camp San Luis Obispo airfield is a heliport facility owned by the US Army, which is approximately 8 miles northeast of the DCPP site. As such, the NUREG-0800 criteria would be $500D^2$ or 32,000 flight operations annually. Data on annual flight operations at this facility are very limited, but based on the guidance in DOE-STD-3014-96 (Reference 5), the type of aircraft using this facility, and the distance to the DCPP site, the threat from aircraft using this facility is considered insignificant. Therefore, based on current data, Criterion 1 is met for the above facilities. However, the airways that are in the vicinity of the Diablo Canyon ISFSI have been analyzed below.

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

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

Evaluation of Airways

For situations where federal airways or aviation corridors pass through the vicinity of the ISFSI site, the probability per year of an aircraft crashing into the site (P_{fa}) is estimated in accordance with NUREG-0800. The probability depends on factors such as altitude, frequency, and width of the corridor and corresponding distribution of past accidents. Per NUREG-0800, the following expression is used to calculate the probability:

$P_{fa} = C \times N \times A/w$

Where:

C = Inflight crash rate per mile for aircraft using airway
w = Width of airway (plus twice the distance from the airway edge to the site when the site is outside the airway) in miles
N = Number of flights per year along airway
A = Effective area of the site in square miles

The following analysis was completed per DOE-STD 3014-96 (Reference 5) to determine effective crash area. In this analysis conservative factors have been used for maximum skid distance and maximum wingspan. Based on the available information on aircraft type, size, and the location of the site these factors are very conservative.

In DOE-STD-3014-96:

The effective crash area is: $A_{eff} = A_f + A_s$

where:

$A_f = (WS + R)\ (H\cot\Phi) + (2)(L)(W)(WS)/R + (L)(W)$

and

$A_s = (WS + R)(S)$

where:

A_f	=	effective fly-in area;
A_s	=	effective skid area;
WS	=	aircraft wingspan; (reference Table B-16 of DOE-STD 3014-96)
R	=	length of diagonal of the facility,
H	=	facility height;
$\cot\Phi$	=	mean on the cotangent of the aircraft impact angle; (reference Table B-17 of DOE-STD 3014-96)
L	=	length of facility;
W	=	width of facility;
S	=	aircraft skid distance; (reference Table B-18 of DOE-STD 3014-96)

For Commercial Aircraft:

$$A_f = (98 + 511)(20)(10.2) + (2)(500)(105)(98)/511 + (500)(105)$$
$$A_f = 196{,}872 \text{ ft}^2/(5{,}280 \text{ ft/mile})^2 = 0.0071 \text{ sq miles}$$

and

$$A_s = (WS + R)(S) = (98 + 511)(700) = 426{,}300 \text{ ft}^2/(5{,}280 \text{ ft/mile})^2$$
$$= 0.0153 \text{ sq miles}$$

For General Aviation Aircraft:

$$A_f = (73 + 511)(20)(10.2) + (2)(500)(105)(73)/511 + (500)(105)$$
$$A_f = 186{,}636\text{ft}^2/(5{,}280 \text{ ft/mile})^2 = 0.0067 \text{ sq miles}$$

and

$$A_s = (WS + R)(S) = (73 + 511)(700) = 408{,}800 \text{ ft}^2/(5{,}280 \text{ ft/mile})^2$$
$$= 0.0147 \text{ sq miles}$$

For Military Aircraft:

$$A_f = (110 + 511)(20)(10.2) + (2)(500)(105)(110)/511 + (500)(105)$$
$$A_f = 201{,}787 \text{ ft}^2/(5{,}280 \text{ ft/mile})^2 = 0.0072 \text{ sq miles}$$

and

$$A_s = (WS + R)(S) = (110 + 511)(700) = 0.0156 \text{ sq miles}$$

For calculating A_s the skid distance is based on the layout of the facility which is surrounded on three sides by hills and is actually up against one of these hills, which limits the potential crash angle and limits the possible skid distance. The fourth side is protected by a drop off in terrain with a slope of greater than 1:1.

The maximum distance on the unprotected side is estimated at less than 700 ft. Since the site is protected and limited from skidding aircraft on three sides, the use of the 700 ft is conservative.

Commercial = A_{eff} = A_f + A_s = 0.0071 + 0.0153 = 0.0224 sq miles

General Aviation = A_{eff} = A_f + A_s = 0.0067 + 0.0147 = 0.0214 sq miles

Military = A_{eff} = A_f + A_s = 0.0072 + 0.0156 = 0.0228 sq miles

For local traffic on V-27:

V-27 use for local aircraft is usually limited to instrument landings for aircraft arriving from the south and instrument departures to the south from runway 11, or circle to land approaches on runway 29, and instrument departures to the south from runway 29 at San Luis Obispo County Regional Airport. As stated above, there are on average approximately 16,000 AT landings and takeoffs per year. It is estimated, using the San Luis Obispo County Regional Airport scheduled airline flight information located at the web address: http://www.sloairport.com/flightinfo.html, that 65 percent of the AT traffic is coming from or departing to the south. Based on airport data over a four-year period from 1998 to 2001 there was an average of 1,781 AT landings per year at San Luis Obispo County Regional Airport under instrument conditions. This would result in (1,781 x 0.65) or 1,157 landings per year, which is doubled to 2,314 operations to account for takeoffs. For the private aircraft usage, including military operations, there are on average approximately 7,560 total landings and takeoffs per month at the San Luis Obispo County Regional Airport of which it is estimated that 65 percent are from or to the south. Based on airport data over a four-year period from 1998 to 2001 there was an average of 1,430 general aviation landings per year at San Luis Obispo County Regional Airport under instrument conditions. As a result, N for general aviation (1,430 x 0.65) or 930 landings, which is doubled to 1,860 operations to account for takeoffs.

Published holding patterns exist for arrivals at CREPE and CADAB intersections and for missed approaches at Morro Bay VOR. The CREPE Intersection is 11 miles and the CADAB Intersection 21 miles from the ISFSI site. Both holding patterns place the aircraft further from the ISFSI site and therefore do not need to be considered. The ISFSI site distance to the Morro Bay VOR is approximately 6 miles and the holding pattern places the aircraft closer to the ISFSI. Since the Morro Bay VOR holding pattern is used for missed approaches, it is conservatively estimated that 5 percent of all instrument landing approaches are missed and each aircraft remains in the holding pattern for 10 passes. For commercial traffic N is increased by 579 flights (2,314/2 x 0.05 x 10) and general aviation by 465 flights (1,860/2 x 0.05 x10).

Per NUREG-0800, C for commercial aircraft is provided as 4×10^{-10}. Per the Aircraft Crash Risk Analysis Methodology Standards (ACRAM), a C value for general aviation of 1.55×10^{-7} was used in this analysis. Per federal guidelines, the width of the airway is

8 miles and the center is approximately 5 miles from the site. As a result, (w) is conservatively taken to equal 10 miles.

For commercial flights:

$$P1a_{fa} = CxNxA/w = (4 \times 10^{-10}) \times (2{,}314 + 579) \times (0.0224)/(10) = 2.59 \times 10^{-9}$$

For general aviation flights:

P1bfa = CxNxA/w =(1.55 x 10-7) x (1,860 + 465) x (0.0214)/10 = 7.7 x 10-7

Total local aircraft crash potential:

P1fa = P1afa + P1bfa = 2.59 x 10-9 + 7.7 x 10-7 = 7.72 x 10-7

<u>For commercial traffic flying on V-27 and not landing locally:</u>

V-27 is a federal flight route from the Santa Barbara area northwest to the Big Sur area. Most of the aircraft on this route are normally flying at altitudes above 10,000 ft, with some smaller aircraft at elevations as low as 3,500 ft. Per the FAA Standards Office, the number of aircraft on this route is conservatively estimated at 20 per day or 7,300 per year. Using the same data as above and adjusting for the number of flights:

$$P2_{fa} = CxNxA/w = (4 \times 10^{-10}) \times (7{,}300) \times (0.0224)/(10) = 6.53 \times 10^{-9}$$

<u>For military aircraft flying on VR-249:</u>

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

$$P3_{fa} = CxNxA/w = (2.736 \times 10^{-8}) \times (75) \times (0.0228)/(1) = 4.68 \times 10^{-8}$$

<u>Military ordnance on aircraft on VR-249</u>

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

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

Summary of aircraft hazards

As stated above, and with the exception of the traffic related to VR-249, Morro Bay VOR and from V-27, the landing patterns and distance to the local airports would not significantly increase the probability of a crash at the ISFSI site. In addition, there are no designated airspaces, which are within the limits of Criterion 2 of NUREG-0800. As result, the total aircraft hazard probability at the Diablo Canyon ISFSI site is equal to the sum of the individual probabilities calculated above.

$$\text{Total} = P1_{fa} + P2_{fa} + P3_{fa} = (7.72 \times 10^{-7}) + (6.53 \times 10^{-9}) + (4.68 \times 10^{-8}) = 8.26 \times 10^{-7}$$

Based on the above calculation, the total aircraft hazard probability is determined to be approximately 8.26×10^{-7}, which is less than the threshold of 1×10^{-6} specified in the Private Fuel Storage SER for acceptable frequency of aircraft impact into a facility from all types of aircraft.

PG&E is aware the NRC is considering revising security regulations, which may affect aircraft hazard requirements relating to aircraft hazards. Following adoption of any new security regulations by the NRC, PG&E will comply with any such revised requirements as appropriate.

2.2.1.3.1 Estimates of Future Potential Hazards from Air Crashes

The projected growth of civilian flights can be based on Federal Aviation Administration (FAA) long-range forecast (FAA, 1999). This includes commercial aircraft operations for air carriers and commuter/air taxi takeoff and landings at all US towered and non-towered airports. In the FAA forecasts, the commercial aircraft operations are projected to increase from 28.6 million in 1998 to 47.6 million in 2025. That results in a projected increase of 66 percent by 2025.

In addition, the annual number of general aviation operations at all towered and non-towered airports in the US is projected by the FAA to increase from 87.4 million in 1998 to 99.2 million in 2025. That results in a projected increase of 14 percent by 2025.

Based on the above potential increases in traffic, the crash probability for local traffic on VR-27 would increase to 8.82 x 10^{-7} and for commercial traffic not landing locally to 1.08 x 10^{-8} by the year 2025.

The FAA also predicts that the military traffic will not increase appreciably, if at all in the foreseeable future. As a result the probability of a crash on VR 249 will remain at 5.6 x 10^{-8}.

Considering all of the FAA projections, the cumulative aircraft crash probabilities increases to 9.4 x 10^{-7} in 2025, which is still less than the threshold of 1 x 10^{-6} specified in the Safety Evaluation Report concerning the Private Fuel Storage Facility, Docket No. 72-22, as an acceptable frequency for impact into the facility from all types of aircraft.

2.2.2 ONSITE POTENTIAL HAZARDS

2.2.2.1 Structures and Facilities

At the DCPP site, including the ISFSI storage site, there are no cooling towers or stacks with a potential for collapse. Therefore, such hazards need not be considered for any potential effects on the ISFSI.

There are 500-kV transmission lines that run in close proximity of the ISFSI storage site and on the hill above it (Figure 2.2-1). A 500-kV transmission line drop is postulated as a result of a transmission tower collapse or transmission line hardware failure near the ISFSI storage site and the cask transfer facility (CTF), as discussed in Section 8.2.8. The worst-case fault condition for a cask is that which places a cask in the conduction path for the largest current. This condition is the line drop of a single conductor of one phase with resulting single line-to-ground fault current and voltage-induced arc at the point of contact.

It is concluded that the postulated transmission line break will not cause the affected cask components to exceed either normal or accident condition temperature limits and that localized material damage at the point of arc on the shell of the overpack and transfer cask water jacket is bounded by accident conditions discussed in Sections 8.2.2 (tornado missile) and 8.2.11 (loss of shielding, HI-TRAC transfer cask water jacket). As a result of the considerations, it is apparent that the postulated transmission line break does not adversely affect the thermal performance of either system.

In addition to the 500-kV lines, the towers that support these lines were evaluated for any potential effect (Figure 2.2-1). They have been evaluated, and although the towers could fail as a result of a severe wind event, there would be no separation of the towers from their foundations, and the towers on the hillside would not have credible contact with the ISFSI storage site. However, the towers, which are located near the ISFSI storage site could, in these events, collapse and strike either the MPC while at the CTF or the loaded overpacks stored on the pads. As a result, as discussed in Section 8.2.16, this impact potential has been evaluated, and it does not adversely affect the MPC or the loaded overpacks.

2.2.2.2 Hazards from Fires

The ISFSI or the fuel storage systems have no credible exposure to fires caused by offsite transportation accidents, pipelines, or manufacturing facilities because of the distance to these transportation routes and the lack of facilities in the proximity of the site. However, there are onsite sources that were evaluated.

Fires are classified as human-induced or natural phenomena design events in accordance with ANSI/ANS 57.9, Design Events III and IV (Reference 3). To identify sources and to establish a conservative design basis for onsite exposure, a walkdown was performed of the CTF, ISFSI storage site, and the complete transportation route from the FHB/AB to the CTF and ISFSI storage site. Based on that walkdown, the following fire events are postulated:

(1) Onsite 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) Fire from mineral oil from the Unit 2 transformers

The potential for fire is addressed for both the HI-STORM 100 overpack and the HI-TRAC transfer cask. Locations where the potential for fire is addressed include the ISFSI storage pad; the area immediately surrounding the ISFSI storage pad, including the CTF; and along the transport route between DCPP and the ISFSI storage pad. These design-bases fires and their evaluations are detailed in Section 8.2.5. This section also discusses various administrative controls to ensure that any fire cannot exceed a design basis for the transfer and storage cask. These administrative controls are further defined in Section 8.2.5 and the evaluations done in support of that section.

For the evaluation of the onsite transporter and other onsite vehicle fuel tank fires (Events 1 and 2), it is postulated that the fuel tank is ruptured, spilling all the contained fuel, and the fuel is ignited. The fuel tank capacity of the onsite transporter is limited to a maximum of 50 gallons of fuel. The maximum fuel tank capacity for other onsite vehicles in proximity to the transport route and the ISFSI storage pads is assumed to be 20 gallons. As discussed in Section 8.2.5, the results of the Holtec analyses for transporter fuel tank rupture fire indicate that neither the storage cask nor the transfer cask undergoes any structural degradation and that only a small amount of shielding material (concrete and water) is damaged or lost. This analysis bounds the 20-gallon onsite vehicle fuel tank fire (Event 2).

The location of any transient sources of fuel in larger volumes, such as tanker trucks, will be administratively controlled to provide a sufficient distance from the CTF, and transport route during transport operations to ensure the total energy received is less than the design-basis fire event. As discussed in Section 8.2.5 an analysis was performed for a ruptured 2000-gallon gasoline tanker truck and determined that it does not result in exceeding the design basis of the storage casks. (The actual tanker truck containing gasoline which would be in the area of the ISFSI will be administratively controlled to no more than 800 gallons.)

All onsite stationary fuel tanks (Event 3) are at least 100 ft from the nearest storage cask, the transport route, and the CTF (Figure 2.2-1). Therefore, there is at least a 100-ft clearance between combustible fuel tanks and the nearest cask in transport, at the CTF, or on the ISFSI storage pads. These existing stationary tanks have been evaluated, but due to their distances to the transport route or the storage pads, the total energy received by the storage cask or the transporter is insignificant compared to the design basis fire event. These tanks will be periodically filled by standard tanker trucks with a capacity of three to four thousand gallons. As discussed in Section 8.2.5, the location of any tanker truck will be administratively controlled to ensure the total energy potentially received at the ISFSI is less than the design basis event or that the risk to the CTF/ISFSI is maintained below the Regulatory Guide 1.91 (Reference 1) credible limit of 1.0×10^{-6}. In addition, during transport operations, all filling will be suspended and these gasoline tanker trucks will not be allowed within 1,100 ft of a cask being transported or the CTF/ISFSI facility. This will be administratively controlled in accordance with the Diablo Canyon ISFSI Technical Specification Cask Transportation Evaluation Program.

For the ISFSI site, the restricted area not covered by the storage pads is paved with asphalt concrete. The outer fence is separated from the inner fence by a distance of approximately 20 ft. The isolation zone (i.e., the region between the fences) is also paved with asphalt concrete. A maintenance program will control any significant growth of vegetation through the pavement. Therefore, the surface of the restricted area will be noncombustible.

No combustible materials will be stored within the security fence around the ISFSI storage pads at any time. In addition, prior to any cask operation involving fuel transport, a walkdown of the general area and transportation route will be performed to assure all local combustible materials (Event 4), including all transient combustibles, are controlled in accordance with administrative procedures.

The native vegetation (Event 5) surrounding the ISFSI storage pad is primarily grass, with no significant brush and no trees. Maintenance programs will prevent uncontrolled growth of the surrounding vegetation. As discussed in Section 8.2.5, a conservative fire model was established for evaluation of grass fires, which has demonstrated that grass fires are bounded by the 50-gallon transporter fuel tank fire evaluation.

The potential fire from mineral oil in the Unit 2 transformers (Event 6) has been evaluated in Section 8.2.5 and found to be bounded by the design basis fire.

In summary, as discussed in Section 8.2.5, the potential effects of any of these postulated fires have been found to be insignificant or acceptable. The physical layout of the Diablo Canyon ISFSI and the administrative controls on fuel sources ensure that the general design criteria related to fire protection specified in 10 CFR 72.122(c) are met (Reference 4).

2.2.2.3 Onsite Explosion Hazards

The storage site has no credible exposure to explosion caused by transportation accidents, pipelines, or manufacturing facilities because of the distance to these transportation routes and the lack of facilities in the proximity of the site. However, there are potential onsite hazards that must be evaluated.

Explosions are classified as human-induced or natural phenomena design events in accordance with ANSI/ANS 57.9 Design Events III and IV. To determine the potential explosive hazards, which could affect the ISFSI or the fuel transportation system, a walkdown of the ISFSI storage area and the transportation route from the FHB/AB was completed. The following explosion sources and event categories have been identified and evaluated in Section 8.2.6:

(1) Detonation of a cask transporter or an onsite vehicle fuel tank

(2) Detonation of propane bottles transported past the ISFSI storage pad

(3) Detonation of compressed gas bottles transported past the ISFSI storage pad

(4) Detonation of large stationary fuel tanks in the vicinity of the transport route

(5) Explosive decompression of a compressed gas cylinder

(6) Detonation of the bulk hydrogen storage facility

(7) Detonation of acetylene bottles stored on the east side of the cold machine shop

Figure 2.2-1 shows the location of the stationary potential sources (sources 4, 6 and 7). Events 1, 2, 3, and 5 are assumed to occur in the vicinity of the ISFSI storage pads, CTF, or transport route and potentially affect both the loaded overpack and the transfer cask. Events 4, 6, and 7 occur in the vicinity of the transport route and affect the transfer cask. This section also discusses various administrative controls to ensure that any potential explosion hazards will meet the Regulatory Guide 1.91 criteria or methodologies. These administrative controls are further defined in Section 8.2.6 and the evaluations done in support of that section.

In all of the above evaluations, the effects on the loaded overpacks or transport cask are either minimal or not credible, and there will be no loss of function. For Events 1 through 3, as discussed in Section 8.2.6, the risk of exceeding the Regulatory Guide 1.91 overpressure criterion of 1 psi is not significant. In addition the transportation practices and the physical distance to the storage pads, CTF, or transporter are controlled by administrative procedures. For Event 4, the distance of the existing fuel tanks from the transportation route precludes any effect on the transportation of the spent fuel to the storage pads or CTF. Event 5 concerns decompression of gas cylinders and the possible missile damage to the transfer cask and overpack. The evaluation performed in Section 8.2.6 shows that this is not a credible event and that there would be no significant damage or loss of function by this event. Event 6 involves the transportation of the transfer cask past a potential hydrogen explosion hazard (Figure 2.2-1). Section 8.2.6 discusses the evaluation that was performed for this event. The evaluation shows that the probability of a detonation at the moment the transporter is in the vicinity is so small that it is not credible per the guidelines of Regulatory Guide 1.91. Event 7 was evaluated in Section 8.2.6 where it is shown that the detonation of the acetylene bottles stored on the east side of the cold machine shop was not a credible event based on configuration, restraints, and lack of an ignition source.

Also under Event 1, it not only refers to an average 20-gallon vehicle fuel capacity, but gasoline tanker trucks that transport fuel near the ISFSI facility. The onboard fuel volumes being carried by any trucks that pass within 1,000 ft of the CTF/ISFSI facility will be administratively controlled to a maximum of 800 gallons. These trucks will only be in this area momentarily while passing by the ISFSI facility and will be under administrative controls for their speed and continued movement through the area on its way to and from the vehicle maintenance shop that is located approximately 2,000 ft northeast of the ISFSI pad. As discussed in Section 8.2.6, a probabilistic risk assessment was performed and it was determined, based on the use of administrative controls and the restriction for movement and stopping within the separation distance calculated based on the 1 psi Regulatory Guide 1.91 criterion, that the risk is insignificant.

The Cask Transportation Evaluation Program will be developed, implemented, and maintained to ensure that no additional hazards are introduced either at the storage pads, CTF, or on the transportation route during onsite transport of the loaded overpacks or transfer cask. That program will include limitation on hazards and will require a transportation route walkdown prior to any movement of the transporter with nuclear fuel between the FHB/AB and the CTF, and between the CTF and the storage pads. The walkdown will require the evaluation or removal of any identified hazards prior to the movement of the transporter. DCPP procedures will control all movement of vehicles or activities during onsite transport that could have an adverse effect on the loaded overpacks or transfer cask.

2.2.2.4 Chemical Hazards

A walkdown of all chemical hazards was performed in the ISFSI storage pad and CTF areas, and along the transportation route. Chemical hazards were identified that could have an effect on the ISFSI or the transportation system. To ensure minimum potential for chemical hazards, the administrative program provided to control fire and explosive hazards will also include identification, control, and evaluation of hazardous chemicals.

2.2.2.5 Helicopter Activities

Helicopters are periodically used to perform inspections and washes of the transmission infrastructure adjacent to the ISFSI storage site and CTF to support maintenance activities. To prevent onsite aircraft hazards, administrative controls do not allow aircraft to fly directly above or within the ISFSI storage site.

2.2.3 Summary

In summary, there are no credible accident scenarios involving any offsite industrial, transportation, or military facilities in the area around the DCPP site that will have any significant adverse impact on the ISFSI. In addition, there are no potential onsite fires, explosions, or chemical hazards that would have a significant impact on the ISFSI.

2.2.4 REFERENCES

1. Regulatory Guide 1.91, Evaluations of Explosions Postulated to Occur on Transportation Routes near Nuclear Power Plants, US Nuclear Regulatory Commission, February 1978.

2. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, USNRC, NUREG-0800, July 1981.

3. ANSI/ANS 57.9, 1992, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type), American National Standards Institute.

4. 10 CFR 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

5. DOE-STD-3014-96 Accident Analysis for Aircraft Crash Into Hazardous Facilities, US Department of Energy, October 1996.

2.3 METEOROLOGY

The meteorology of the Diablo Canyon area is described in Section 2.3 of the DCPP FSAR Update. Information in the FSAR Update includes discussion of the regional climatology, local meteorology, topographical information, onsite meteorological measurement program, and diffusion estimates for the Diablo Canyon owner-controlled area, which includes the ISFSI site. Relevant tables and figures supporting the discussion are included in the FSAR Update.

Meteorological conditions for the ISFSI site are expected to be the same as for DCPP since the ISFSI site is located approximately 0.22 miles and slightly uphill from the DCPP facilities. No significant changes in climate or meteorological characteristics can occur within such a short distance and, thus, existing meteorological measurements for DCPP are expected to be equally applicable to the ISFSI. Diffusion estimates at the ISFSI site are provided in Section 2.3.4.

The FSAR Update is maintained up to date by PG&E through periodic revisions made in accordance with 10 CFR 50.71(e). Hence, the information contained in the FSAR Update is current, and no further revision is necessary for applicability to the ISFSI. Therefore, in accordance with the guidance of Regulatory Guide 3.62, material from Section 2.3 of the FSAR Update is incorporated herein by reference in support of the ISFSI license application. The following paragraphs provide a brief summary of various discussions from Section 2.3 of the FSAR Update.

2.3.1 REGIONAL CLIMATOLOGY

The climate of the area is typical of the central California coastal region and is characterized by small diurnal and seasonal temperature variations and scanty summer precipitation. The prevailing wind direction is from the northwest, and the annual average wind speed is about 10 mph. In the dry season, which extends from May through September, the Pacific high-pressure area is located off the California coast, and the Pacific storm track is located far to the north. Moderate to strong sea breezes are common during the afternoon hours of this season while, at night, weak offshore drainage winds (land breezes) are prevalent. There is a high frequency of fog and low stratus clouds during the dry season, associated with a strong low-level temperature inversion.

The mountains that extend in a general northwest-to-southeast direction along the coastline affect the general circulation patterns. This range of mountains is indented by numerous canyons and valleys, each of which has its own land-sea breeze regime. As the air flows along this barrier, it is dispersed inland by the valleys and canyons that indent the coastal range. Once the air enters these valleys and canyons, it is controlled by the local terrain features.

The annual mean number of days with severe weather conditions, such as tornadoes and ice storms at west coast sites, is zero. Thunderstorms and hail are also rare phenomena, the average occurrence being less than 3 days per year. The maximum-recorded precipitation in the San Luis Obispo region is 5.98 inches in 24 hours at San Luis Obispo. The 24-hour maximum occurred on March 4, 1978.

The maximum-recorded annual precipitation at San Luis Obispo was 54.53 inches during 1969. The average annual precipitation at San Luis Obispo is 21.53 inches. There are no fastest mile wind speed records in the general area of Diablo Canyon, surface peak gusts at 46 mph have been reported at Santa Maria, California, and peak gusts of 84 mph have been recorded at the 250 ft level at the Diablo Canyon site.

The monthly average temperatures for San Luis Obispo from 1948 to 2000 are provided in Table 2.3-1.

2.3.2 LOCAL METEOROLOGY

The average annual temperature at the ISFSI site is approximately 55°F (based on measurements made at the DCPP primary meteorological tower). Generally, the warmest mean monthly temperature occurs in October, and the coldest mean monthly temperature occurs in December. The highest and lowest hourly temperature, as recorded at one of the recording stations, is 97°F in October 1987, and 33°F in December 1990, respectively. Essentially no snow or ice occurs at the ISFSI site.

Solar radiation data considered representative of the Diablo Canyon ISFSI site is collected by the California Irrigation Management Information System (CIMIS), Department of Water Resources, at the California Polytechnic State University in San Luis Obispo, California. The CIMIS collection site is about 12 miles northeast of the Diablo Canyon ISFSI site. For a period of record between May 1, 1986 and December 31, 1999, the maximum measured incident solar radiation (insolation) values at the CIMIS site were 766 g-cal/cm^2 per day for a 24-hour period and 754 g-cal/cm^2 per day for a 12-hour period, both on June 1, 1989. The daily (24-hour) average for the period of record was 430 g-cal/cm^2 per day. For the Diablo Canyon ISFSI site, the insolation values would likely be lower than the CIMIS values because of more frequent fog in the ISFSI area.

The average annual precipitation at the DCPP site is approximately 16 inches. The highest monthly total recorded between 1967 and 1981 was 11.26 inches. The greatest amount of precipitation received in a 24-hour period was 3.28 inches. These maxima were recorded in January 1969 and March 1978, respectively. The maximum hourly amount recorded in the Diablo Canyon area during the same period is 2.35 inches.

The highest recorded peak wind gust at the primary meteorological tower is 84 mph, and the maximum-recorded hourly mean wind speed is 54 mph. Persistence analysis of wind directions in the Diablo Canyon area shows that, despite the prevalence of the marine inversion and the northwesterly wind flow gradient along the California coast, the long-term accumulation of emissions in any particular geographical area downwind is virtually impossible. Pollutants injected into the marine inversion layer of the coastal wind regime are transported and dispersed by a complex array of land-sea breeze regimes that exist all along the coast wherever canyons or valleys indent the coastal range.

Topographical influences on both short-term and long-term diffusion estimates are pronounced in that the ridge lines east of the ISFSI location extend at least to the average height of the marine inversion base. The implications of this barrier are:

(1) Any material released that is diverted along the coastline will be diluted and dispersed by the natural valleys and canyons, which indent the coastline.

(2) Any material released that is transported over the ridgeline will be distributed through a deep layer because of the enhanced vertical mixing due to topographic features.

2.3.3 ONSITE METEOROLOGICAL MEASUREMENT PROGRAM

The current onsite meteorological monitoring system supporting DCPP operation will serve as the onsite meteorological measurement program for the ISFSI. The system consists of two independent subsystems that measure meteorological conditions and process the information into useable data. The measurement subsystems consist of a primary meteorological tower and a backup meteorological tower. The program has been designed and continually updated to conform with Regulatory Guide 1.23.

A supplemental meteorological measurement system is also located in the vicinity of DCPP. The supplemental system consists of three Doppler SODAR (Sonic Detection and Ranging) systems and seven tower sites. Data from the supplemental system are used for emergency response purposes to assess the location and movement of any radioactive plume.

2.3.4 DIFFUSION ESTIMATES

For ISFSI dose calculations required by 10 CFR 72.104, (normal operations and anticipated occurrences), site boundary χ/Q values range from 9.2×10^{-8} to 3.4×10^{-6} sec/m^3 and nearest residence χ/Q values range from 2.0×10^{-8} to 4.2×10^{-7} sec/m^3. These values are taken from Table 11.6-13 of the DCPP FSAR Update and have been determined to be applicable to the ISFSI site. They will be used, as appropriate, for dose calculations related to normal operations and anticipated occurrences.

Compliance with 10 CFR 72.106 requires calculation of design basis accident doses at the controlled area boundary (site boundary for the Diablo Canyon ISFSI), which is about 400 meters from the ISFSI at its closest point. Based on information from the DCPP FSAR Update, Section 2.3.5 and Table 2.3-41, a χ/Q of 4.5×10^{-4} sec/m^3 has been determined to be a conservative estimate applicable to the ISFSI site and will be used for accident dose calculations.

2.4 SURFACE HYDROLOGY

Hydrologic information pertaining to the Diablo Canyon area in general has been documented in the DCPP FSAR Update (Reference 1). Much of this information pertains also to the ISFSI location since the hydrologic characteristics in the Diablo Canyon area do not vary significantly in the general vicinity of the ISFSI and power plant facilities. Specific features relevant to hydrologic engineering at the ISFSI location are described in this section, with reference to supporting information in the DCPP FSAR Update where appropriate.

2.4.1 HYDROLOGIC DESCRIPTION

The topography and an outline of the drainage basin in the region surrounding the ISFSI site are shown in Figure 2.4-1. This map is reproduced from the US Geological Survey (USGS) Port San Luis and Pismo Beach 7.5-minute topographic quadrangles. The basin drains to Diablo Creek, which discharges into the Pacific Ocean. Figure 2.4-2 shows the Diablo Creek drainage basin to a larger scale. The basin encompasses approximately 5 square miles and is bounded by ridges reaching a maximum elevation of 1,819 ft above mean sea level (MSL) at Saddle Peak, located approximately 2 miles to the east of the ISFSI.

The hydrologic characteristics of the ISFSI site are influenced by the Pacific Ocean on the west and by local storm runoff collected from the basin drained by Diablo Creek. The maximum and minimum flows in Diablo Creek are highly variable. Average flows tend to be nearer the minimum flow value of 0.44 cfs. Maximum flows reflect short-term conditions associated with storm events. Usually within 1 or 2 days following a storm, flows return to normal. Flows during the wet season (October-April) vary daily and monthly. Dry season flows are sustained by groundwater seepage and are more consistent from day to day, tapering off over time. There is no other creek or river within the site area or the drainage basin.

Water is supplied to DCPP from two sources: one site well, and an ocean water desalinization plant that has been used since 1985.

2.4.2 FLOODS

The DCPP FSAR Update addresses flood considerations pertinent to the power plant facilities at Diablo Canyon. The following discussion identifies flood considerations from the DCPP FSAR Update that are pertinent to the ISFSI location. Topography and ISFSI site structures limit flood design considerations to local floods from Diablo Creek. The canyon confining Diablo Creek will remain intact and is more than sufficient to channel any conceivable flood without any hazard to the ISFSI. Channel blockage from any landslides downstream of the ISFSI location and to an extent sufficient to flood the ISFSI area is not possible because of the topographic location and elevation of the ISFSI.

There are no dams or natural features in Diablo Creek that would hinder or retain runoff for a significant period of time. At the ISFSI, runoff can be efficiently drained by the adjacent natural and constructed drainage features.

If the culverts and drainage out of the ISFSI area become plugged during periods of high precipitation, water may locally and temporarily pond. Drainage in the vicinity of the ISFSI is shown in Figure 2.4-3. No significant ponding should occur since, due to the open terrain and location, any additional runoff into the ISFSI area will drain away from the facility toward Diablo Creek or the ocean. No adverse impact is expected on ISFSI operation or spent fuel confinement.

Two water reservoirs constructed in rock and located in the vicinity of the ISFSI maintain redundant water supplies in support of operation of Units 1 and 2. If the reservoirs were to overflow due to an unlikely accumulation of runoff from high precipitation, the local topography would cause water to drain toward the creek and ocean. No adverse impact on the ISFSI would be expected from overflow of the reservoirs.

2.4.3 PROBABLE MAXIMUM FLOOD (PMF) ON STREAMS AND RIVERS

Diablo Creek is the only significant channel for the drainage basin within which the ISFSI is located. This drainage basin includes approximately 5.2 square miles. The potential PMF upstream of the location of the power plant facilities was found to have a peak discharge of approximately 6,900 cfs, with a total volume of approximately 4,300 acre-ft for a 24-hour storm.

As documented in the DCPP FSAR Update, the drainage capacity of Diablo Creek through this area is more than sufficient to efficiently channel the PMF volume directly into the Pacific Ocean with no retention time. This volume of water discharged from the Diablo Creek basin will not cause any local flooding around the power plant or overtop the switchyards, even if the 10-ft diameter culvert passing under the switchyards were to temporarily plug. If the culvert were plugged, any water impounded east of the 500-kV switchyard would be discharged along Diablo Creek Road (elevation of approximately 250 ft MSL opposite the ISFSI) and through the stilling basin located between the switchyards. The floodwaters would pass through the diversion scheme with adequate freeboard near each switchyard, on the opposite side of the canyon, and below the elevation of the ISFSI (310 ft MSL). The water released would not cause any flooding of the ISFSI.

2.4.4 POTENTIAL DAM FAILURES (SEISMICALLY INDUCED)

There are no dams in the watershed area. Outside the watershed area, any seismic-induced failure of dams would not affect the ISFSI.

2.4.5 PROBABLE MAXIMUM SURGE AND SEICHE FLOODING

Due to the elevation of the ISFSI, there is no credible scenario that would create any flooding from a maximum surge or seiche.

2.4.6 PROBABLE MAXIMUM TSUNAMI FLOODING

Due to the elevation of the ISFSI, a maximum tsunami would not cause any flooding to the ISFSI.

The maximum combined wave runup from a distantly generated tsunami is 30 ft (Reference 1, Section 2.4.6.1.3), and the maximum combined wave runup for near shore tsunamis is 34.6 ft relative to a mean lower low water (MLLW) reference datum (Reference 1, Section 2.4.6.1.4). This is significantly lower than the elevation of the Diablo Canyon ISFSI site at 310 ft above mean sea level (MSL) (312.6 ft above MLLW) or the transporter route at 80 ft above MSL.

Additional data and analysis related to the maximum possible tsunami are provided in Reference 2 (PG&E Response to NRC Question 2-14).

2.4.7 ICE FLOODING

Flooding due to ice melt events is not credible because of the mild climate and infrequency of freezing temperatures in the region.

2.4.8 FLOOD PROTECTION REQUIREMENTS

No cooling water canals, reservoirs, rivers or streams are used in operation of the ISFSI. There are no channel diversions in the region that can alter any water flow patterns so as to affect the ISFSI. Hence, low flow conditions need not be considered.

Based on these considerations, there are no credible hydrological scenarios that can adversely affect the ISFSI. Thus, specialized hydrological engineering considerations and flood protection requirements for the ISFSI facilities are not necessary. Only typical grading and drainage provisions for storm runoff are needed.

2.4.9 ENVIRONMENTAL ACCEPTANCE OF EFFLUENTS

Section 3.3.1.7.2 indicates that there are no radioactive wastes created by the HI-STORM 100 System while in storage at the storage pads, transport to or from the CTF, or at the CTF.

Environmental Report Sections 2.5, 4.1, and 4.2 address the environmental effects of potential effluents from the ISFSI. It is concluded that surface runoff from the ISFSI has no radioactive contamination and will not adversely affect the surrounding ecosystem.

Ocean water is the only source of surface water used at DCPP for support of power plant operation. No water is used to support ISFSI operation. Potable water used to support ISFSI administration is provided by existing systems at DCPP. Such support of ISFSI administrative activities will be provided according to plant procedures. No other surface or groundwater sources (except Well No. 2) exist or are used in this area. There is no public use of any surface waters or groundwater from the Diablo Canyon site. Therefore, no detailed analysis of acceptance of effluents by surface waters or groundwater due to ISFSI operation is relevant.

2.4.10 REFERENCES

1. PG&E, Units 1 and 2 Diablo Canyon Power Plant, Final Safety Analysis Report Update.

2. PG&E Letter DIL-02-009 to the NRC, Response to NRC Request for Additional Information for the Diablo Canyon ISFSI Application, October 15, 2002.

2.5 SUBSURFACE HYDROLOGY

This section is based on information provided in the DCPP FSAR Update and recent geotechnical investigations performed to characterize the ISFSI, CTF, and transport route.

2.5.1 GROUNDWATER IN DCPP AREA

Groundwater in the DCPP area is found in the narrow, relatively thin gravel alluvium along Diablo Creek, in fractures in the bedrock of the Obispo Formation, and along the contact that marks the top of bedrock and the base of some of the extensive terrace deposits that flank the coast. Two seeps and a small spring were encountered during excavations for the power plant.

The main groundwater table beneath the coastal terrace north and south of the power plant is controlled by sea level at the coastline and gradually rises beneath the hills southeast of the power plant. Hence, this water table beneath the power plant and the ISFSI is about the elevation of Diablo Creek, sloping upward from sea level at the coast to 200 ft above the 500-kV switchyard.

Groundwater in the alluvium of Diablo Creek has been documented from the makeup water wells. Makeup water wells No. 1 and No. 2 with collar elevations at 232 ft above mean sea level (MSL) and 329 ft MSL, respectively, have produced water from the alluvium in Diablo Creek and from fractured sandstone and dolomite of the Obispo formation. Well No. 1 is no longer in use (since 2008). The water table varies, depending on the month of the year, but is generally controlled by flows in the alluvium near elevation 200 ft MSL.

Groundwater above the base of the thick terrace deposits is recorded in several places. On the terrace north of Diablo Creek, monitoring wells MW-1 through MW-4 (collar elevations range between 115 and 210 ft MSL) at the closed waste holding pond showed water levels in 1985 at elevations between 64 and 127.5 ft MSL. In parking lot 7, south of DCPP, two piezometers in 1996 and 1997, recorded groundwater at a depth of 40 to 77 ft and recording a perched water table near the top of the wave-cut bedrock platform. Groundwater seeps also issue from a perched water table on the marine terrace platform (about 30 ft MSL) in Patton Cove. Local perched water tables also occur within the Obispo Formation above the marine bedrock platforms. These perched water tables occur on impermeable strata, such as clay beds, within the Obispo Formation. An example is the small spring that issues from the hillslope above and east of Patton Cove at elevation about 600 ft MSL. A few areas of dense vegetation indicative of seeps also issue from bedrock along the lower canyon walls of Diablo Creek below the raw water reservoir.

2.5.2 GROUNDWATER AT ISFSI

As discussed above, groundwater beneath the ISFSI site is controlled by the elevation of water in Diablo Creek that is at about elevation 100 ft MSL opposite the ISFSI. This is at least 190 ft below the ISFSI pads, which are at elevation 310 ft MSL.

Clay beds beneath the ISFSI could impede groundwater infiltration and form temporary perched water tables during the rainy season. In all but one of the 15 borings drilled at and near the ISFSI site, no evidence of a perched water table was found during drilling. Typically the clay beds in the core were moist, but not saturated, indicating no perched water at the time of drilling. However when boring 01-F was being drilled on the slope above the ISFSI a rainstorm soaked the Diablo Canyon area in the night. The next morning clear water was observed issuing from the borehole that was 29 ft deep, but the flow stopped and was at 6.5 ft deep by the time the drilling was started; analysis of the boring shows a very thin clay on bedding at 6.8 ft but other clay beds are deeper than 29 ft. These data confirm that temporary perched water can accumulate locally in the slope above less permeable beds (Reference 1, Data Report B). In addition the dense vegetation, indicative of moist rock, is 20 to 30 ft above Diablo Creek in the lower canyon wall north of the ISFSI. This and other seeps are evident in the upper canyon wall north of the ISFSI site mark perched water seeping out above impermeable beds.

2.5.3 GROUNDWATER AT CTF

Groundwater levels below the CTF are near the elevation of Diablo Creek, at elevation 100 ft MSL, as described in Section 2.5.2. This is at least 190 ft below the CTF, which is at elevation 310 ft MSL.

2.5.4 GROUNDWATER ALONG TRANSPORT ROUTE

The main groundwater levels beneath the transport route are controlled by the elevation of water in Diablo Creek (25 to 75 ft MSL) near DCPP and the ISFSI and by sea level along the coastal terrace. Estimated groundwater levels beneath the transport route are as follows:

Plant View and Shore Cliff Roads - The route crosses the lower marine terraces and the regional groundwater table probably is slightly above sea level and is more than 50 to 100 ft below the roadway. In places, a perched groundwater table occurs, locally above the contact between the bedrock and the overlying marine terrace. This perched water is 30 to over 50 ft below the roadway.

Reservoir Road - The route generally follows the hillside where the road has been cut into dolomite and sandstone bedrock of the Obispo Formation. The strata dips into the hillslope away from the road. The regional groundwater in this area lies near the same elevation (about 100 ft MSL) as beneath the ISFSI site (Section 2.5.2), some 50 to 140 ft below the roadway. The clay beds in the sandstone bedrock may become temporary groundwater barriers that slow the percolation of water through the fractured rock of the slope, but these beds dip into the slope away from the road and no seeps indicative of perched water are known along this part of the slope.

2.5.5 GROUNDWATER SUMMARY

Based on available information, groundwater quality or quantity is not expected to be affected by construction or operation of the ISFSI, CTF, or access road. Construction and operation of the ISFSI does not involve the use of groundwater, and there is no public use of onsite groundwater. The occurrence of temporary perched water over clay beds in the dolomite and sandstone bedrock that underlie the slopes above the ISFSI and the transport route has no adverse effects on the ISFSI or the transport route because any potential effect will be mitigated by drains in the proposed cutslopes above the ISFSI, or other means as described in Section 4.2.1.1.9.

2.5.6 REFERENCES

1. Geologic Data Reports A through K, William Lettis & Associates, December 2001.

2.6 GEOLOGY AND SEISMOLOGY

The Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI) will be located directly inland from the power plant on a graded bedrock hillslope adjacent to the DCPP raw water reservoir (Figure 2.6-1). It was desirable to select a site having bedrock properties and earthquake response characteristics comparable to those of the bedrock beneath the Diablo Canyon power block, such that existing DCPP design-basis ground motions could be used in the design of the ISFSI.

In this section, the geologic and seismologic conditions in the region are described and evaluated. Detailed information is provided regarding the earthquake vibratory ground motions, foundation characteristics, and slope stability at the ISFSI and CTF sites. Information regarding foundation characteristics and slope stability also is provided for the transport route between the power block and the CTF. The information is in compliance with the criteria in Appendix A of 10 CFR 100, and 10 CFR 72.102, and meets the format and content recommendations of Regulatory Guide 3.62. Several commercial technical computer software programs were used to assist in the analyses performed for Section 2.6.

An external, independent Seismic Hazards Review Board advised on the studies carried out for this section. A letter summarizing the conclusions of the consulting board is provided in Reference 1, as are the names and affiliations of the project team responsible for preparation of this section.

Definitions

For the purposes of Section 2.6, the following definitions and boundaries were used to describe the ISFSI study area and plant site region, as illustrated on Figure 2.6-1 (definitions of other terms used in this report are in the glossary at the front of the report):

- plant site region: the area of the Irish Hills and vicinity within a 10-mile radius of the Diablo Canyon ISFSI

- plant site area: the area within the DCPP boundary

- ISFSI study area: the area extending along the nose of the ridge behind the power plant and encompassing the ISFSI site and CTF site

Conclusions

Geologic, seismologic, and geotechnical investigations for the ISFSI yielded the following conclusions:

- The ISFSI will be founded on bedrock that is part of the same continuous, thick sequence of sandstone and dolomite beds upon which the DCPP power block is sited. The shear wave velocity characteristics of the rock at the ISFSI and CTF sites are within the same range as those at the power block. Additionally, the ISFSI and CTF sites are approximately the same distance from the Hosgri fault zone, the controlling earthquake source for the DCPP. Thus, the foundation conditions and ground-motion response characteristics are the same as those at the DCPP (discussed in Section 2.6.1.10).

- Because the ground-motion response characteristics at the ISFSI are the same as those at the DCPP, the DCPP earthquake ground motions are appropriate for use in the licensing of the ISFSI, in accordance with 10 CFR 72.102(f) (discussed in Section 2.6.2).

- Because ISFSI pad sliding, slope stability, and the stability of the transporter are affected by longer-period ground motions than those characterized by the DCPP ground motions, response spectra having a longer-period component were developed. The longer-period component conservatively incorporates the near-fault effects of fault rupture directivity and fling. These spectra, referred to as the ISFSI long-period ground motions (ILP), and associated time histories, were used to analyze elements that may be affected by longer-period ground motions (discussed in Section 2.6.2.5).

- Several minor bedrock faults were observed at the ISFSI and CTF sites. These minor faults are not capable; hence, there is no potential for surface faulting at the ISFSI or CTF sites (discussed in Section 2.6.3).

- The sandstone and dolomite bedrock, including zones of friable rock, that underlies the ISFSI and CTF sites area is stable, and has sufficient capacity to support the loads imposed by the ISFSI pads and casks and the CTF without settlement or differential movement (discussed in Section 2.6.4).

- There are no active landslides or other evidence of existing ground instability at the ISFSI and CTF sites, or on the hillslope above the ISFSI site (discussed in Section 2.6.1.12).

- The stability of the hillslope and the slopes associated with the pads, CTF, and transport route under static and seismic conditions was analyzed using conservative assumptions regarding slope geometry, material properties, seismic inputs, and analytical procedures (discussed in Section 2.6.5). The analyses show that the slopes have ample factors of safety under static conditions. The cutslope above the ISFSI site may experience local wedge movements or small displacements if exposed to the design-basis earthquakes. Mitigation measures to address these movements are described in Sections 4.2.1.1.9.1 and 4.2.1.1.9.2.

- The transport route follows existing paved roads, except for a portion of the route that will be constructed to avoid a landslide at Patton Cove along the coast. The route will have foundation conditions satisfactory for the transporter (discussed in Section 4.3.3). Small debris flows could potentially close portions of the road during or immediately following severe weather (discussed in Section 2.6.5.4). Because the transport route will not be used during severe weather, the flows will not be a hazard to the transporter.

2.6.1 GEOLOGIC, SEISMOLOGIC AND GEOTECHNICAL INVESTIGATIONS

Extensive geologic, seismologic and geotechnical investigations were performed to characterize the ISFSI and CTF sites. These investigations included compilation and review of pre-existing information developed for construction of the power plant, the raw water reservoir, and the 230-kV and 500-kV switchyards, as well as extensive detailed investigations performed in the ISFSI study area. These investigations are described in References 2 and 3. The investigations focused on collecting information to address four primary objectives:

- to evaluate foundation properties beneath the ISFSI pads, the CTF facility, and the transport route

- to evaluate the stability of the proposed cutslopes and existing hillslope above the ISFSI pads and along the transport route

- to identify and characterize bedrock faults at the site

- to compare bedrock conditions at the ISFSI site with bedrock conditions beneath the DCPP power block for the purpose of characterizing earthquake ground motions

Investigations in the plant site area included interpretation of aerial photography, review of existing data and literature, and field reconnaissance. In particular, borehole and trench data collected in the 1960s and 1970s for the power plant were compiled, reviewed, and used to evaluate stratigraphic conditions beneath the power block and between the power block and the ISFSI site.

Investigations in the ISFSI study area and along the transport route were conducted to develop detailed information on the lithology, structure, geometry, and physical properties of bedrock beneath the ISFSI and CTF sites, and beneath the transport route. Investigations of the ISFSI and CTF site geology included 17 borings at 14 locations, 22 trenches and test pits, a seismic refraction survey, down-hole geophysics and televiewer surveys, petrographic analysis of rock samples, laboratory analysis of soil and rock properties, and detailed surface mapping (Reference 3). These data were used to develop a detailed geologic map of the plant site area, the

ISFSI study area and transport route, and 12 geologic cross sections to illustrate the subsurface distribution of bedrock lithology and structure (Reference 37).

2.6.1.1 Existing Geologic, Seismologic, and Geotechnical Information

Existing geologic, seismologic, and geotechnical information includes that collected for licensing the operating DCPP, construction of the raw water reservoir, and construction of the 230-kV and 500-kV switchyards. Regional and site-specific geologic, seismologic and geotechnical investigations at the DCPP site are documented in Sections 2.5.1 and 2.5.2 of the DCPP Final Safety Analysis Report (FSAR) Update, submitted in support of continued operation of Units 1 and 2 (Reference 4). In response to License Condition Item 2.C.(7) of the Unit 1 Operating License DPR-80, issued in 1980, PG&E was required to reevaluate the seismic design bases for the DCPP. This reevaluation became known as the Long Term Seismic Program (LTSP). The program was conducted between 1985 and 1991. In June 1991, the Nuclear Regulatory Commission (NRC) issued Supplement Number 34 to the Diablo Canyon Safety Evaluation Report (Reference 5), in which the NRC concluded that PG&E had satisfied License Condition Item 2.C.(7). The LTSP evaluations are docketed in the LTSP Final Report (Reference 6) and the Addendum to the Final Report (Reference 7). The information presented herein summarizes and refers to the DCPP FSAR Update and LTSP reports.

Existing regional and site-specific geologic, seismologic and geotechnical information for the ISFSI is discussed in the following docketed references:

Regional physiography:	DCPP FSAR Update, Section 2.5.1.1.1
Geologic setting:	DCPP FSAR Update, Section 2.5.1.1.2.1 LTSP Final Report, Chapter 2
Tectonic features:	DCPP FSAR Update, Sections 2.5.1.1.2.2 and 2.5.1.1.2.3 LTSP Final Report, Chapter 2
Geologic history:	DCPP FSAR Update, Section 2.5.1.1.3 LTSP Final Report, Chapter 2
Regional geologic structure and stratigraphy:	DCPP FSAR Update, Sections 2.5.1.1.4 and 2.5.1.1.5 LTSP Final Report, Chapter 2
Geologic structure and stratigraphy of the plant site area:	DCPP FSAR Update, Section 2.5.1.2 LTSP Final Report, Chapter 2
Slope stability of the plant site area:	Slope Stability Report

Earthquake history and association of earthquakes with geologic structures:	DCPP FSAR Update, Sections 2.5.2.5 and 2.5.2.6 LTSP Final Report, Chapter 2
Maximum earthquakes affecting the plant site area:	DCPP FSAR Update, Section 2.5.2.9 LTSP Final Report, Chapter 3
Earthquake ground accelerations and response spectra:	DCPP FSAR Update, 2.5.2.10 and 3.71 LTSP Final Report, Chapter 4

The ISFSI is sited on a bedrock slope that was previously used as a source of fill materials for construction of the 500-kV and 230-kV switchyards. The first geologic and geotechnical studies in the area were performed by Harding Miller Lawson & Associates (HML) (Reference 9). The study was conducted prior to the borrow excavation to obtain information regarding the excavatability and suitability of the site materials for switchyard fills. Their investigations included borings, test pits, and refraction surveys. The depth of their explorations, however, was limited to the depth of the planned (and as-built) borrow excavation, and did not extend below the present post-excavation site elevations. All the material investigated by HML was removed during the borrow excavation and used for construction of the switchyard fills.

In addition, an assessment of slope stability near the DCPP was performed following the heavy winter storms of 1996-1997 (Reference 8). This report includes a map of landslides in the plant site area, and a slope stability analysis of the natural hillslope and cutslope between the power plant and the ISFSI.

2.6.1.2 Detailed ISFSI Study Area Investigations

Additional detailed geologic, seismic, and geotechnical studies were performed in the ISFSI study area. References 2 and 3 further describe the method, technical approach, and results of the detailed studies. The following field, office, and laboratory investigations were performed:

Activity	Documented in
Interpretation of 1968 aerial photography, by PG&E Geosciences Department (Geosciences) and William Lettis & Associates, Inc. (WLA)	Reference 37
Evaluation of previous geologic investigations in the power plant area, including borings by HML and others, by Geosciences and WLA	Reference 37
Detailed geologic mapping of structures and lithology, by Geosciences and WLA	Reference 37

Activity	Documented in
Analysis of rock mass strength, by Geosciences and WLA	Reference 37
Evaluation of rock fractures, by Geosciences and WLA	Reference 37
Analysis of potential rock slope stability, by Geosciences, WLA, and Geomatrix Consultants	References 38 thru 42
Geologic mapping of the power plant site and ISFSI study area, by WLA	Reference 43, Data Report A
Drilling and logging of 17 exploratory diamond-core borings at 14 locations at and near the ISFSI and CTF sites, by WLA	Reference 44, Data Report B
Implementation of two surface seismic refraction lines to measure compressional wave and shear wave velocities of shallow bedrock across the ISFSI site, by GeoVision	Reference 45, Data Report C
Suspension logging of compression and shear wave velocities from boreholes 98BA-1, -3 and -4, by GeoVision	Reference 45, Data Report C
Excavation and logging of 22 exploratory trenches at 14 locations to expose bedrock structures and lithology at the ISFSI site, by WLA	Reference 46, Data Report D
Natural gamma and caliper logging of borings 00BA-1 and -2, and optical televiewer imaging of all borings drilled in 2000 and 2001, by NORCAL Geophysical Consultants	Reference 47, Data Report E
Compilation of discontinuity data, by WLA	Reference 48, Data Report F
Soil testing of clay beds, by Cooper Testing Laboratories	Reference 49, Data Report G
Characterization of rock mass strength, by WLA	Reference 50, Data Report H
Rock strength testing of representative core samples, by GeoTest Unlimited	Reference 51, Data Report I
Petrographic analyses and x-ray diffraction testing of rock samples, by Spectrum Petrographics, Inc.	Reference 52, Data Report J
X-ray diffraction testing and petrographic analysis of clay beds, by Schwein/Christensen Laboratories, Inc.	Reference 53, Data Report K

2.6.1.3 General Description of the ISFSI Study Area

The location and topography of the ISFSI and CTF sites and the transport route are shown in Figures 2.6-1, 2.6-2, and 2.6-3. Detailed investigations of the seismotectonic setting performed for the LTSP (Reference 6, Chapter 2) show that the plant site area lies along the active tectonic boundary between the Pacific and North American plates in coastal Central California. It is located within the San Andreas fault system, about 48 miles west of the main San Andreas fault, and about 3 miles east of the offshore Hosgri fault zone. Current tectonic activity in the region is dominated by active strike-slip faulting along the Hosgri fault zone, and reverse faulting within the Los Osos/Santa Maria domain. The plant site area is on a structural subblock of the San Luis Range (the Irish Hills subblock, Figure 2.6-4), bordered on the northeast and southwest by the Los Osos and southwestern boundary zone reverse faults, respectively, and on the west by the Hosgri fault zone (Reference 6, Chapter 2, Figure 2-50). Since the end of the early Quaternary, the Irish Hills subblock has been slowly elevated along these bounding faults. Detailed mapping and paleoseismic investigations performed for the LTSP (Reference 6) and the DCPP FSAR Update, Section 2.5.1 show that no capable faults are present within the plant site area.

Within the Irish Hills structural subblock, the principal geologic structure is the northwest-trending Pismo syncline (termed the San Luis-Pismo syncline in the DCPP FSAR Update, Section 2.5.1.1.5.2). This 20-mile-long regional structure deforms rocks of the Miocene Monterey and Obispo formations, and the Pliocene Pismo Formation. Fold deformation occurred primarily during the Pliocene, and ceased sometime in the late Pliocene or early Quaternary. Detailed mapping of Quaternary marine terraces across the axis and flanks of the syncline during the LTSP (Reference 6, Plates 10 and 12) documents the absence of fold deformation and associated faulting within the Irish Hills structural subblock for at least the past 500,000 to 1,000,000 years (Reference 6, page 2-34).

The plant site area is situated on the eroded southwestern limb of the Pismo syncline (Figure 2.6-4), within Miocene bedrock of the Obispo Formation (Figure 2.6-5). This regional structure has subsidiary folds that are hundreds to 10,000 ft long and hundreds of feet wide (DCPP FSAR Update, Section 2.5.1.1.5.2, p. 2.5-19, -20). One of these structures, a small northwest-trending syncline, is located directly northeast of the power block (DCPP FSAR Update, Section 2.5.1.2.4.2, p. 2.5-32, -33, Figure 2.5-8). This is the same small syncline that extends across the western part of the ISFSI site (Figures 2.6-5, 2.6-6, and 2.6-7).

Along the coast, the Obispo and Monterey formations have been eroded and incised by former high stands of sea level, leaving a preserved sequence of marine terraces and terrace remnants (Figures 2.6-2 and 2.6-7). The foundation for the power block was excavated into rock below the lower two marine terraces, which are approximately 80,000 and 120,000 years old, respectively (Reference 6, Chapter 2). The power block is founded on competent sandstone and siltstone of the Obispo Formation, the same stratigraphic unit that underlies the ISFSI site.

The ISFSI will be on a prominent ridge directly south of the raw water reservoir and east of the DCPP (Figures 2.6-7 and 2.6-8). The ridge area was used formerly as a borrow source to derive fill material for construction of the 230-kV and 500-kV switchyards. The borrow excavation, completed in 1971, removed up to 100 ft of material from the ISFSI site area and extended deep into bedrock (Figures 2.6-2, 2.6-3, and 2.6-9 through 2.6-12). As a result, the ISFSI and CTF facilities will be founded on bedrock, and the foundation stability and seismic response will be controlled by the bedrock properties. The borrow area cutslope is 900 by 600 ft in plan view, and 300 ft high. The slope of the cut face varies between 2.5:1 and 4:1 (22 to 14 degrees). The former borrow activity at the site stripped surficial soil and weathered rock from the hillside above the ISFSI site, leaving a bedrock slope covered with a veneer of rock rubble. The proposed cutslopes south of the ISFSI pads will be cut entirely in bedrock.

The ISFSI site will be accessed via the transport route, which will follow existing paved roads, except where the road is routed inland from Patton Cove. The transport route starts at the power block, and ends at the ISFSI (Figures 2.6-1, 2.6-3, and 2.6-7).

2.6.1.4 Stratigraphy

2.6.1.4.1 Plant Site Area Stratigraphy

The plant site area is underlain by bedrock of the early and middle Miocene Obispo Formation, and middle Miocene diabase intrusions (References 10 and 6, Chapter 2). Geologic studies for the original DCPP FSAR Update classified bedrock at the power plant site as strata from the middle and late Miocene Monterey Formation. Subsequent studies published by the U.S. Geological Survey (Reference 10), and conducted during the LTSP (Reference 6, Chapter 2) and this ISFSI study reclassified most of the bedrock in the plant site area as part of the Obispo Formation.

Hall and others (Reference 10) divided the Obispo Formation into two members: a fine-grained, massively bedded, resistant zeolitized tuff (mapped as Tor), and a thick sequence of interbedded marine sandstone, siltstone, and dolomite (mapped as Tof) (Figures 2.6-6 and 2.6-7). During the current geologic investigations, the marine sedimentary deposits were further divided into three units, a, b, and c, based on distinct changes in lithology. Unit Tof_a occurs in the eastern part of the plant site area (entirely east of the ISFSI study area) and consists primarily of thick to massively bedded diatomaceous siltstone and tuffaceous sandstone. Unit Tof_b occurs in the central and west-central part of the plant site area, including the entire ISFSI study area and beneath the power block, and consists primarily of medium to thickly bedded dolomite, dolomitic siltstone, dolomitic sandstone, and sandstone. Unit Tof_c occurs in the western part of the plant site area and consists of thin to medium bedded, extensively sheared shale, claystone and siltstone.

Diabase and gabbro sills and dikes intrude the Obispo Formation in the plant site area. These intrusive rocks originally were mapped as a member of the Obispo Formation (Tod) by Hall (Reference 11), but later were reclassified as a separate volcanic formation (Tvr) by Hall and others (Reference 10), because the rocks intrude several different formations, and are not confined to the Obispo Formation. The nomenclature of Hall and others (Reference 10) has been adopted for this study, and these rocks are mapped as Tertiary volcanic rock (Tvr) in the plant site area. These intrusive rocks are well exposed in the north wall of Diablo Canyon, across from the ISFSI site (Reference 11). The diabase typically is a dark, highly weathered, low-hardness rock. It is altered and weak, has a fine crystalline structure, and weathers spheroidally. Petrographic analysis of hand samples shows the diabase is an altered cataclastic gabbro and diorite. The large diabase sill that intruded between dolomite and sandstone beds in the raw water reservoir area was entirely removed during borrow area excavation (Figures 2.6-10 and 2.6-11). There are no exposures of diabase remaining on the borrow area cutslope and no diabase was encountered in any of the boreholes or trenches excavated at the ISFSI study area. Deeper parts of the original intrusion are still exposed in the roadcut along Tribar Road east of and below the raw water reservoir (Figure 2.6-6).

Quaternary deposits generally cover bedrock within the plant site area (Figure 2.6-7). These unconsolidated sediments are discussed in Section 2.6.1.5.

2.6.1.4.2 ISFSI Study Area Stratigraphy

The ISFSI is sited on folded and faulted marine strata of unit Tof_b of the Obispo Formation (Figures 2.6-7 and 2.6-8). Unit Tof_b in the ISFSI study area has undergone a complex history of deposition, alteration, and deformation. Understanding the complexity of the geology and the various geologic processes giving rise to the current geologic conditions at the site is important for interpreting the stratigraphy and structural geology at the site. Based on analysis of surface and subsurface data, supplemented by petrographic analyses of rock lithology, mineralogy, and depositional history, the following events produced the current lithology and stratigraphic character of bedrock at the site (Figures 2.6-13 and 2.6-14). A detailed description of each of these processes is presented in Reference 37.

(1) Original marine deposition, including vertical and lateral facies changes within the dolomite and sandstone sequence

(2) Burial and lithification, followed by diagenesis and dolomitization

(3) Localized addition of petroliferous fluids

(4) Diabase intrusion, hydrothermal alteration, and associated deformation

(5) Tectonic deformation (folding and faulting)

(6) Surface erosion and weathering (both chemical and mechanical)

(7) Borrow excavation and stress unloading

Across the ISFSI site, unit Tof_b is significantly influenced by a lateral and vertical facies change from dolomite to sandstone. In the ISFSI site area, therefore, unit Tof_b has been further divided into a dolomite unit (Tof_{b-1}) and a sandstone unit (Tof_{b-2}). Figure 2.6-15 provides a generalized stratigraphic column illustrating the distribution of rock types within these two subunits. Unit Tof_{b-1} consists primarily of dolomite, dolomitic siltstone, fine-grained dolomitic sandstone, and limestone. Unit Tof_{b-2} consists primarily of fine- to medium-grained dolomitic sandstone and sandstone. Thin clay beds also are present in both units. The dolomite appears to be a diagenetic product of alteration from a limestone and/or calcareous siltstone and very fine sandstone parent rock. Primary deposition of limestone ($CaCO_3$) or calcareous siltstone occurred in a shallow to moderately deep marine environment. Following burial and lithification, the replacement of calcium by magnesium (dolomitization) during diagenesis of the limestone or siltstone formed dolomite ($CaMg(CO_3)_2$).

The contact between the dolomite and sandstone (units Tof_{b-1} and Tof_{b-2}) marks a facies change from a deep marine dolomite sequence to a sandstone turbidite sequence. The contact varies from sharp to gradational, and bedding from one unit locally interfingers with bedding of the other unit. For purposes of mapping, the contact was arbitrarily defined as the first occurrence (proceeding down-section) of medium- to coarse-grained dolomitic sandstone below the dolomite. Surface and subsurface geologic data were used to construct 12 cross sections across the site and transport route (Reference 37). The interfingering nature of the dolomite/sandstone contact beneath the ISFSI study area is illustrated on cross sections A-A', B-B''', D-D', F-F', I-I', and L-L' (Figures 2.6-10, 2.6-11, 2.6-16, 2.6-17, 2.6-18, and 2.6-19, respectively). Some of the thin, interfingering beds provide direct evidence for the lateral continuity and geometry (attitude) of bedding within the hillslope (for example, between boring 01-F and 00BA-1 on section I-I').

Analysis of the cross sections shows that the facies contact between the dolomite and sandstone (units Tofb-1 and Tofb-2) generally extends from northwest to southeast across the ISFSI study area, with sandstone of unit Tofb-2 primarily in the north and northeast part of the area, and dolomite of unit Tofb-1 primarily in the south and southwest part of the area. The three-dimensional distribution of the facies contact is well illustrated by comparing cross sections B-B''' and I-I' (Figures 2.6-11–and 2.6-18). This distribution of the two units reflects a cyclic transgressive/regressive/transgressive marine sequence during the Miocene.

The division of unit Tof_b into two subunits also allows for a detailed interpretation of the geologic structure (folds and faults) in the ISFSI study area. This understanding provides the basis for evaluating the distribution of rock types in the area, and for selecting appropriate rock properties for foundation design and slope stability analyses at the ISFSI, as discussed in Sections 2.6.1.7, 2.6.1.8, 2.6.4.2, and 2.6.5.

2.6.1.4.2.1 Dolomite (Unit Tof_{b-1})

The slope above the ISFSI, including most of the borrow area excavation slope, is underlain by dolomite (Figure 2.6-8). The dolomite is exposed as scattered outcrops across the excavated slope, along the unpaved tower access road (Reference 43, Data Report A), in the upper part of most borings in the ISFSI study area (Reference 44, Data Report B), and in most exploratory trenches (Reference 46, Data Report D). The dolomite consists predominately of tan to yellowish-brown, competent, well-bedded dolomite, with subordinate dolomitic siltstone to fine-grained dolomitic sandstone, and limestone (Figure 2.6-20). Petrographic analyses of hand and core samples from, and adjacent to, the ISFSI study area show that the rock consists primarily of clayey dolomite, altered clayey carbonate and altered calcareous claystone, with lesser amounts of clayey fossiliferous, bioclastic and brecciated limestone, fossiliferous dolomite, and friable sandstone and siltstone (Reference 52, Data Report J, Tables J-1 and J-2). As described in the petrographic analysis, the carbonate component of these rocks is primarily dolomite; thus the general term dolomite and dolomitic sandstone is used to describe the rock.

The dolomite crops out on the excavated borrow area slope as flat to slightly undulating rock surfaces. The rock is moderately hard to hard, and typically medium strong to brittle, with locally well defined bedding that ranges between several inches to 10 ft thick in surface exposures and boreholes. Bedding planes are laterally continuous for several tens of feet, as observed in outcrops, and may extend for hundreds of feet based on the interpreted marine depositional environment. The bedding planes are generally tight and bonded. Unbonded bedding parting surfaces are rare and generally limited to less than several tens of feet, based on outcrop exposures.

2.6.1.4.2.2 Sandstone (Unit Tof_{b-2})

Sandstone of unit Tof_{b-2} generally underlies the ISFSI study area below about elevation 330 ft (Figure 2.6-8). Typically, the rocks in this subunit are well-cemented, hard sandstone and dolomitic sandstone, and lesser dolomite beds.

The well-cemented sandstone encountered in the borings and trenches is tan to gray, moderately to thickly bedded, and competent (Figure 2.6-21). The rock is well sorted, fine- to coarse-grained, and is typically well cemented with dolomite. Petrographic analyses show that the sandstone is altered, and that its composition varies from arkosic to arenitic, with individual grains consisting of quartz, feldspar, dolomite, and volcanic rock fragments (Reference 52, Data Report J). The rock is of low to medium hardness, is moderately to well cemented, and is medium strong. The matrix of some samples contains a significant percentage of carbonate and calcareous silt to clay matrix (probably from alteration). Petrographic analyses show that the carbonate is primarily dolomite. Thus, these rocks are referred to as sandstone and dolomitic sandstone. Bedding in places is well defined, and bedding plane contacts are tight and well bonded. Similar to the dolomite beds, unbonded bedding surfaces within the sandstone are rare and generally limited to less than several tens of feet, based on limited outcrop exposure.

2.6.1.4.2.3 Friable Bedrock

Distinct zones of friable bedrock are present within the generally more cemented sandstone and dolomite (Figures 2.6-8, 2.6-15, and 2.6-22). In some cases, the friable bedrock appears to reflect the original deposit, with no subsequent dolomitization. In other cases, the friable bedrock appears to be related to subsequent chemical weathering or hydrothermal alteration. The friable beds within units Tof_{b-1} and Tof_{b-2} have been designated with the subscript (a). Unit Tof_{b-1a} consists primarily of altered or weathered dolomite or dolomitic siltstone that has a block-in-matrix friable consistency, or simply a silt and clay matrix with friable consistency. The friable rock is of low hardness and is very weak to weak. Unit Tof_{b-2a} consists primarily of friable sandstone, is of low hardness, and is very weak to weak. In many cases, the friable sandstone is the original sandstone that has been chemically weathered or altered to a clayey sand (plagioclase and lithics altered to clay). In other cases, the friable sandstone simply lacks dolomite cementation and retains its original friable nature.

The vertical thickness of the friable rock encountered in borings ranges from less than 1 ft to 32 ft. The friable zones extend laterally for tens of feet in trench exposures, and up to about 200 ft were assumed to correlate between borings (Reference 37). As illustrated on cross section I-I' (Figure 2.6-18), the zones of friable rock are more common, and possibly more laterally continuous, in the sandstone than in the dolomite.

2.6.1.4.2.4 Clay Beds

Clay beds are present within both the sandstone and dolomite units. Clay beds were observed in several trenches (Figures 2.6-23 and 2.6-24) and in many of the borings (Figures 2.6-25 and 2.6-26). Because clay beds are potential layers of weakness in the hillslope above the ISFSI site, they were investigated in detail, (Reference 37). The clay beds generally are bedding-parallel, and commonly range in thickness from thin partings (less than 1/16 inch thick) to beds 2 to 4 inches thick; the maximum thickness encountered was about 8.5 inches. The clay beds are yellow-brown, orange-brown, and dark brown, sandy and silty, and stiff to hard. Petrographic analyses show that the clay contains marine microfossils and small rock inclusions; the rock inclusions are angular pieces of dolomite that are matrix-supported, and have no preferred orientation or shear fabric (Reference 53, Data Report K). In the trenches, the clay beds locally have slickensides and polished surfaces. The clay beds typically are overconsolidated (due to original burial), as supported by laboratory test data (Reference 49, Data Report G), and, where thick, have a blocky structure.

The clay beds encountered in the borings were recorded on the boring logs (Reference 44, Data Report B). In addition, in most of the borings, the clay beds also were documented in situ by a borehole televiewer. The televiewer logs show that the clay beds generally are in tight contact with the bounding rock, and are bedding-parallel

(Reference 47, Data Report E). The clay beds range from massive having no preferred shear fabric, to laminated having clear shear fabric. The shear fabric is interpreted to be the result of tectonic shearing during folding and flexural slip of the bedding surfaces. The shear fabric does not reflect gravitational sliding, because features indicative of sliding, such as disarticulation of the rock mass, tensional fissures, and geomorphic expression of a landslide on pre-construction aerial photographs, are not present.

Clay Beds in Dolomite

Clay beds are more common, thicker, and more laterally continuous in the dolomite (unit Tof_{b-1}). Examination of the continuity of clay beds within and between adjacent trenches, roadcuts, and borings provided data on the lateral continuity (persistence) of the clay beds (Reference 37). Individual clay beds exposed in the trenches and roadcuts appear to be persistent over distances of between tens of feet to more than 160 ft, extending beyond the length of the exposures. The exposed clay beds are wavy and have significant variations in thickness along the bed. Thinner clay beds (less than about 1/4 inch thick) typically contain areas where asperities on the surfaces of the bounding adjacent hard rock project through or into the thin clay. The bedding surfaces are all irregular and undulating, with the height (amplitude) of the undulation greater than the thickness of the clay beds, such that the clay beds likely will have rock-to-rock contact locally during potential sliding, producing an overall increase in the average shear strength of the clay bed surface. A correlation of clay beds within the slope above the ISFSI site is shown on cross section I-I' (Figure 2.6-18). These correlations indicate that at least some clay beds extend over several hundred feet into the hillslope. However, some beds clearly do not correlate: for example, the clay beds exposed in trenches T-14 and T-15 are not found in nearby boring 01-I.

Clay Beds in Sandstone

Clay beds are less common, generally thinner, and less laterally continuous in the sandstone (unit Tof_{b-2}). Clay beds observed in the sandstone generally are less than 1/4 inch thick. These thinner clay beds are difficult to correlate laterally between borings and, at least locally, are less than 50 to 100 ft in lateral extent. For example, as shown on cross sections B-B' and I-I' (Figures 2.6-11 and 2.6-18), clay beds were not encountered in boring 01-B, but were encountered in borings 01-A and 01-H, 50 to 100 ft away.

Clay Moisture Content

The clay beds encountered in the borings and trench excavations in both the dolomite and sandstone were moist. Clay beds uncovered in the trenches dried out after exposure during the dry season, and became hard and desiccated. When wetted during the rainy season, the clay in the trenches became soft and sticky (Reference 46, Data Report D, Trench T-11). Possible local perched water tables, as observed in boring 01-F and evident elsewhere in the plant site area (Section 2.5, Subsurface

Hydrology), also may soften the upper portions of the clay beds during the rainy season in the ISFSI study area.

Clay Composition

X-ray diffraction analyses (Reference 53, Data Report K) show that the clay-size fraction of the clay beds consists of three primary minerals: kaolinite (a clay), ganophyllite (a zeolite), and sepiolite (a clay). The silt-size fraction of the sample consists primarily of rock and mineral fragments of quartz, dolomite/ankerite, and calcite. Petrographic examination of the clay (Reference 53, Data Report K) shows a clay matrix having matrix-supported angular rock fragments and no shear fabric. Included rock fragments have evidence of secondary dolomitization of original calcite (limestone), and localized post-depositional contact alteration. Some samples contain microfossils (benthic foraminifera). The ganophyllite minerals appear to be expansive, as evidenced by swelling of one sample (X-1 from trench T-14A) after thin-section mounting. Sample X-2 also had a significant percentage of ganophyllite, and a high plasticity index (PI) of 63 (References 49 and 53, Data Reports G & K).

The presence of microfossils confirms the clay is depositional in origin, and was not formed by alteration or weathering of a lithified host rock. The clay is interpreted to reflect pelagic deposition in a marine environment.

2.6.1.4.2.5 Diabase (Tvr)

Diabase is exposed in the roadcut along Tribar Road and probably underlies the eastern portion of the raw water reservoir area. The diabase is part of the Miocene diabase intrusive complex in Diablo Canyon near the switchyards (Reference 10). A large diabase body was removed during grading for the raw water reservoir pad and the borrow cut area. This body of diabase likely was continuous with the diabase exposed along Tribar Road. Currently, no diabase is exposed on the borrow cut slope, and diabase was not encountered in any of the borings or trenches in the ISFSI study area. The diabase exposed along Tribar Road has been altered to a friable rock, and is soft to dense and easily picked apart; it is judged to be similar in engineering properties to the friable sandstone and friable dolomite found in the ISFSI study area. Though diabase was not encountered elsewhere in the ISFSI study area during field investigations, it is possible that other small dikes or sills of diabase may be encountered during excavation for the ISFSI pads or cutslope.

2.6.1.5 Geomorphology and Quaternary Geology

The geomorphology and Quaternary geology of the plant site area is dominated by a flight of coastal marine terraces, deep fluvial incision along Diablo Creek, and deposition of alluvial and colluvial fans at the base of hillslopes. Quaternary deposits cover bedrock across most of the power plant property, except in the ISFSI study area, where extensive borrow excavation in the 1970s removed the Quaternary deposits. These deposits accumulated in distinctive geomorphic landforms that include coastal marine terrace platforms, debris and colluvial fans at the base of hills and swales, landslides on hillslopes and sea cliffs, and alluvium along the floor of Diablo Canyon. The distribution of Quaternary deposits and landforms are shown on Figures 2.6-7 and 2.6-8.

2.6.1.5.1 Marine Terraces

Several marine terraces form broad coastal platforms within the western part of the power plant property (Figure 2.6-7). The power plant and associated support facilities and buildings are constructed on these terraces (Figure 2.6-2). Discontinuous remnants of older and higher terraces also are present locally across the ISFSI study area. Each of these marine terraces consists of a relatively flat, wave-cut bedrock platform, a thin layer of marine sand and cobble sediments, and surficial deposits of colluvium, alluvium, and eolian sediments. The "staircase" of bedrock platforms resulted from a combination of regional uplift, sea level fluctuations, and wave erosion.

The locations and elevations of marine terraces along the coast from Avila Beach to Montaña de Oro and Morro Bay, including the area of the power plant, were initially characterized during studies for PG&E's Long Term Seismic Program (Reference 6). Several terraces were mapped in more detail for the ISFSI studies, and the location of the inner edge (or shoreline angle) of the terraces was estimated (Figure 2.6-7). Well-developed, wave-cut bedrock platforms and their associated terraces exist in the plant site area at elevations of about 30 to 35 ft (Q_1 terrace), 100 to 105 ft (Q_2 terrace), and 140 to 150 ft (Q_3 terrace), and form relatively level bedrock surfaces under the surficial Quaternary deposits along the coast. The platforms slope gently seaward at angles from 2 degrees to 3 degrees, and are bordered landward by steep (50 degrees to 60 degrees, Reference 6) former sea cliffs that are now largely covered by thick surficial deposits. A sequence of Pleistocene to Holocene colluvial fans covers the landward portion of the coastal terraces. These deposits consist of crudely bedded clay, clayey gravel, and sandy clay, and have distinct paleosol and carbonate horizons. The lower, Pleistocene fan deposits are very stiff and partly consolidated; they have highly weathered clasts, carbonate horizons, and an oxidized appearance. The upper, Holocene deposits are unconsolidated and have a higher organic contact; they do not have argillic or carbonate horizons.

Near the ISFSI site, discontinuous remnants of a higher marine terrace are present. The terrace has an approximate shoreline angle elevation of 290 ft (Q_5 terrace) (Figure 2.6-7). The terrace deposits consist of a basal layer of marine sand and gravel overlain by colluvial sandy clay and clayey gravel. This terrace may be coeval with an estuarine deposit of black clay having interfingering white shell hash that crops out beneath the edge of the 500-kV switchyard fill (Figure 2.6-7). The clay appears to have been deposited in an estuarine environment by an ancient marine embayment into Diablo Canyon. Most of the Q_5 terrace, however, has been eroded by incision along Diablo Creek, or is buried by younger stream terrace and landslide deposits, or switchyard and road fills.

The thickness of the terrace deposits (depth to bedrock) varies greatly, from less than 10 ft to greater than 80 ft. Extensive grading for the DCPP and related facilities and parking areas have substantially modified the morphology and thickness of terrace deposits in some locations. The current thickness of terrace deposits, therefore, is locally dependent on site-specific grading activities.

2.6.1.5.2 Inland Quaternary Deposits

Diablo Creek has carved a deep channel into bedrock, causing oversteepening of the slopes along the canyon walls. Some thin, narrow, channel deposits, and one locally preserved stream terrace veneered by colluvial deposits, are present in the canyon. The rate and extent of erosion, however, generally has been dominant over sedimentation in the canyon, and alluvial deposits are relatively thin and of limited extent. Substantial reaches along the lower part of the creek were artificially filled, channeled, and altered during development of the power plant and related facilities, particularly around the 230-kV and 500-kV switchyards, which are constructed on large fill pads across the bottom of the canyon.

Slopes in the Irish Hills are extensively modified by mass wasting processes, including landslides, debris flows, creep, gully and stream erosion, and sheet wash. Extensive grading to form level platforms for the power plant and related facilities along the back edge of the coastal terraces has greatly modified the lower portions of most slopes in the plant site area. Large, deep-seated landslide complexes are present on the slopes of Diablo Canyon south of the 230-kV and 500-kV switchyards (Figure 2.6-7). These features consist of large (exceeding 100 acres), deep-seated, coalescing, bedrock landslides. The dip of bedrock strata in the vicinity of these large slides is downslope, suggesting the failure planes for these slides probably occurred within the bedrock along clay beds and bedding contacts. Some slides may have occurred at the contact between bedrock and overlying weathered bedrock and colluvium, or along contacts between Obispo Formation bedrock and relatively weaker diabase.

The large landslide complexes have been considerably modified by erosion, and fluvial terraces and possible remnants of the Q_5 marine terrace appear to have been cut into the toes of some of the slides. These conditions suggest they are old features that likely formed prior to the Pleistocene-Holocene transition, during a wetter climate. These large slide complexes, therefore, appear to have a stable configuration under the present climatic conditions, which have persisted during the Holocene (past 10,000 years or so).

Debris-flow scars and deposits are found along some of the steeper slopes (Figure 2.6-7). The debris flows originate where colluvium collects in topographic swales or gullies on the upper and middle slopes. Debris flows usually are triggered by periods of severe weather that allow development of perched groundwater within hillside colluvial deposits. Following initial failure, the saturated mass flows rapidly down drainage channels, commonly scouring the bottom of the channel and increasing in volume as it travels downslope. The flow stops and leaves a deposit of poorly sorted debris at a point where the slope angle decreases. Debris fans formed by accumulation of successive debris flows are present at the mouths of the larger canyons and gullies in the area (Figure 2.6-7).

2.6.1.6 Structure

2.6.1.6.1 Regional Structure

Bedrock structure in the plant site region is dominated by the northwest-trending Pismo syncline (Figure 2.6-4), which forms the core of the Irish Hills (References 10 and 11). The regional bedrock structure and tectonic setting are described in the DCPP FSAR Update, Section 2.5.1.1 (Reference 4), and LTSP Final Report, Chapter 2 (Reference 6), and are summarized in Section 2.6.2 of this report. The following sections describe the structural setting of the ISFSI study area, including the distribution of bedrock folds, faults, and joints in the area.

2.6.1.6.2 ISFSI Study Area Structure

Bedrock in the ISFSI study area has been deformed by tectonic processes and possibly by the intrusion of diabase. The detailed stratigraphic framework described above provides the basis for analyzing the geologic structure in the site area.

Geologic structures in the ISFSI study area include folds, faults, and joints and fractures. The distribution and geometry of these structures is important for evaluating rock mass conditions and slope stability because: (1) folds in the bedrock produce the inclination of bedding that is important for evaluating the potential for out-of-slope, bedding-plane slope failures; and (2) faults and, to a lesser extent, joints in the bedrock produce laterally continuous rock discontinuities along which potential rock failures may detach in the proposed cutslopes.

The distribution and geometry of folds and faults in the bedrock were evaluated through detailed surface geologic mapping, trenches, and borings (References 2 and 3). Data from these studies were integrated to produce geologic maps (Figures 2.6-6, 2.6-7, and 2.6-8) and geologic cross sections (for example, Figures 2.6-10, 2.6-11, and 2.6-16 through 2.6-19). The cross sections were prepared at various orientations to evaluate the three-dimensional distribution of structures. Bedding attitudes were obtained from surface mapping (including roadcut and trench exposures) and from boreholes (based on visual inspection of rock core integrated with oriented televiewer data). These bedding attitudes were used to constrain the distribution of bedrock lithologies and geometry of bedding shown on the cross sections.

2.6.1.6.2.1 Folds

Similar to the power plant, the ISFSI is located on the southwestern limb of the regional Pismo syncline (Figure 2.6-4). As shown on the geologic maps (Figures 2.6-6, 2.6-7, and 2.6-8) and cross sections (Figures 2.6-10, 2.6-11, and 2.6-16 through 2.6-19), bedrock in the ISFSI study area is deformed into a small, northwest-trending syncline and anticline along the western limb of the larger regional Pismo syncline. On the ridge southeast of the ISFSI study area, nearly continuous outcrops of resistant beds expose the small anticline and an en echelon syncline (Figures 2.6-6, 2.6-7, and 2.6-17). These folds are relatively tight and sharp-crested, have steep limbs, and plunge to the northwest.

Within the ISFSI study area, the northwest-plunging anticline appears to be the northwestward continuation of the anticline that is exposed in the ridge top at the Skyview Road overlook area (Figure 2.6-1). The anticline varies from a tight chevron fold southeast of the ISFSI study area, to a very broad-crested open fold across the central part of the area. The northwestward shallowing of dips along the anticlinal trend appears to reflect a flattening of fold limbs up-section. In the ISFSI study area, the broad crest of the fold is disrupted by a series of fold-parallel, minor faults (Figure 2.6-11). The minor faults displace the fold axis, as well as produce local drag folding, which tends to disrupt and complicate the fold geometry. The axis of this broad-crested anticline is approximately located on the geologic map (Figure 2.6-8).

The en echelon syncline, at the ridge crest along Skyview Road, projects to the northwest along the southwestern margin of the ISFSI study area. From the southeast to the northwest, the syncline changes into a northwest-trending monocline, and then back into a syncline (Figures 2.6-6 and 2.6-7). In the ISFSI study area, the syncline opens into a broad, gently northwest plunging (generally less than 15 degrees) fold with gently sloping limbs (generally less than 20 degrees). Bedding generally dips downslope to the northwest in the upper part of the slope above the ISFSI site, and perpendicular to the slope to the southwest and west in the lower part of the slope. Small undulations in the bedding reflect the transition from a tight syncline to a relatively flat monocline, or “shoulder,” and then back to a broad, northwest-plunging syncline. These localized interruptions to the northwestern plunge of the fold may be caused by the diabase intrusion and localized doming associated with the intrusion (compare diagrams C and D on Figure 2.6-13).

As discussed above and shown on cross sections B-B''', D-D', and F-F' (Figures 2.6-11, 2.6-16, and 2.6-17), the western limb of the small syncline varies from steeply dipping (approximately 70 degrees northwest) across the southern part of the plant site area, to gently dipping (approximately 30 degrees northwest) beneath the power block. This change in the dip of the syncline across the plant site area mirrors the change in dip described above across the ISFSI study area. Based on the geometry of the syncline, bedrock beneath the power block consists of sandstone (unit Tof_{b-2}), underlain by dolomite (unit Tof_{b-1}) (Figure 2.6-11). The power block is located on the same stratigraphic sequence exposed in the ISFSI study area; however, the sequence is approximately 400 ft lower in the stratigraphic section. As shown on cross section B-B''' (Figure 2.6-11), boreholes drilled during foundation exploration for the power block encountered calcareous siltstone having abundant foraminifera. This description of the rock is very similar to the dolomite of unit Tof_{b-1}; thus, the lower contact between units Tof_{b-1} and Tof_{b-2} is interpreted to be beneath the power block area.

Folding occurred during growth of the northwest-trending, regional Pismo syncline in the Pliocene to early Quaternary (Reference 6). The smaller folds at and near the ISFSI study area are parasitic secondary folds along the southwestern limb of the larger Pismo syncline. Because of their structural association with the Pismo syncline, the folding in the area is interpreted to have occurred during the Pliocene to early Quaternary (Figure 2.6-8). Some localized fold deformation also may have accompanied the earlier Miocene diabase intrusions.

2.6.1.6.2.2 Faults

Numerous minor, bedrock faults occur within the ISFSI study area (Figures 2.6-27 and 2.6-28). Based on displaced lithologic and bedding contacts, most of the faults have vertical separations of a few inches to a few feet. At least five faults show vertical separation of several tens of feet. Slickensides and mullions on the fault surfaces generally show strike-slip to oblique strike-slip displacement.

The faults trend generally northwest, subparallel to the local fold axes (Figure 2.6-29). They dip steeply to near-vertical, generally 70 to 90 degrees, both northeast and southwest. They consist of interconnecting and anastomosing strands, in zones up to 5 ft wide. The faults have documented lengths of tens of feet to a few hundred feet, and are spaced from several tens of feet to hundreds of feet apart across the ISFSI study area, based on trench exposures and surface geologic mapping.

The fault surfaces within bedrock vary from tightly bonded or cemented rock/rock surfaces, to relatively soft slickensided clay/rock and clay film contacts. Individual faults are narrow, ranging in width from less than an inch to about 2 ft. Fault zones contain broken and slickensided rock, intermixed clay and rock, and locally soft, sheared, clayey gouge. The thickness of fault gouge and breccia is variable along the faults.

Cross section B-B''' (Figure 2.6-11) illustrates the subsurface stratigraphy and structure beneath the ISFSI pads. As shown on the map (Figure 2.6-8) and cross section, five minor faults clearly juxtapose dolomite (Tof_{b-1}) against sandstone (Tof_{b-2}), and truncate individual friable beds. Vertical separation across individual faults ranges from about 10 ft to greater than 50 ft, based on displacements of friable beds and the contact between units Tof_{b-1} and Tof_{b-2}. Total vertical separation across the entire fault zone exceeds 50 ft; cumulative displacement is down on the northeast. As described previously, the contact between dolomite and sandstone (units Tof_{b-1} and Tof_{b-2}) beneath the pads is based on the first occurrence of medium to coarse-grained sandstone, and there is no evidence of significant facies interfingering between the two units beneath the pads that would obscure the amount of displacement. Therefore, the interpretation of vertical separation of bedrock along the faults is given a relatively high degree of confidence.

Subhorizontal slickensides indicate that the minor faults in the ISFSI study area have predominantly strike-slip displacement (Figure 2.6-30). Using a typical range of a 10-degree to 20-degree rake on the slickensides and the vertical separation, total fault displacement is estimated to be several tens to several hundreds of feet. The faults trend subparallel to the axis of the Pismo syncline, and trend approximately 35 to 55 degrees more westward than the offshore Hosgri fault zone (Figure 2.6-29).

The faults in the ISFSI study area may be continuous with several other minor faults having similar characteristics exposed along strike in dolomite in the Diablo Creek roadcut about 800 ft to the north (Figures 2.6-6, 2.6-7, and 2.6-30). Given this correlation and the presence of several hundred feet of strike-slip displacement, the faults may be at least several thousand feet long. Interpretation of pre-borrow excavation aerial photography shows that the faults are not geomorphically expressed in the ISFSI study area (Figure 2.6-31) and there is no evidence of displaced Quaternary deposits along the fault traces.

In the analysis of slope stability (Section 2.6.5), the faults are assumed to form high-angle parting surfaces along the lateral margins of potential rock slides, rock wedges, and topple blocks. Fault-bounded structural blocks are shown on Figure 2.6-8, and on cross section B-B''' (Figure 2.6-11). The age and noncapability of the faults are discussed in Section 2.6.3.

2.6.1.6.2.3 Bedrock Discontinuities

Extensive data on bedrock discontinuities were collected from the borings and trenches within the ISFSI study area to assess their orientation, intensity, and spatial variability (Reference 48, Data Report F). The discontinuity data were used in the failure analysis of the ISFSI cutslopes (Section 2.6.5). Bedrock discontinuities include joints, faults, bedding, and fractures of unknown origin. These discontinuities, in particular joints, are pervasive throughout bedrock in the ISFSI study area (Figure 2.6-20). Steeply dipping faults and joint sets are the dominant discontinuities, giving the rock mass a subvertical fabric. Random and poorly developed low-angle joints also occurs subparallel to bedding. The fault discontinuities are described in Section 2.6.1.6.2.2. Joint discontinuities are described below.

Joint contacts vary from tight to partially tight to slightly open; joint surfaces are slightly smooth to rough, and have thin iron oxide or manganese coatings (Reference 50, Data Report H). Joint lengths in trenches and outcrops typically range from a few feet to about 20 ft, and typical joint spacing range from about 6 inches to 4 ft, with an observed maximum spacing of about 14 ft (Reference 48, Data Report F, Table F-6). The intersections of various joints, faults, and bedding divide the bedrock into blocks generally 2 ft to 3 ft in dimension, up to a maximum of about 14 ft. Rock blocks formed by intersecting joints larger than those described above generally are keyed into the rock mass by intact rock bridges or asperity interlocking. The largest expected “free” block in the rock mass is, therefore, estimated to be on the order of about 14 ft in maximum dimension.

Both the well-cemented sandstone and the dolomite contain numerous joints. The jointing typically is confined to individual beds or groups of beds, giving the bedrock a blocky appearance in outcrop. Joints are less well developed and less common in the friable sandstone and friable dolomite. Linear zones of discoloration in the friable rock may represent former joints and small faults, but these zones are partially re-cemented, and not as frequent or obvious as joints in the harder rock.

The character of joints also differs between the upper, dilated zone of bedrock (generally within the upper 4 ft in the ISFSI study area, but conservatively estimated to extend to a maximum of 20 ft deep, particularly toward the edges of the old borrow cut where the amount of rock removed in 1971 is minimal) and the underlying zone of “tight” bedrock. Joints are generally tight to open in the upper zone. In the lower zone, the joints and other structures are tight and, in places, bonded and healed. This is well demonstrated in the borehole optical televiewer logs (Reference 47, Data Report E), which show the joints are typically tight and/or partly bonded throughout the borings. In both zones, the joints are locally clay-filled, and commonly contain thin fillings of clay, calcite, dolomite, and locally, gypsum. Joints and fractures in the borings are very closely to widely spaced (less than 1/16-inch to 3-ft spacing), with local crushed areas between joints.

In general, the joints group into two broad sets: a west- to west-northwest-striking set, and a north-northwest-striking set. In some trenches, fractures from both sets are present, whereas some show a scatter in orientation within a general northwest-southeast orientation. The variation in orientation and density of the joints with both strata and location across the ISFSI study area shows that the joints are limited in continuity.

The general northwest-southeast-trending character of the joints in the ISFSI study area is consistent with both the overall northwest-trending regional structural grain. Local variations in discontinuity orientations and intensity are attributed to rheological differences between dolomite and sandstone and their friable zones, as well as to proximity to the minor faults that cut across the area.

2.6.1.7 Stratigraphy and Structure of the ISFSI Pads Foundation

Figure 2.6-32 illustrates the expected bedrock conditions that will be encountered in the foundation excavation for the ISFSI pads at the assumed pad subgrade elevation of 302 ft (Reference 37). The pads will be founded primarily on dolomitic sandstone of unit Tof_{b-2} and dolomite of unit Tof_{b-1}. Dolomitic sandstone generally underlies most of the site; dolomite underlies the eastern end of the site. The proposed cutslopes above the site are generally underlain by dolomitic sandstone in the western and central parts of the cut, and by dolomite in the upper and eastern parts of the cut.

Locally, friable sandstone (Tof_{b-2a}) and friable dolomite (Tof_{b-1a}) underlie the foundation of the ISFSI pads and the proposed cutslopes (Figure 2.6-32). Because the zones are highly variable in thickness and continuity, their actual distribution likely will vary from that shown on Figure 2.6-32. In particular, a large body of friable dolomite underlies the southeastern portion of the proposed cutslope. Other smaller occurrences of friable sandstone and dolomite probably will be encountered in the excavation. These friable rocks locally have dense, soil-like properties; thus, specific analyses were performed to assess the foundation properties and slope stability of these friable rock zones (Reference 51, Data Report I). Small zones of friable diabase may be found in the excavation, as discussed in Section 2.6.1.4.2.5. This rock has properties similar to the friable sandstone.

In two places beneath the foundation of the ISFSI pads, clay beds within dolomite and sandstone are expected to daylight or occur within 5 ft of the base of the foundation (Figure 2.6-32). Additional clay beds may be exposed in the foundation of the pads. Although available geologic data do not document the presence of clay beds that will daylight in the ISFSI cutslope, some may be encountered when the cuts are made.

In addition, a zone of minor non-capable faults trends northwest across the central and eastern part of the ISFSI pads (Figures 2.6-8 and 2.6-11) (Section 2.6.3). The faults have vertical separations of 10 ft to 30 ft, and locally juxtapose different bedrock units.

2.6.1.8 Stratigraphy and Structure of the CTF Foundation

The CTF site lies about 100 ft directly northwest of the northwest corner of the ISFSI site (Figure 2.6-8). The CTF site is on the same west limb of the small anticline that underlies the ISFSI site (Figure 2.6-8, Section 2.6.1.6.2.1). Borings 00BA-3 and 01-CTF-A show the CTF will be founded on sandstone (unit Tof_{b-2}) and friable sandstone (unit Tof_{b-2a}), similar to the rock at the ISFSI site (Figures 2.6-8 and 2.6-32). The CTF site is located along the northwestern projection of the small bedrock faults at the ISFSI site, and similar faults and joints are expected to be encountered in the excavation for the CTF. Although no clay beds were encountered in borings 00BA-3 and 01-CTF-A, clay beds may underlie the site at deeper elevations (Reference 37). The dip of the bedrock at the CTF site appears to be near-horizontal. In the cutslope west of the CTF site, bedrock dips moderately to the northeast, into the slope (Figure 2.6-7).

2.6.1.9 Stratigraphy and Structure of the Transport Route

The transport route begins at the power block and ends at the ISFSI. The route will follow existing paved Plant View, Shore Cliff, and Reservoir roads (Figure 2.6-1), except where routed north of the intersection of Shore Cliff and Reservoir roads to avoid an existing landslide at Patton Cove. The lower two-thirds of the route traverse thick surficial deposits, including marine terrace, debris-flow, and colluvial deposits of varying thicknesses (Reference 37). These surficial deposits overlie two units of the Obispo Formation bedrock: unit Tof_b sandstone and dolomite, and unit Tof_c thinly to thickly bedded claystone, siltstone, and shale. The upper third of the route is on engineered fill, directly above dolomite and sandstone bedrock (units Tof_{b-1} and Tof_{b-2}) of the Obispo Formation (Figure 2.6-7). Locally, the road is on a cut-and-fill bench cut into the bedrock.

In the geologic description below, approximate stations have been assigned to assist in defining distances between locations, starting from the power block and ending at the ISFSI (Figure 2.6-7). Although not surveyed, this informal stationing is standard engineering format to represent the distance, in feet, from the beginning of the route outside the power block to the station location (for example, 21+00 is 2,100 ft from the beginning). The specific conditions along the route are discussed below.

Station 00+00 (south side of power block) to 20+00 (near Reservoir Road): The transport route generally follows Plant View Road and Shore Cliff Road. The route starts at the power block and crosses flat, graded topography on the lower coastal marine terrace (Q_2) (Figure 2.6-3). Behind the power block, the route is founded on sandstone (Tof_b) of the Obispo Formation. From there to near Reservoir Road, the transport route is founded on surficial deposits 10 to 40 ft thick, and engineered fill in excavations made during construction of the power plant. The surficial deposits consist primarily of debris-flow and colluvial deposits that overlie the marine bedrock terrace platform (Figures 2.6-7 and 2.6-16). These deposits range in age from middle Pleistocene to Holocene, and consist of overconsolidated to normally consolidated clayey sand and gravelly clay. The deposits contain some carbonate cementation and paleosols, and typically are stiff to very stiff (medium dense to dense). Bedrock below the marine terrace platform consists of east-dipping sandstone (Tof_b) from station 00+00 to about 07+00, and steeply dipping claystone and shale (Tof_c) from about 07+00 to 20+00. Because of the thickness of the overburden, bedrock structure will have no effect on the foundation stability of the road.

Station 20+00 to 34+00 (near Shore Cliff Road to Hillside Road): From station 20+00 to 26+00, the transport route will be on a new road north of the intersection of Shore Cliff Road and Reservoir Road to avoid an existing landslide at Patton Cove (Section 2.6.1.12.1.1; Figures 2.6-6, 2.6-7, and 2.6-19). A 5- to 50-ft-thick prism of engineered fill will be placed to achieve elevation from the lower part of the marine terrace to the upper part of the marine terrace as the road U-turns uphill. The engineered fill will overlie over-consolidated to normally-consolidated Pleistocene debris-flow and colluvial deposits 20 to 80 ft thick that cover the marine bedrock platform (Q2), which in turn overlie steeply dipping claystone and shale of unit Tofc below the marine platform.

Along Reservoir Road, the route follows the higher part of this terrace, generally over the marine platforms Q_2 and Q_3. The surficial deposits consist of debris-flow and colluvial deposits up to 80 ft thick along the base of the ridge behind parking lot 8 (Figure 2.6-19). Bedrock below the marine terrace is claystone and shale (Tof_c) from station 26+00 to 29+50, and sandstone (Tof_b) from station 29+50 to 34+00.

Station 34+00 (Reservoir Road at Hillside Road) to 49+00 (along Reservoir Road): The route follows Reservoir Road to the raw water reservoir area. The road traverses the west flank of the ridge on an engineered cut-and-fill bench constructed over unit Tof_b dolomite and sandstone, and thin colluvium and debris-flow fan deposits. Bedding, as exposed in the roadcut, dips 30 to 50 degrees into the hillslope, away from the road. Engineered fill on sandstone and dolomite underlies the inboard edge of the road, and a wedge of engineered fill over colluvium generally underlies the outboard edge of the road (Figures 2.6-7, 2.6-11, and 2.6-19).

Bedrock joints exposed in this part of the route are similar to those at the ISFSI site. Joints are generally of low lateral persistence, confined to individual beds, and are tight to open. Joint-bounded blocks are typically well keyed into the slope, with the exception of a 1- to 3-ft-thick outer dilated zone. No large unstable blocks or adverse structures prone to large-scale sliding were observed.

Station 49+00 (along Reservoir Road) to 53+50 (ISFSI pads): The route leaves the existing Reservoir Road and crosses the power plant overview parking area. The route will be placed on new engineered fill up to 5 ft thick that will overlie thin engineered fill (up to 4 ft thick) that was placed over sandstone and friable sandstone (Tof_{b-2} and Tof_{b-2a}), the same rock types that underlie the ISFSI pads and CTF site.

Bedrock structures beneath this part of the route are inferred to be joints and small faults, similar to those exposed at the ISFSI site (Figure 2.6-8). The faults would trend generally northwest, and dip steeply northeast and southeast, to vertical. The primary joint sets are near-vertical (Section 2.6.1.6.2.3). This part of the road is on flat topography, and bedrock structure will have no effect on the foundation stability of the road.

An expanded description of this section of the transport route is provided in Reference 76 (PG&E Response to NRC Request 5).

2.6.1.10 Comparison of Power Block and ISFSI Bedrock

Bedrock beneath the ISFSI was compared to bedrock beneath the power block based on stratigraphic position, lithology, and shear wave velocity. Based on these three independent lines of evidence, the bedrock beneath the ISFSI and the power block is interpreted to be part of the same stratigraphic sequence, and to have similar bedrock properties and lithology.

Stratigraphic Position

Cross section B-B''' illustrates the stratigraphic correlation of bedrock between the ISFSI site and the power block site (Figure 2.6-11). As shown on the cross section, the power block and ISFSI are located on the same continuous, stratigraphic sequence of sandstone and dolomite of unit Tof_b of the Obispo Formation. As mentioned previously, the sequence at the power block is approximately 400 ft lower in the stratigraphic section.

The bedrock of the same continuous, stratigraphic sequence as that beneath the power block is exposed directly along strike in roadcuts along Reservoir Road (Figure 2.6-2).

The bedrock exposed in the roadcut consists of dolomite, dolomitic siltstone, and dolomitic sandstone of unit Tof_{b-1}.

Lithology

As described in the DCPP FSAR Update, Section 2.5.1.2.5.6, p. 2.5-42, Figures 2.5-9 and 2.5-10) bedrock beneath the power block consists predominantly of sandstone, with subordinate thin- to thick-bedded slightly calcareous siltstone (for examples, see boring descriptions on Figures 2.6-11 and 2.6-19). The rocks are described as thin-bedded to platy and massive, hard to moderately soft and "slightly punky," but firm. These lithologic descriptions are similar to those for the rocks at the ISFSI site, and the rocks are interpreted to be the same lithologies.

The "calcareous siltstone" described in the DCPP FSAR Update is probably dolomite or dolomitic siltstone comparable to unit Tof_{b-1}. For example, based on the geologic descriptions of the rocks, the "siltstone" and "sandstone" encountered in 1977 in power block boring DDH-D is interpreted to be the dolomite and dolomitic sandstone of unit Tof_{b-1} observed at the ISFSI site.

Boring logs from the hillslope between the power block and the ISFSI site, included in the DCPP FSAR Update (Figures 2.5-22 to 2.5-27; Appendix 2.5C, plates A-1 to A-19), describe bedrock as tan and gray silty sandstone and tuffaceous sandstone (Figures 2.6-11 and 2.6-19). These rocks are moderately hard and moderately strong. The rock strata underlying this slope dip into the hillside and correlate with the sandstone and dolomite strata exposed on the west flank of the ridge (and west limb of the syncline) that are exposed in roadcuts along Reservoir Road south of the ISFSI site (Figures 2.6-6, 2.6-7, and 2.6-20) and in the deeper part of the borings at the ISFSI site.

Shear Wave Velocity

Shear wave velocity data from the power block site and the ISFSI site are summarized on Figures 2.6-33 and 2.6-34. Velocity data in Figure 2.6-35 are from borehole surveys at the ISFSI site (Reference 45, Data Report C), and comparative velocities at the power block site are from the DCPP FSAR Update. As evident from the figures, shear-wave velocities from surface refraction and borehole geophysical surveys at the ISFSI site are within the same range as those obtained at the power block site. The velocity profiles at both sites are consistent with a classification of "rock" for purposes of characterizing ground-motions (Reference 12).

2.6.1.11 Groundwater

Refer to Section 2.5, Subsurface Hydrology.

2.6.1.12 Landslides

2.6.1.12.1 Landslide Potential in the Plant Site Area

Slopes in the Irish Hills are subjected to mass-wasting processes, including landslides, debris flows, creep, gully and stream erosion, and sheet wash (Reference 9). Extensive grading in the plant site area to create level platforms for structures along Diablo Canyon and the coastal terraces has modified the lower portions of most of the slopes near the plant site.

Debris-flow scars and deposits occur on, and at the base of, slopes in the plant site area. The debris flows initiate where colluvium collects in topographic swales or gullies, and are usually triggered by periods of severe weather. Debris-flow fans, caused by the accumulation of successive debris flows, form at the mouths of the larger canyons and gullies in the area. Several typical gullies that have colluvium-filled swales, debris-flow chutes, and debris-flow fans at the bottom of the chutes are found on the slope above parking lots 7 and 8, south of the power plant (Figures 2.6-5 and 2.6-7).

During landslide investigations in 1997, PG&E identified a large, (exceeding 100 acres) ancient landslide complex on the slopes of Diablo Canyon, directly east of the 230- and 550-kV switchyards (Reference 9, Figure 2.6-7). The dip of the bedrock in the vicinity of these large slides is downslope, contributing to slope instability (Reference 9, Figure 21). This structure suggests the failure planes for these slides are probably within the bedrock along bedding contacts, clay beds, and possibly along the intrusive contacts between Obispo Formation bedrock and the altered diabase.

The large landslide complex is subdued, and has been considerably modified by erosion. Thin stream-terrace deposits and remnants of the Q_5 430,000-year-old marine terrace at elevation 290 $\pm$5 ft appear to have been cut into the toes of some of the slides. These relations indicate the landslides are old and likely formed in a wetter climate during the middle to late Pleistocene. The landslides appear to be stable under the present climatic conditions. There is no geomorphic evidence of activity in the Holocene (past 10,000 years or so). Additionally, the 500-kV switchyard embankment fill in the canyon provides a partial buttress to the toe of the old landslide deposit, and serves to help stabilize the landslide. The switchyard shows evidence of no post-construction slope movement. The complex lies entirely east of the ISFSI, and does not encroach, undermine, or otherwise affect the ISFSI.

Patton Cove Landslide

The Patton Cove landslide (Figure 2.6-36) is a deep-seated rotational slump located at a small cove adjacent to Shore Cliff Road along the coast, about one-half mile east of the power plant (Figures 2.6-6, 2.6-7, and 2.6-17) (Reference 9, p. 78-83). Shore Cliff Road was constructed on engineered fill benched into marine-terrace and debris-flow fan deposits directly east of the slide. Cracks within Shore Cliff Road suggest that the landslide may be encroaching headward beneath the road. The landslide is approximately 125 ft long, 400 ft wide, and 50 ft deep. The slide occupies nearly the full height of the bluff face, which is inclined about 1.3:1 (H:V).

Slide movement was first documented in 1970 by Harding Lawson Associates (HLA) (Reference 13). In 1970, the head scarp of the slide was approximately 15 ft south of the toe of the fill that supports Shore Cliff Road. In the 31 years since slide movement was first documented, the slide mass has been episodically reactivated by heavy rains and continued wave erosion at the toe of the slide along the base of the sea cliff.

Renewed activity of the landslide in the winter of 1996/1997 coincided with development of numerous en echelon cracks in the asphalt roadway and walkway along Shore Cliff Road. In the winter of 1999/2000, a water line separated beneath the paved roadway in the vicinity of the cracks. Comparison of pre- and post-construction topographic maps shows that the locations of these cracks coincide approximately with the contact between the road fill wedge and the underlying colluvium, suggesting that deformation in this area may be caused by fill settlement or creep. However, the arcuate pattern of the cracks and proximity to the Patton Cove landslide suggest that incipient landsliding is encroaching into the roadway. The cracks also are located in the general area of a crescent-shaped marine terrace riser mapped prior to road construction (Reference 4, Figure 2.5-8). The mapped terrace riser is more likely a subdued landslide headscarp.

To avoid the potential hazard of the landslide and unstable fill, the transport route will be constructed north of the existing road (Patton Cove Bypass, Figure 2.1-2), where the Patton Cove slide will pose no hazard (Section 2.6.1.12.3). The closest approach of the transport route will be about 100 ft north of the cracks at the intersection of Shore Cliff and Reservoir roads.

Significant movement of the upper Patton Cove landslide, if it occurs, will not impact the proposed transport route of the Patton Cove Bypass because its headward migration is limited by the depth of the slide, which is controlled by the elevation of the higher wavecut platform. The geometry of the landslide is such that it is unlikely to extend much farther landward because it would require either: (a) an extremely low slide plane angle in the alluvial fan deposits, or (b) a deeper slide plane that cuts through the bedrock materials. These scenarios are both considered to be very unlikely. The current head of the upper slide is located 110 ft from the edge of the proposed transport route. Continued movement of the lower slide, however, will probably continue to destabilize the upper slide and cause additional movements and increased cracking in Shore Cliff Road.

As discussed in FSAR Section 2.2.2.3, a Cask Transportation Program requires a walkdown of the transportation route prior to any transport operations. This walkdown ensures that no hazards are present from the Patton Cove landslide as evidenced by severe cracking of the roadway surface.

Additional information on the potential impact of the Patton Cove landslide on the transport route is provided in Reference 75 (PG&E Response to NRC Question 2-17).

An inclinometer was installed on the road shoulder closest to the slide in November 2000 to monitor the depth and rate of future movements. The inclinometer has recorded small movements near the contact between the base of the fill and the underlying colluvium and debris-flow deposits.

2.6.1.12.2 Landslide Potential at the ISFSI and CTF Sites

Detailed investigation of landslides in the plant site area (Reference 9) shows there are no existing deep-seated landslides or shallow slope failures at the ISFSI and CTF sites. Field mapping and interpretation of 1968 aerial photography (Figure 2.6-31) during the ISFSI site investigations confirmed the absence of deep-seated bedrock slides or shallow slope failures at the site.

Excavation of the existing slope at the ISFSI site was completed in 1971. No stability problems were encountered during excavation using bulldozers and scrapers, and the slope has been stable, with minimal surface erosion, since 1971. Prior to excavation of the slope, Harding Miller Lawson (HML) (Reference 9) described a shallow landslide in weathered bedrock (Figure 2.6-10) along a “shale seam” in their exploratory trench A (Reference 9, Plate D-3). This feature was less than 15 ft deep, and was removed entirely, along with underlying intact bedrock, to a depth of about 75 ft during excavation of the slope. Zones of “fractured, decomposed, and locally brecciated sandstone, siltstone, and shale” and “breccia and clay zones” described in HML trench A are interpreted to be friable dolomite zones and steep faults.

The Harding Miller Lawson landslide is apparent on 1968 black-and-white aerial photography, and is expressed by a subtle, arcuate headscarp, hummocky landscape, and locally thicker vegetation, probably reflecting high soil moisture within the slide debris (Figure 2.6-31). The slide was located along a slight swale in colluvial soils and possibly weathered bedrock that mantled the slope prior to excavation. The slide mass appears to have moved northeast along the axis of the swale, and not directly downslope. Because bedding is interpreted to dip to the northwest in this area, the landslide probably was not a bedrock-controlled failure. There is no evidence of deep-seated bedrock landslides on the 1968 aerial photographs; the ISFSI study area appears as a stable, resistant bedrock ridge in the photos.

Because surficial soils were removed from the ISFSI site area during past grading, there is no potential for surficial slides to adversely affect the site. There is no evidence of bedrock landslides below the ISFSI site or along the southern margin of Diablo Canyon near the raw water reservoir. Reservoir facilities (including the water treatment plant) and paved areas between the ISFSI and CTF sites and Diablo Canyon show no evidence of sliding or distress. Because the 290-ft Q_5 marine terrace is preserved locally across the ISFSI study area, it is apparent that no deep-seated bedrock slides have occurred since formation of the 430,000-year-old terrace, and the ridge is interpreted to be stable. Some shallow debris-flow failures and slumps were identified in surficial soil on the outermost 3 ft to 4 ft of weathered rock in the steep (45 to 65 degrees) slope below the raw water reservoir (Figure 2.6-7). These failures are shallow, and do not pose a stability hazard to the ISFSI or CTF sites, which are set more than 180 ft back from the top of the slope.

2.6.1.12.3 Landslide Potential Along the Transport Route

The transport route is located 100 ft north of the headscarp of the active Patton Cove landslide (Figure 2.6-7). Based on detailed mapping, borings, and an inclinometer, the landslide headscarp is defined by a series of cracks at the intersection of Shore Cliff and Reservoir Roads. A cross section through the landslide is shown in Figure 2.6-17. The geometry and depth of the slide plane indicate further headward encroachment of the landslide toward the transport route is not likely.

Where the transport route follows Reservoir Road at the base of the bedrock hillslope north from near Hillside Road, there are no bedrock landslides. Sandstone beds in the hillslope above the road dip obliquely into the slope at about 30 to 50 degrees (Figures 2.6-7, 2.6-11, and 2.6-19). These beds extend continuously across much of the hillside, providing direct evidence of the absence of bedrock slope failures (Figure 2.6-5). Small faults and joints in the rock mass do not appear to adversely affect potential slope stability, and the existing roadcut and natural slopes have no evidence of any slope failures.

Kinematic analyses of the bedding and fractures along the road were performed where the road borders the bedrock slope (Section 2.6.5.4.1). Two portions of the route were analyzed: a northern part from approximately station 43+00 to 49+00 (Figure 2.6-37), and a northwesterly stretch from approximately station 35+00 to 42+00 (Figure 2.6-38). The rock mass is stable against significant wedge or rock block failures; however, the analysis indicates that rock topple failure from the cutslope into the road is possible. Field evaluations indicate such failures would be localized and limited to small blocks. The existing drainage ditches on the inboard edge of the road would catch these small topple blocks.

Several colluvial or debris-flow swales are present above the transport route along Reservoir Road (Figures 2.6-5 and 2.6-7). These swales have been the source of past debris flows that primarily have built the large fans on the marine terraces over the past tens of thousands of years. Additional debris flows could develop within these swales during severe weather events, similar to those described elsewhere in the Irish Hills following the 1997 storms (Reference 8). Holocene debris-flow fan deposits extend to just below the road alignment, indicating that future debris flows could cross the road. However, large graded benches for an abandoned leach field system are present above a portion of the Reservoir Road, and concrete ditches and culverts are present in swale axes. These existing facilities will catch and divert much of the debris from future debris flows above the road. However, two debris-flow chutes are present above the road northwest of Hillside Road; this part of Reservoir Road is not protected from these potential debris flows. Based on the thickness of the colluvium in the swales (5 to 10 ft), and the slope profile, the maximum depth of debris on the road following severe weather is estimated to be less than 3 ft, which easily could be removed after the event.

2.6.1.13 Seismicity

A detailed analysis of the earthquake activity in south-central coastal California was presented in Reference 6. The report included the historical earthquake record in the region since 1800, instrumental locations from 1900 through May 1988, and selected focal mechanisms from 1952 to 1988. From October 1987 through May 1988, the earthquake catalog incorporated data recorded by the PG&E Central Coast Seismic Network (CCSN). This station network has operated continuously since then to monitor earthquake activity in the region.

The seismicity in the region is illustrated in two frames on Figure 2.6-39: (a) historical earthquakes of magnitude 5 and greater since 1830, and (b) instrumentally recorded seismicity of all magnitudes from 1973 through September 1987. Epicentral patterns of the microearthquakes (Figure 2.6-39) show that most of activity within the region occurs to the north, beneath the Santa Lucia Range and north of San Simeon, and in the southern onshore and offshore region south of Point Sal. Earthquakes in the southern offshore region extend westward to the Santa Lucia Bank area. Within about 15 miles of the ISFSI, small, scattered earthquakes occur between the Los Osos fault and faults of the Southwest Boundary fault zone (including the Irish Hills subblock (Section 2.6.1.3), in the nearshore region within Estero Bay, and along the Hosgri fault zone. Focal mechanisms along the Hosgri fault zone show right-slip displacement along nearly vertical fault planes (Reference 6, Figures 2-30 and 2-36).

McLaren and Savage (Reference 14) updated the earthquake record and present well-determined hypocenters and focal mechanisms for earthquakes recorded from October 1987 through January 1997 by the CCSN and by the U.S. Geological Survey, from north of San Simeon to the southern region near Point Arguello (Figure 2.6-40). No significant earthquakes occurred during this time period, and no significant change in the frequency of earthquake activity was observed. The largest event recorded was the local (Richter) magnitude 5.1 (duration magnitude 4.7) Ragged Point earthquake on September 17, 1991, northwest of San Simeon (Figure 2.6-40 inset). The focal mechanism of this event is oblique thrust, typical of nearby recorded earthquakes. Earthquake data since January 1997 also do not show any significant change in the frequency or epicentral patterns of seismic activity in the region.

The seismicity data presented in Reference 14 is consistent with the LTSP observations and conclusions (Reference 6). Specifically:

- Epicentral patterns of earthquakes have not changed. As shown in Figure 2.6-40, microearthquakes continue to occur to the north, along a northwest trend to San Simeon, east of the Hosgri fault zone, and in the southern offshore region.

- Selected seismicity cross sections A-A' through D-D' along the Hosgri fault zone (Figure 2.6-41) show that onshore and nearshore hypocenters extend to about 12-kilometers depth, consistent with the seismogenic depth range reported for the region (Reference 6). Seismicity cross section B-B', across the Hosgri fault zone, shows the Hosgri fault zone is vertical to steeply dipping. The earthquakes projected onto cross sections C-C' and D-D' are evenly distributed in depth.

- Focal mechanisms along the Hosgri fault zone (Figure 2.6-42) are primarily strike slip, consistent with the LTSP conclusion that the Hosgri is a northwest-trending, vertical, strike-slip fault (Reference 6). Mechanisms from events within the Los Osos/Santa Maria domain show oblique slip and reverse fault motion, consistent with the geology.

- The location of the 1991 Ragged Point earthquake in the San Simeon region, as well as its size and focal mechanism, are consistent with previous earthquakes in the region.

2.6.2 VIBRATORY GROUND MOTIONS

2.6.2.1 Approach

10 CFR 72.102(f) states the following "The…DE for use in the design of structures must be determined as follows:

> For sites that have been evaluated under the criteria of Appendix A of 10 CFR 100, the DE must be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant."

Thus, DCPP ground motions are considered to be the seismic licensing basis, in accordance with 10 CFR 72.102(f), for the evaluation ISFSI design ground motions. Seismic analyses for the ISFSI used ground motions that meet or exceed the DCPP ground motions.

Vibratory ground motions were considered in the design and analyses (Section 8.2.1) of: (1) the ISFSI pads, (2) the CTF, including the reinforced concrete support structure and structural steel, (3) the ISFSI casks and cask anchorage, (4) ISFSI pad sliding and cutslope stability, (5) transport route slope stability, and (6) transporter stability.

The approach used for developing the ground motion characteristics to be used for design and analysis of the ISFSI SSCs consisted of the following.

- Use the DCPP ground motions (Section 2.6.2.2) as the basis for developing the ISFSI design ground motions, in accordance with 10 CFR 72.102(f).

- Compare the earthquake source and distance and ISFSI site conditions with those at the DCPP to confirm the applicability of the DCPP ground motions to the ISFSI site.

- Because ISFSI pad sliding and cutslope stability, transport route slope stability, and transporter stability may be affected by longer-period ground motions than those characterized by the DCPP ground motions, develop appropriate response spectra for the analysis of these elements, conservatively taking into account the additional influence of near-fault effects, such as fault rupture directivity and fling, that have been recorded in recent large earthquakes.

- Develop, as necessary, spectra-compatible time histories for use in analyses and design.

2.6.2.2 DCPP Licensing-Basis Ground Motions

The basis for the DCPP design ground motions is discussed in the DCPP FSAR Update, Sections 2.5.2.9, 2.5.2.10, and 3.7.1. There are three design ground motions for the DCPP: the design earthquake (DE), DCPP FSAR Update, Figures 2.5-20 and 2.5-21; the double design earthquake (DDE), DCPP FSAR Update, Section 3.7.1.1; Reference 4; and the Hosgri earthquake (HE), DCPP FSAR Update, Figures 2.5-29 through 2.5-32, which was incorporated into the DCPP seismic design basis as part of the seismic reevaluation of applicable existing structures by PG&E, and is now required as part of the licensing basis at the plant.

As discussed in the DCPP FSAR Update, the seismic qualification basis for the plant is the original design earthquakes (DE and DDE), plus the HE evaluation, along with their respective analytical methods, acceptance criteria, and initial conditions. Future additions and modifications to the plant are to be designed and constructed in accordance with these seismic design bases. In addition, as discussed in the DCPP FSAR Update, certain future plant additions and modifications are to be checked against the insights and knowledge gained from the Long Term Seismic Program (LTSP) to verify that the plant's "high-confidence-of-low-probability-of-failure" (HCLPF) values remain acceptable (Reference 4). As part of the LTSP, response spectra were developed for verification of the adequacy of seismic margins of certain plant structures, systems, and components (Reference 6). The DE, DDE, HE, and LTSP spectra are defined for periods up to 1.0 second, 1.0 second, 0.8 second, and 2.0 seconds, respectively.

2.6.2.3 Comparison of Power Block and ISFSI Sites

The Diablo Canyon ISFSI site is located in the plant site area of the licensed DCPP; therefore, the applicability of the DCPP ground motions to the ISFSI site was evaluated by comparing the ground-motion response characteristics of the ISFSI site with those of the plant site, and by comparing the distance from the controlling seismic source to the plant with the distance from the controlling source to the ISFSI.

As described in Section 2.6.1.4.2 and shown in Figures 2.6-6 and 2.6-11, the power block and the ISFSI are sited on bedrock that is part of the same, continuous, thick sequence of sandstone and dolomite beds of unit b of the Obispo Formation. In the classification of site conditions used for purposes of ground-motion estimation, both of these sites are in the "rock" classification (Reference 12).

Shear-wave velocity profiles from both sites are compared in Figure 2.6-35. As these comparisons indicate, shear-wave velocities from surface refraction and borehole geophysical surveys at the ISFSI site are within the same range as those obtained at the power block site. The velocity profiles at both sites are consistent with the "rock" classification for purposes of ground-motion estimation (Reference 12).

The earthquake potential of the significant seismic sources in the region was characterized during development of the DCPP FSAR Update and the LTSP (Reference 6). The Hosgri fault zone, at a distance of 4.5 kilometers, was assessed to be the controlling seismic source for the DCPP (Reference 4, Sections 2.5.2.9 and 2.5.2.10; Reference 6, Chapters 3 and 4). The ISFSI is approximately 800 ft to 1,200 ft east of the power block, and is thus only slightly farther from the Hosgri fault zone (Figure 2.6-4).

Therefore, because both sites are classified as rock, and because within the rock classification they have similar ranges of shear-wave velocities, and the distance to the controlling seismic source is essentially the same, the DCPP ground motions are judged to be applicable to ISFSI design.

2.6.2.4 Spectra for ISFSI Pads, Casks and Cask Anchorage, and CTF

The DE, DDE, HE, and LTSP spectra (Figures 2.6-43 and 2.6-44; the DE is one-half the DDE and is not shown) are applicable to the analysis of the pads, casks and cask anchorage, and CTF (Reference 55) (Section 8.2.1.2).

For cask anchorage design, the design spectra were defined by the HE spectrum for periods up to 0.8 second, and the LTSP spectrum for periods up to 2.0 seconds. New three-component, spectrum-compatible time histories were developed for the HE and LTSP by modifying recorded ground motions using the spectral matching procedure described by Silva and Lee (Reference 15). The recorded time histories used in the spectral matching were selected based on their similarity to the DDE, HE and LTSP earthquakes. The NRC Standard Review Plan spectral matching criteria (Section 3.7.1, NUREG-0800) were followed for 4-percent, 5-percent, and 7-damping; however, the NUREG requirement for a minimum value for the power spectral density (PSD) based on an NRC Regulatory Guide 1.60 spectral shape is not applicable to the spectral shapes of the HE or LTSP. The objective of the minimum PSD requirement was met by requiring the spectrum of each time history to be less than 30 percent above and 10 percent below the target spectrum. This ensures that no Fourier amplitudes are deficient in energy for the frequency range of interest.

2.6.2.5 ISFSI Long-Period Earthquake (ILP) Spectra and Time Histories For Pad Sliding, Slope Stability, and Transporter Stability Analyses

Because ISFSI pad sliding and cutslope stability, transport route slope stability, and transporter stability may be affected by longer-period ground motions than those characterized by the DCPP ground motions, PG&E has developed longer-period spectra and associated time histories for the analysis of pad sliding, slope stability, and transporter stability (References 54, 56, 57, and 58). These are referred to as the ISFSI long period (ILP) ground motions (Figures 2.6-45 and 2.6-46). The ILP spectra represent 84th percentile horizontal and vertical spectra, at damping values of 2 percent, 4 percent, 5 percent, and 7 percent, which extend out to a period of 10 seconds.

New information has become available from analytical studies of near-fault strong-motion recordings of large earthquakes in the past decade to evaluate the influence of near-fault effects, such as fault rupture directivity and tectonic deformation (fling), especially on ground motions in the longer-period range. PG&E has incorporated the influence of rupture directivity and fling in the ILP spectra and time histories (References 54, 56, 57, and 58) used for the analyses of pad sliding, slope stability, and transporter stability.

Development of the ILP horizontal spectra (Figure 2.6-46) incorporated the following assumptions and considerations:

- Although the LTSP (Reference 6) considered alternative styles of faulting for the Hosgri fault zone, the weight of the evidence favored strike-slip, and subsequent earthquake data and geologic and geophysical data interpretations (References 14 and 16) indicate the style of faulting is strike slip. Therefore, ground-motion characteristics appropriate for strike-slip earthquakes were used.

- The effect of directivity was analyzed for the case in which rupture begins at the southern end of the Hosgri fault zone, progresses 70 kilometers to the northwest where it passes at a closest distance of 4.5 kilometers from the plant site, and continues an additional 40 kilometers to the northwest end of the Hosgri fault zone. This assumption is conservative, because this rupture scenario has the greatest directivity effects at the site.

- The ILP horizontal spectrum at 5-percent damping at periods less than 2.0 seconds envelopes the DDE, HE, and LTSP spectra.

- The spectrum based on the Abrahamson and Silva (Reference 17) attenuation relation is consistent with the envelope of the DDE, HE, and LTSP spectra at 2 seconds, and has the same slope-with-period as the Sadigh (Reference 18) and Idriss (References 19, 20, and 21) attenuation relations, so it was used to extrapolate the envelope spectrum to 10 seconds. This spectrum is the 84th percentile horizontal spectrum.

- The 5-percent-damped horizontal spectra were increased to assure they envelope the Hosgri spectra at 4-percent and 7-percent damping ratios. Scaling factors for computing spectra at damping values other than 5 percent are from Abrahamson and Silva (Reference 17).

- Abrahamson's (Reference 22) and Somerville and others' (Reference 23) models were used to scale the average horizontal spectrum, to compute the fault-normal and fault-parallel ground-motion components, incorporating directivity effects.

- The fault-normal component was increased in the period range of 0.5 second to 3.0 seconds to account for possible directivity effects for earthquakes having magnitudes less than 7.2 at periods near 1 second.

- Because fling can occur on the fault-parallel component for strike-slip faults, a model was developed (Reference 57) to compute the 84th percentile ground motion due to tectonic fling deformation at the ISFSI accompanying fault displacement on the Hosgri fault zone in a magnitude 7.2 earthquake. The fling arrival time was selected, and the fling and the transient fault-parallel ground motion were combined so as to produce constructive interference of the fling and the S-waves, resulting in a conservative fault-parallel ground motion.

Development of the ILP vertical spectra (Figure 2.6-46) incorporated the following assumptions and considerations:

- The ILP vertical spectrum at 5-percent damping at periods less than 2 seconds is defined by the envelope of the DDE, HE, and LTSP (Reference 4) spectra.

- Current empirical attenuation relations (References 17, 18, and 24) were used to estimate the vertical-to-average-horizontal ratio for periods greater than 2 seconds; the value of two-thirds is conservative. The envelope vertical spectrum at 5-percent damping at periods less than 2 seconds was extended to a period of 10 seconds using two-thirds the average horizontal spectrum.

- The 5-percent-damped vertical spectra were increased to assure they envelop the Hosgri spectra at 4 percent and 7 percent damping ratios. Scaling factors for computing spectra at damping values other than 5 percent are from Abrahamson and Silva (Reference 17).

Five sets of spectrum-compatible acceleration time histories were developed to match the ILP ground motions spectra (References 56 and 58). The recordings in the table below were selected because they are from strike-slip earthquakes of magnitude 6.7 or greater, recorded at distances less than 15 kilometers from the fault, and contain a range of characteristics of near-fault ground motions.

Earthquake	Magnitude	Recording	Distance (km)	Site Type
1992 Landers	7.3	Lucerne	1.1	Rock
1999 Kocaeli	7.4	Yarimca	8.3	Soil
1989 Loma Prieta	6.9	DCPP	6.1	Rock
1940 Imperial Valley	7.0	El Centro #9	6.3	Soil
1989 Loma Prieta	6.9	Saratoga	13.0	Soil

The NRC Standard Review Plan spectral matching criteria (Section 7.1, NUREG-0800) recommends 75 frequencies for spectral matching. Augmented frequency sampling at 104 frequencies was used to account for the broader frequency range being considered for the ISFSI analyses. The interpolation of the response spectral values was done using linear interpolation of log spectral acceleration and log period. The NRC requirement permits not more than 5 of the 75 frequencies to fall below the target spectrum, and no point to fall below 0.9 times the target spectrum. This requirement was adhered to with the 104 frequencies.

The time histories were matched to the target spectra at 5-percent damping. The mean response spectrum of the five sets must envelop the target to meet the criteria of SRP 3.7.1. This criterion was applied to the damping values of 2 percent, 4 percent, 5 percent, and 7 percent.

The fault-parallel time histories were modified to include the effects of fling.

2.6.2.6 Transport Route and Transporter Design-Basis Ground Motions

As discussed in Section 2.6.1.9 and shown in Figures 2.6-6 and 2.6-7, the transport route is underlain by Obispo Formation bedrock consisting of unit b dolomite and sandstone (the same bedrock as at the power block and ISFSI sites), and unit c claystone and shale. Varying thicknesses of dense soil deposits overlie the bedrock.

Because the transport route is about the same distance from the Hosgri fault zone as the DCPP and the ISFSI sites, the ILP spectra are appropriate for use along the transport route. An evaluation of the impact of a seismic event occurring during cask transport is discussed in Section 8.2.1.2.1.

2.6.3 SURFACE FAULTING

Potentially active faults at Diablo Canyon and in the surrounding region were identified and characterized in the DCPP FSAR Update, Section 2.5.3, the LTSP Final Report, Chapter 2, and the LTSP Addendum (Reference 7, Chapter 2). Together, these documents provide a comprehensive evaluation of the seismotectonic setting and location of capable faults in the plant site region, and document the absence of capable faults beneath the power block and in the plant site area. These studies used detailed mapping of Quaternary marine terraces and paleoseismic trenching to document the absence of middle to late Pleistocene faulting in the plant site area, including the ISFSI study area (Reference 6, p. 2-38, Plates 10 and 12). Hence, there are no capable faults at the ISFSI site.

Several minor bedrock faults were encountered in trenches at the ISFSI site during site characterization studies (described in Section 2.6.1.6.2.2). These faults are similar to minor faults that are commonly observed throughout the Miocene Obispo and Monterey formations in the Irish Hills (DCPP FSAR Update, Section 2.5.1; References 9 and 11). Similar minor bedrock faults encountered beneath the power block strike generally northwest to west, dip 45 degrees to 85 degrees, and have displacements of up to several tens of feet (Reference 4, Section 2.5.1.2.5.6, Figure 2.5-14).

The faults at the ISFSI site (Figure 2.6-8) are near-vertical (dip generally 70 to 90 degrees) and trend northwest, subparallel to the regional structural trend of the Pismo syncline (Figure 2.6-29). As described in Section 2.6.1.6.2.2, individual faults have vertical separation of a few tens of feet or less; cumulative vertical separation across the fault zone is greater than 50 ft, down on the northeast (Figure 2.6-11). Subhorizontal slickensides on the fault plane indicate a significant component of oblique strike slip, so total displacement is hundreds of feet. Detailed site investigations, including mapping and trench excavations, show that the individual faults generally extend across the ISFSI site and at least across the lower slope above the ISFSI.

The faults do not align with any significant bedrock fault in the plant site area (Figures 2.6-4 and 2.6-6), nor do the faults have major stratigraphic displacement. The origin of the faults may be related to one or more of three possible causes, all prior to 1 million years ago.

The faults most likely formed during a period of regional transtensional deformation during the Miocene, when normal and strike-slip faulting occurred in the region. This most directly explains the observed normal-oblique slip on the fault zone. A transition to transpressional deformation occurred during the late Miocene to Pliocene, and is well expressed in the offshore Santa Maria Basin and along the Hosgri fault zone (Reference 6). The minor bedrock faults at the ISFSI site were subsequently rotated during the growth of the Pismo Syncline, although the faults occur near the flat-lying crest of a small parasitic anticline and, thus, have not been rotated significantly. Given this origin, the faults formed during the Miocene, contemporaneous with the transtensional formation of Miocene basins along the south-central coast of California, prior to 5 million years ago.

Alternatively, the minor faults may be secondary faults related to growth of the regional Pismo syncline (Figure 2.6-4), as concluded for the small bedrock faults at the power block (Reference 4, p. 2.5-49, -50). As shown on Figure 2.6-29, the faults trend subparallel to the axis of the Pismo syncline, and are located near the crest of a small anticline on the southwestern limb of the syncline. The apparent oblique displacements observed on the faults may be related to bending-moment normal faults and right shear along the axial plane of the small anticline that formed in the Pliocene to early Quaternary. The zone of minor faulting may have used the area of diabase intrusion as an area of crustal weakness to accommodate tensional stresses along the axial plane of the anticline. As described in Reference 4, pages 2.5-14, -33, -34, and in the LTSP reports (Reference 6, p. 2-34 to -38; and Reference 7, p. 2-10), growth of the Pismo syncline and related folds ceased prior to 500,000 years to 1,000,000 years ago. Thus, the observed minor faults also ceased activity prior to 500,000 years to 1,000,000 years ago.

A third alternative explanation for origin of the minor bedrock faults is that they are related to intrusion of the diabase into the Obispo Formation. Diabase is present locally in the ISFSI study area. Forceful intrusion, or magmatic stoping of the diabase may have produced faulting in response to stresses induced by the magma intrusion in the adjacent host rock. Hydrothermal alteration is extensive in the diabase. The friable sandstone and dolomite in the ISFSI study area are spatially associated with the zone of faulting (Figures 2.6-8 and 2.6-11), indicating the faults may have acted as a conduit for hydrothermal solutions. Assuming the hydrothermal fluids were associated with the diabase intrusion, the minor faults predate, or are contemporaneous with, intrusion of the diabase. Diabase intrusion into the Obispo Formation occurred in the middle Miocene (References 10 and 11), indicating the faulting would have occurred prior to or contemporaneous with the diabase intrusion in the middle Miocene, more than 10 million years ago. The faulting may have originated by transtensional regional deformation, as described above, then subsequently was modified by diabase intrusion.

In addition to their probable origin related to transtensional deformation in the Miocene (or to growth of the Pismo syncline in the Pliocene to early Quaternary, or to intrusion of the diabase in the middle Miocene), several additional lines of evidence indicate the minor faults are not capable and do not present a surface faulting hazard at the site:

- As described in the Reference 6, pages 2-37 to -39, Plates 10 and 12), the Quaternary marine terrace sequence in the plant site vicinity is not deformed, providing direct stratigraphic and geomorphic evidence demonstrating the absence of capable faulting. The minor faults observed at the ISFSI site project northwest across, but do not visibly displace, any of the lower marine terrace platforms, within a limit of resolution of $\pm$5 ft, indicating the absence of deformation in the past 120,000 years. Assuming the displacement does not die out at the coast, this resolution is enough to recognize the greater-than-50-ft of vertical separation on the faults at the ISFSI site.

- As described in Reference 4, p. 2.5-35 to -50, Figures 2.5-13 to 2.5-16, similar northwest-trending minor faults were mapped in bedrock in the power block area. Detailed trenching investigations of these faults and mapping of the power block excavation provided direct observation that they do not displace and, hence, are older than the late Pleistocene (120,000 years old) marine terrace deposits. By analogy, the minor faults at the ISFSI site also would be older than late Pleistocene.

- Interpretation of aerial photographs taken before the 1971 excavation of the ISFSI site area (former borrow area) and construction of the raw water reservoir (Figure 2.6-31), shows there are no geomorphic features in the ISFSI study area (tonal lineaments, drainage anomalies, scarps) indicative of displacement of the minor faults prior to grading. The landscape in the ISFSI study area is interpreted to have formed in the middle to late Quaternary, about 430,000 years ago, based on the preserved remnants of marine terraces in the surrounding site area.

Based on these lines of evidence, the minor faults observed in bedrock at the ISFSI site are not capable; hence, there is no potential for surface faulting at the ISFSI site.

2.6.4 STABILITY OF SUBSURFACE MATERIALS

2.6.4.1 Scope

An extensive program of field investigations, in situ testing, and laboratory testing was conducted to define the static and dynamic characteristics of the soil and rock materials. The scope of the program is summarized in Table 2.6-1. A detailed discussion of the test procedures and results is presented in References 44, 45, and 48 through 51, Data Reports B, C, F, G, H, and I, and Reference 9. The results are summarized below.

2.6.4.2 Subsurface Characteristics

The geology at the subgrade of the foundation of the ISFSI pads (elevation about 302 ft, 8 ft below the pad grade) is shown in Figure 2.6-32. The subsurface beneath the ISFSI pads consists of dolomite (Tof_{b-1}), sandstone (Tof_{b-2}), friable dolomite (Tof_{b-1a}), and friable sandstone (Tof_{b-2a}) (Section 2.6.1.7). The bedrock contains minor faults and joints (Section 2.6.1.6.2). The groundwater table is near elevation 100 ft, about 200 ft below the foundation elevation. Clay beds of limited extent occur at a few locations under the ISFSI pads, below the surface of the cutslope, and in the existing slope above the pads (Section 2.6.1.7).

The geology at the CTF foundation grade is shown in Figure 2.6-32. At this grade (elevation about 286 ft), the bedrock consists of sandstone and friable sandstone (Section 2.6.1.8). At the site, the sandstone may have a few minor faults and joints, similar to those described in Section 2.6.1.6.2. Because the rocks are the same, the static and dynamic engineering properties of the rock at the foundation of the CTF were selected to be the same as those at the ISFSI pads.

The transport route traverses thick surficial deposits along nearly two-thirds of its route, including a new 500-foot-long section of thick, engineered fill near Patton Cove. It is constructed on engineered fill placed on dolomite and sandstone for the rest of its length (Section 2.6.1.9).

The detailed geologic characteristics of these rock units are described in Section 2.6.1.4.2.

2.6.4.3 Parameters for Engineering Analysis

2.6.4.3.1 ISFSI and CTF Sites

The static and dynamic engineering properties for use in foundation analyses of the rock at the ISFSI and CTF sites are as follows:

Density: A density of 140 pounds per cubic ft (pcf) was chosen as appropriate for foundation analyses (Reference 51, Data Report I).

Strength: A friction angle for the rock mass of 50 degrees was chosen as appropriate for foundation analyses. This friction angle is consistent with that used in the slope stability analyses (Section 2.6.5.1.2.3).

Poisson’s ratio: A representative value of Poisson’s ratio of 0.22 for dolomite and sandstone was selected as appropriate for analyses. A representative value of 0.23 was selected for friable rock. These values were derived from seismic velocity measurements in the bedrock below the footprint of the pads (Reference 59), and laboratory-based measurements (Reference 60) on samples of bedrock from beneath the pads (Reference 37 and Reference 51, Data Report I).

Young's modulus: Representative values of Young's modulus of between 1.34 times 10^6 psi (mean) and 2.0 times $l0^6$ psi (84^{th} percentile upper bound) for dolomite and sandstone were selected as appropriate for analyses. A representative value of 0.2 times 10^6 psi was selected for friable rock. These values were derived from seismic velocity measurements in the bedrock below the footprint of the pads (Reference 59), and laboratory-based measurements (Reference 60) on samples of bedrock from beneath the pads (Reference 37 and Reference 51, Data Report I).

2.6.4.3.2 Slopes

Static and dynamic engineering properties of soils and rock at the ISFSI site for use in slope stability analyses are as follows:

Rock Strength: A friction angle of 50 degrees for the rock mass was selected for stability analyses of the hillslope above the ISFSI pads. A range of friction angles between 16 degrees and 46 degrees for rock discontinuities was selected for stability analyses of the cutslopes behind the ISFSI pads. Further discussion of rock strength parameters is provided in Sections 2.6.5.1.2.3 and 2.6.5.2.2.3.

Clay Bed Strength and Unit Weight: The following parameters were defined for clay:

- unit weight, 115 pcf (Reference 49, Data Report G)
- shear strength, drained, c' = 0 psf; ø' = 22 degrees
- shear strength, undrained, lower of c = 800 psf and ø=15 degrees or ø = 29 degrees.

Further discussion of clay strength parameters is provided in Section 2.6.5.1.2.3.

Shear wave velocities: Representative values of shear wave velocities were selected for stability analyses (Section 2.6.5.1.3.2). These values were based on suspension geophysical surveys in boreholes beneath the footprint of the pads, as well as on data summarized in the Addendum to the LTSP Final Report (Reference 7, Chapter 5, Response to Question 19).

Dynamic shear modulus and damping values: Relationships of the dynamic shear modulus and damping values with increasing shear strain were selected for stability analyses (Section 2.6.5.1.3.2), based, in part, on literature review and dynamic tests of DCPP rock core samples performed in 1977 and 1988 (Reference 41).

Additional considerations for the selection of rock and clay properties for specific static and dynamic stability analyses, and the calculation of seismically induced displacements are presented in Section 2.6.5.1.3.

2.6.4.3.3 Transport Route

As described earlier, the transport route generally follows existing Plant View, Shore Cliff, and Reservoir roads (Figure 2.6-7). The specifications for the construction of these roads required all fills to be compacted to 90-percent relative density, and the upper 2.5 ft to be compacted to 95-percent relative density. Fills on slopes were benched and keyed a minimum of 6 ft into the hillside. Based on these requirements, the road base and subgrade material are expected to be at least as capable for transporter loads and earthquake ground motions as the underlying rock and soil.
The new section of the transport route near Patton Cove will be constructed on engineered fill, as will a section of the route near the CTF (Figure 2.6-7). These fills and the overlying road subgrade also will be constructed to the same specifications as the existing roads. Both new roadway sections will support the imposed loads.

Where the transport route follows Plant View, Shore Cliff and Reservoir roads to Hillside Road, the alignment is founded on marine terrace deposits overlain by dense colluvial deposits. The remaining portions of the route, on Reservoir Road (beyond station

34+00), are founded on cuts made in the dolomite and sandstone. The static and dynamic engineering properties of the rock and soil deposits underlying the transport route are summarized in Reference 9, Tables 1 and 2.

2.6.4.4 Static Stability

The ISFSI pads will be founded on dolomite, sandstone, friable dolomite, and friable sandstone (Figure 2.6-32). The CTF will be founded on sandstone and friable sandstone (Figure 2.6-32). This bedrock will support the proposed facilities without deformation or instability (References 61 and 71). The NRC reached the same conclusion using an alternate analysis method in their Safety Evaluation Report (SER) for the Diablo Canyon ISFSI License Application (Reference 77). Although the NRC used a more conservative methodology, PG&E's calculation methodology is acceptable for use in future ISFSI design activities (Reference 78).

The borrow excavation removed between 75 ft and 100 ft of rock from the ISFSI and CTF sites. As a result, the existing rock is over-consolidated, and facility loads are likely to be much less than the former overburden loading on the rock (calculated to be about 10,000 to 14,000 psf). The over-consolidated state of the rock mass in the foundation precludes any settlement, including differential settlement between rock types, under the planned loading conditions.

As discussed in the DCPP FSAR Update, Section 2.5.4, there are no mines or oil wells in the plant site area. Two makeup-water wells have drawn water from fractured bedrock that is fed groundwater from the shallow alluvium along Diablo Creek (Section 2.5). One of these wells (Well No. 1) is no longer in use (since 2008). No subsidence has been observed, nor is any expected, near these wells, which are approximately 2,500 ft east of the ISFSI.

Similarly, there is no evidence of solution features or cavities within the dolomite and sandstone strata, or in the friable dolomite and friable sandstone, beneath the ISFSI, or in the plant site area. Hence, there is no potential for karst-related subsidence or settlement at the ISFSI or CTF sites.

There is no potential for differential settlement across the different rock units (sandstone, dolomite, friable sandstone, friable dolomite) at the ISFSI, because the rock is well consolidated, joints and fractures are tight, and the friable rocks have almost no joints. Although no piping voids in the friable rocks are expected beneath the ISFSI pads, very small voids (a few inches across) are possible, as found in the friable dolomite in one of the trenches (Reference 46, Data Report D, trench T-20A). The foundation will be below the dilated zone for the borrow area cutslope (observed to be at about 4 ft in the trenches), and the rock mass is expected to be tight, with no open fractures. The rock mass is also over-consolidated, having had 100 ft of rock overburden removed from the location of the borrow area in the vicinity of the ISFSI for construction of the raw water reservoir and the 230-kV and 500-kV switchyards (Figure 2.6-10).

There is no potential for displacement on the faults at the sites, because the faults are not capable (Section 2.6.3). No differential displacement or settlement is expected during potential ground shaking.

2.6.4.5 DYNAMIC STABILITY

The ISFSI is located entirely within bedrock. There are no loose, saturated deposits of sandy soil beneath the pads or CTF site, and the groundwater table is near elevation 100 ft, about 200 ft below the foundation level. Therefore, there is no potential for liquefaction at either site.

The CTF foundation is embedded into rock at least 20 ft below grade, as shown in Figure 2.6-32. This precludes the development of unstable foundation blocks under static or dynamic loading conditions.

Because the transport route subgrade will be on engineered fill on rock and well-consolidated surficial deposits, no liquefaction or other stability problems are expected.

An analysis was performed to verify pad stability during an earthquake (References 72 and 79; also see Section 8.2.1.2.3.2). This analysis included the inertial effects of the pad and pre-existing structures. Additional information is provided in Reference 75 (PG&E Response to NRC Question 2-16) and Reference 76 (PG&E Response to NRC Request 2). The NRC reached the same conclusion using an alternate method in their SER for the Diablo Canyon ISFSI License Application (Reference 77). Although the NRC used a more conservative methodology, PG&E's calculation methodology is acceptable for use in future ISFSI design activities (Reference 78).

2.6.4.6 POTENTIAL FOR CONSTRUCTION PROBLEMS

No significant construction-related problems are anticipated for preparation of the ISFSI and CTF foundations subgrade. The permanent groundwater table is about 200 ft below the planned foundation elevations (Section 2.5), and groundwater is not expected to rise to within the zone of foundation influence. The rock mass is generally tight, and does not have significant voids or soft zones that would require grouting or dental work, with the possible exception of small piping voids related to the friable dolomite. The fractures are tight or filled, and are tightly confined by the surrounding competent rock. The prepared foundation pads will be level, and will be a considerable distance from descending slopes, thus precluding development of unstable blocks or foundation loads into slopes.

2.6.5 SLOPE STABILITY

The ISFSI is located on the lower portion of a hillslope that has been modified by excavation for borrow materials during the construction of the DCPP. Construction of the ISFSI pads, the CTF, and portions of the transport route includes cutslopes and fills. The purpose of this section is to examine the stability of the hillslope and the cuts and fills. For each slope, the static and seismic stability were analyzed, and the potential seismically induced displacements were estimated.

The analyses, which are summarized in Table 2.6-2, show that the hillslope and the cutslopes above the ISFSI are generally stable under modeled seismic inputs, slope geometries, and material properties. The seismically induced displacements of the rock mass above the ISFSI, estimated using very conservative assumptions, are small. Under the modeled seismic loads, small rock wedges appear to be susceptible to movement in the cutslopes around the pads. These potential hazards are mitigated by setbacks in slope design, rock anchors, and debris fences, as discussed in Section 4.2.1.1.9.1. The slopes along the transport route and below the CTF are stable.

For each slope analysis, the objectives and scope of the stability analysis are defined, and the analysis methods are described. The slope geometry and selection of material properties are then given. Finally, the results of the analyses for the hillslope above the ISFSI, the ISFSI cutslopes, the slope below the CTF, and slopes along the transport route are presented.

2.6.5.1 Stability of the Hillslope above the ISFSI

A critical section of the hillslope above the ISFSI was analyzed to examine the static and dynamic stability of the jointed rock mass along postulated slide surfaces. Analyses also were conducted to estimate potential seismically induced displacements due to the vibratory ground motions derived in Section 2.6.2. In addition, an analysis was conducted to evaluate the conservatism of the analysis parameters and examine the geologic data to estimate past displacements due to earthquakes. In Reference 76 (PG&E Response to NRC Request 3), PG&E has performed additional evaluations to address: (a) the potential for a generalized slip-circle type failure of the cutslopes and hill slope above the ISFSI and (b) the effect of the cutslope (i.e., the excavation for construction of the ISFSI) on the stability of the hill slope above the ISFSI. The results of the evaluations found that the clay bed failure of the slopes governed the design and the effect of the cutslope on slope stability was minimal.

2.6.5.1.1 Geometry and Structure of Rock Mass Slide Models

Cross section I-I' (Figure 2.6-18) parallels the most likely direction of potential slope failure, and illustrates the geometry of bedding in the ISFSI study area for analysis of slope stability. The cross section shows apparent dips, and the facies variation and interfingering of beds between the dolomite and sandstone (units Tof_{b-1} and Tof_{b-2}) beneath the slope. The clay beds, where orientation and extent are critical to this evaluation of slope stability, have been correlated based on stratigraphic position, projection of known bedding attitudes, and superposition of sandstone and dolomite beds (clay beds have not been allowed to cross cut dolomite or sandstone beds, but have been allowed to cross facies changes). These clay beds, as drawn, are a conservative interpretation of their lateral continuity for the analysis of the stability of the slope.

Individual clay beds that are, in places, thick (more than about 0.5-inch thick) in the dolomite, may continue up to several hundred feet. Thinner clay beds are less laterally continuous. On cross section I-I' (Figure 2.6-18), clay beds are not shown to extend continuously through the slope, but are terminated at set distances from exposures in boreholes, trenches, or outcrops, reflecting the estimates of possible lateral continuity. Because of the generally limited lateral continuity of the clay beds, potential large surfaces (greater than several hundred feet in maximum dimension) likely would require sliding on several clay beds, and stepping between beds on joints and in places through rock in a “staircase” profile. Stepping between basal clay failure surfaces would probably be localized where the individual clay beds are stratigraphically close and are thin and pinch out. Other likely locations for stair-stepping failure or structural boundaries for possible rockslide margins are at changes in structural orientation (transitions from monocline to syncline), and along the lateral margins of the slide. These limit the size of potential rock masses. Faults at the site are subparallel to the potential down-slope motion, and impart a strong near-vertical fabric in the rock mass. It is likely that lateral margins for potential larger rockslides would develop, at least partially, along these faults.

Based on the above considerations, three rock mass slide models, comprising ten potential slide surfaces, were defined for cross section I-I' of the hillslope:

Model 1. A shallow slide mass model (Figure 2.6-47) involving sliding rock masses along shallow clay beds encountered in trench T-14A and boring 01-I. It toes out at the upper part of the tower access road.

Model 2. A medium-depth slide mass model (Figure 2.6-48) involving sliding rock masses along clay beds encountered at depths of between about 25 ft and 175 ft in borings 01-F, 00BA-1, and 01-I, and trench T-11D. It toes out on the slope between the ISFSI and below the tower access road.

Model 3. A deep slide mass model (Figure 2.6-49) involving sliding along deep clay beds encountered in borings 01-H, 01-F, 00BA-1, and 01-I at depths of between about 50 ft and 200 ft. It toes out behind or below the proposed ISFSI cutslope and pads.

Model 1 has been segmented into two possible geometries, labeled 1a and 1b on Figure 2.6-47. These two modeled slide blocks daylight at a clay bed encountered in trench T-14A (model 1a), or along the projected dip of a clay bed encountered in boring 01-I. The failure headscarp/tension break-up zone extends upward from the inferred maximum upslope extent of the clay bed in trench T-14A (model 1a), or from the inferred likely uphill extent of a clay bed encountered in boring 01-I.

Model 2 has been segmented into three subblocks: 2a, 2b, and 2c (Figure 2.6-48). The three blocks daylight along a clay bed encountered in trench T-11D (2a and 2b), or along the dip projection of a clay bed encountered in boring 00BA-1 (2b). Model 2a breaks up near trench T-14A at the location of a major structural discontinuity for potential slide blocks; the transition between the monocline and syncline where the bedding geometry (strike and dip) changes. Models 2b and 2c break up from the basal failure planes in a “stair-stepping” manner between clay beds, and have a common headscarp that daylights about 50 ft above the brow of the 1971 borrow cut excavation. The geometry of the headscarp break-up zone is inferred to be controlled by the uphill limit of clay beds encountered in the borings, and dominant steep joint fabric in the rock mass.

Model 3 has been segmented into three subblocks: 3a, 3b, and 3c. The three blocks daylight in the ISFSI pads cutslope, or at the base of the cutslope (Figure 2.6-49). All three modeled blocks have basal slide surfaces along clay beds encountered in borings 01-F, or 00BA-1 and 01-I. Models 3a and 3b break up with headscarp/tension zones at the location of the structural change in bedding geometry described for model 2a (3a and 3b), or about 75 ft above the top of the borrow cut (3c) at an inferred maximum uphill extent of clay beds encountered in boring 01-I. Model 3 has been further segmented into 3c-1, which daylights beyond the ISFSI pads, and 3c-2, which daylights at the base of the first cutslope bench.

For all models, the toe daylight geometry reflects the propensity for failure planes to break out along bedding planes and along the projection of clay beds. In contrast, the geometry of the headscarp/tension failure was inferred to be controlled by the dominant steep (greater than 70 degrees) joint/fault fabric in the rock mass.

2.6.5.1.2 Static Stability Analysis

2.6.5.1.2.1 Method

The static stability analysis of the hillslope was conducted using the computer program UTEXAS3 (Reference 26). Spencer’s method, a method of slices that satisfies force and moment equilibria, was used in the analysis.

2.6.5.1.2.2 Assumptions

The following assumptions were made:

- The clay beds are saturated. This assumption is reasonable, because during the rainy season, rainfall would infiltrate the slope through the fractured rock and perch temporarily on the clay beds, and would saturate at least the upper part of the clay.

- There is little water in the slope. This assumption is reasonable, because the groundwater table is about 200 ft below the ISFSI site, and the rock is fractured and well-drained. There are no springs from perched water tables near the ISFSI slope.

- The lateral margins of the potential slide masses have no strength. This is conservative, because the margins of a potential failure wedge would, in part, follow discontinuous joints, small faults, and, in part, break through rock. Friction between rock surfaces and by asperity overriding, or shearing along the lateral slide margins would provide some resistance to sliding.

- The upper 20 ft of the rock mass forming the head of a potential sliding mass has been modeled as a tension crack, that is, the zone has been given no strength. This assumption is conservative, because the dilated zone is only about 4 ft deep (Reference 37).

- The head of the slide below the tension crack would break irregularly along joints and clay beds and through some rock. The strength assigned to this rock mass is discussed below.

- The orientation, continuity, and extent of the clay beds is assumed to be as shown on cross section I-I'. This is reasonable, because the extent of the clay beds and their dip is based on extensive geologic data from the ISFSI study area.

- The strength of the clay (discussed below) is assumed to apply along the entire length of a clay bed, as shown on cross section I-I'. This is conservative, because the clay beds are commonly thin and irregularly bedded, providing rock contact through the beds, thereby increasing the strength.

2.6.5.1.2.3 Material Properties

Drained and undrained clay-bed strength parameters were developed from the results of strength and index testing performed on clay-bed samples collected from borings and trenches excavated at the site. Strength tests consisted of consolidated-undrained triaxial compression tests (CU) with pore pressure measurements, drained and undrained monotonic direct-shear tests, and undrained cyclic direct-shear tests (Reference 49, Data Report G). Atterberg limits tests were conducted on the clay-bed samples to measure their liquid limits (LL) and plasticity indices (PI). Drained strength parameters were developed from the results of the CU triaxial and drained monotonic direct-shear tests, and from published empirical correlations with Atterberg limits. Drained strength was taken as the post-peak strength (defined as strength at the maximum displacement) from the drained direct-shear tests, and the lower of either the stress at 5 percent axial strain or the post-peak strength for the CU tests. Undrained strength parameters were developed from the results of the CU triaxial, undrained monotonic and cyclic direct-shear, and Atterberg limits tests. As with the drained strength parameters, the undrained strength was taken as post-peak strength from the monotonic direct-shear tests, and the lower of either the stress at 5 percent axial strain or the post-peak strength for the CU tests.

Undrained strength parameters c = 800 psf and ø = 15 degrees were determined from analysis of the undrained strength data (Figure 2.6-50). Similarly, c' = 0 psf, and ø' = 22 degrees were selected based on analysis of the drained strength data (Figure 2.6-51). Because the overburden pressure under the original ground surface is higher than the consolidation pressure used in most of the laboratory strength tests, overconsolidation effects are likely present in the laboratory test results. This effect was conservatively removed at low confining pressures by estimating corresponding undrained shear strengths for a maximum over-consolidation ratio (OCR) of 3.0 and determining an equivalent friction angle, as shown in Figure 2.6-50, of 29 degrees (with no cohesion). Accordingly, undrained strength parameters were selected as the lower of ø = 29 degrees and ø = 15 degrees with c = 800 psf (Figure 2.6-50). Strength envelopes for the clay beds are described in Reference 73.

An expanded description of the development of clay bed strength is provided in Reference 76 (PG&E Response to NRC Request 3).

Two different empirical methods were used to develop in situ rock mass strength envelopes for the dolomite and sandstone (units Tof_{b-1} and Tof_{b-2}): Barton and Choubey (Reference 27), and Hoek and Brown (Reference 28).

The Barton-Choubey method estimates the in situ shear strength of naturally occurring rock discontinuities (joints, bedding planes, faults) in relatively hard rock on the basis of field and laboratory measurements of discontinuity properties. Mean and standard deviation determinations of unconfined compression strengths for hard rock at the DCPP ISFSI are provided in Reference 63. The base friction angle along rock discontinuities based on laboratory tests (Reference 62) was used as input for the Barton-Choubey method. Shear strength envelopes for discontinuity surfaces within the shallow rock mass at the ISFSI site were used in the stability analyses of surficial rock mass sliding, wedge, and topple slope failures in the proposed cutslope above the ISFSI, and frictional sliding along shallow rock discontinuities below the foundation of the ISFSI pads. The range of strength envelopes for dolomite (Tof_{b-1}) and sandstone (Tof_{b-2}) discontinuities calculated using the Barton-Choubey method (Reference 64) are plotted in Figure 2.6-52, using the derived stress-strain data.

The Hoek-Brown method is an empirically based approach that develops nonlinear shear-strength envelopes for a rock mass, and accounts for the strength influence of discontinuities (joints, bedding planes, faults), mineralogy and cementation, rock origin (for example, sedimentary or igneous), and weathering. The resulting rock-mass shear-strength envelopes were used for evaluation of the ISFSI pads and CTF foundation properties, and for stability analyses of potential bedrock failures within jointed confined rock at the ISFSI site. The Hoek-Brown method is for rock masses having similar surface characteristics, in which there is a sufficient density of intersecting discontinuities such that isotropic behavior involving failure along multiple discontinuities can be assumed. The method is not for use when failure is anticipated to occur largely through intact rock blocks, or along discrete, weak, continuous failure planes (such as weak bedding interfaces). The structure (or failure) geometry must be relatively large with respect to individual block size. The rock mass conditions and relative size differences between rock blocks, potential deep-seated masses, and the ISFSI and CTF foundations for which the Hoek-Brown criterion is being applied are appropriate and meet these rock-mass requirements. Strength envelopes for dolomite and sandstone calculated using the Hoek-Brown method (Reference 65) are plotted in Figures 2.6-53 and 2.6-54, using the derived stress-strain data.

A strength envelope having a friction angle, ø, of 50 degrees and cohesion, c, of zero was selected for the portion of the rock mass consisting of dolomite (unit Tof_{b-1}) and sandstone (Tof_{b-2}) below the dilated zone (Figures 2.6-53 and 2.6-54) (Reference 65). This envelope is lower than (but approximately parallel to) the envelopes for either dolomite or sandstone derived from the empirical Hoek-Brown method, and is more nearly equal to the post-peak strength envelope for the friable sandstone derived from laboratory tests of nonjointed rock blocks. The interpreted post-peak strength envelope for the friable rocks has a friction angle, ø, of 51.2 degrees and cohesion, c, of zero (Figure 2.6-55) (Reference 49). Accordingly, a ø of 50 degrees was also selected for the friable rocks.

Reference 76 (PG&E Response to NRC Request 1) provided additional justification for the rock mass strength material properties developed using the Hoek-Brown methodology.

2.6.5.1.2.4 Results

The static factors of safety computed using UTEXAS3 (Reference 26) for the ten slide surfaces analyzed are shown in Table 2.6-3 (Reference 67). The table shows that, in all cases, the computed factor of safety varies between 1.62 and 2.86. It is, therefore, concluded that the hillslope is stable. The NRC reached the same conclusion using an alternate analysis method in their SER for the Diablo Canyon ISFSI License Application (Reference 77). Although the NRC used a more conservative methodology, PG&E's calculation methodology is acceptable for use in future ISFSI design activities (Reference 78).

2.6.5.1.3 Seismically Induced Displacements

2.6.5.1.3.1 Method

The selected slide surfaces were analyzed to estimate the potential for earthquake-induced displacements by using the concept of yield acceleration proposed by Newmark (Reference 29) and modified by Makdisi and Seed (Reference 30). The procedure used to estimate permanent displacements involved the following steps:

- A yield acceleration, k_y, at which a potential sliding surface would develop a factor of safety of unity, was estimated using limit-equilibrium, pseudostatic slope-stability methods. The yield acceleration depends on the slope geometry, the phreatic surface conditions, the undrained shear strength of the slope material, and the location of the potential sliding surface.

- Computations were made using UTEXAS3 (Reference 26) to identify sliding masses having the lowest yield accelerations. A two-stage approach was used that consisted of first calculating the normal stresses on the failure plane under pre-earthquake (static) loading conditions using drained strength properties. For each slice, the normal effective stress on the failure plane was then used to calculate the undrained strength on the failure plane. In the second stage of the analysis, horizontal seismic coefficients were applied to the potential sliding mass, and the stability analysis was repeated using the undrained strengths calculated at the end of the first stage. The yield acceleration was calculated by incrementally increasing the horizontal seismic coefficient until the factor of safety equaled unity.

  The material properties used for the UTEXAS3 analysis (unit weights and shear strength) were the same as those for the static stability calculations. Drained rock strengths were used for both stages of the yield acceleration analysis. Drained clay strengths were used for the first stage and a bilinear undrained strength envelope was used for the clay beds in the second stage of the analysis.

- The seismic coefficient time history (and the maximum seismic coefficient, kmax) induced within a potential sliding mass was estimated using two-dimensional, dynamic finite-element methods. The seismic coefficient is the ratio of the force induced by an earthquake in a sliding block to the total mass of that block. Alternatively, the seismic coefficient time history can be obtained directly by averaging acceleration values from several finite-element nodes within the sliding block at each time interval, as long as variations in the accelerations between nodes are not substantial. Development of seismic coefficient time histories is further discussed in Reference 76 (PG&E Responses to NRC Requests 4 and 7).

- Earthquake-induced seismic coefficient time histories (and their peak values, kmax) for the potential sliding surfaces were computed using the two-dimensional, dynamic finite-element analysis program QUAD4MU (Reference 31). This is a time-step analysis that incorporates a Rayleigh damping approach, and allows the use of different damping ratios in different elements. The program uses equivalent linear strain-dependent modulus and damping properties, and an iterative procedure to estimate the nonlinear strain-dependent soil and rock properties.

- The QUAD4MU program (Reference 31) was used to analyze three slide surfaces (1b, 2c, and 3c) for which the calculated yield acceleration values were the lowest (Table 2.6-3). Because the base of the finite element mesh is at a depth of 300 ft, and because QUAD4MU only allows the input motion to be applied at the base, the base motion was first computed by deconvolving the surface ground motion using a one-dimensional wave propagation analysis (SHAKE, Reference 32) to obtain motions at the level of the base of the two-dimensional finite-element model.

- For a specified potential sliding mass, the seismic coefficient time history of that mass was compared with the yield acceleration, ky. When the seismic coefficient exceeds the yield acceleration, downslope movement will occur along the direction of the assumed failure plane. The movement will decelerate and will stop after the level of the induced acceleration drops below the yield acceleration, and the relative velocity of the sliding mass drops to zero. The accumulated permanent displacement was calculated by double-integrating the increments of the seismic coefficient time history that exceeds the yield acceleration.

Reference 75 (PG&E Response to NRC Question 2-18) provided the results of a two-dimensional FLAC analysis that was performed to demonstrate the reasonableness of the displacements calculated using the described Newmark-type approach.

2.6.5.1.3.2 Material Properties

The material properties needed for the QUAD4MU analyses are the unit weight, the shear modulus at low shear strain, G_{max}, and the relationships describing the modulus reduction and damping ratio increase with increasing shear strain (Reference 40, Figures 7 and 8). The rock mass was modeled as having a unit weight and shear wave velocity that vary with depth, based on field measurements of shear wave velocity and laboratory values for unit weight. The shear wave velocity profile used is shown in Reference 40, Figure 6.

2.6.5.1.3.3 Seismic Input

The seismic input consisted of the five sets of time histories developed to match the ILP ground-motion spectra (Section 2.6.2.5). Both fault-parallel and fault-normal components were defined for each of the five motions postulated to occur on the Hosgri fault zone at a distance of 4.5 kilometers from the site. Because the strike of the Hosgri fault zone is 36 degrees from the orientation of cross section I-I', the input motions were rotated to the direction of cross section I-I'. For a specified angle of rotation, there will be 10 rotated earthquake motions along I-I', because the fault-normal component will be either positive (to the east) or negative (to the west) and each needs to be considered separately.

2.6.5.1.3.4 Analysis

Acceleration time-histories were calculated for 26 locations within the three selected slide surfaces (1b, 2c, and 3c) (Reference 41, Figure 2). Average acceleration time

histories were computed for each rock mass. Sensitivity studies using a cross section having a slightly different orientation indicated that the calculated peak accelerations are not significantly influenced by orientation or the total height of the hillslope.
Because the slope at the ISFSI site is a rock slope and its seismic response is anticipated to be generally similar to the input rock motions, the earthquake-induced deformation was first estimated using a Newmark-type analysis for a sliding block on a rigid plane (Reference 29). An estimated yield acceleration of 0.20 g (Table 2.6-4) was used to calculate the deformation of the sliding block. The displacement was computed for the negative direction (representing down-slope movement) only. The down-slope permanent displacement of the sliding block was integrated by using rock motions in the positive direction (representing up-slope direction) only. These preliminary displacement estimates were used to help in selecting the ground-motion time histories that provided the largest permanent displacement.

Table 2.6-4 shows the calculated down-slope permanent displacements (for the five sets of rotated rock motions) following the Newmark sliding block approach. The results (for a rotation angle ø = 36 degrees) indicate that, on average, ground-motion sets 1, 3, and 5 provided the largest displacements (2.4 ft to 2.9 ft). A sensitivity analysis was performed to evaluate the effect of the uncertainty in the direction of cross section I-I' (Figure 2.6-18) relative to the fault strike (Figure 2.6-29). For this analysis, ø was varied by ±10 degrees. As shown in Table 2.6-4, for a ø of 46 degrees, ground-motion set 1 (with a negative fault-normal component) and set 5 (with a positive fault-normal component) produced the largest displacements (3.3 ft and 2.8 ft, respectively). This is because the fault-normal components are stronger than the fault-parallel components in most cases, and for a ø of 46 degrees, the I-I' direction is closer to the fault-normal direction. Set 3, when combined with the negative fault-normal component, produced 2.8 ft of displacement; however, when combined with the positive fault-normal component, produced a much smaller displacement than that of set 5. Based on the rigid sliding block analyses, two rotated ground motions: set 1 motions (rotated 46 degrees with a negative fault-normal component) and set 5 motions (rotated 46 degrees with a positive fault-normal component) were used in the two-dimensional finite-element analyses (Reference 40).

The potential sliding masses and the node points of the computed acceleration time histories were used to develop average-acceleration time histories for each sliding mass. The seismic coefficient time histories were then double integrated to obtain earthquake-induced displacements for any specified yield acceleration. As mentioned before, the integration was made for the ground-motion amplitudes exceeding the yield acceleration in the positive direction only, and the resulting displacement was computed for potential sliding in the down-slope direction. The relationships between calculated displacement and yield acceleration, k_y, for the three potential sliding masses considered are presented in Reference 41, Figures 5 and 6, for input motion sets 1 and 5, respectively. The normalized relationships between calculated displacement and yield acceleration ration, k_y/k_{max}, for the three potential sliding masses considered are presented on Figures 7 and 8 of Reference 41, for input motions sets 1 and 5, respectively.

2.6.5.1.3.5 Results

The earthquake-induced down-slope displacements for the potential slip surfaces analyzed are summarized on Table 2.6-5. Computed permanent displacements using ground-motion set 1 as input range from about 3.1 ft, for sliding mass 1b, on the upper slope, to about 1.4 ft, for sliding mass 3c, on the lower slope. Computed displacements using ground-motion set 5 as input were lower, and ranged from 2.4 ft, for sliding mass 1b, to about 0.6 foot, for sliding mass 3c.

Sliding mass 1b (located in the upper portion of the slope) daylights at a horizontal distance of about 400 ft from the toe of the cutslope behind the pads. As mentioned above, the computed displacements for this sliding mass ranged between 2.4 ft and 3.1 ft. Sliding mass 2c (located in the middle of the slope) daylights about 100 ft from the toe of the cutslope. The computed displacements for this sliding mass ranged between 2.5 ft and 3 ft. The computed displacements for sliding mass 3c (located in the lower portion of the slope) ranged between 0.6 ft and 1.4 ft. Two additional potential sliding masses were analyzed in addition to 3c: sliding mass 3c-1, which daylights 70 ft beyond the north edge of the ISFSI pads, and sliding mass 3c-2, which daylights at the first bench on the cutslope behind the pads (Figure 2.6-56). The computed displacements for sliding mass 3c-1 ranged between 0.4 ft and 1.2 ft. For sliding mass 3c-2, the computed displacements ranged between 0.8 ft and 2.0 ft, depending on the input motion used in the analysis. Given the mitigation measures for the ISFSI (Section 4.2.1.1.9), none of the potential displacements indicated by any of the rock mass models would impact the ISFSI pads.

2.6.5.1.3.6 Estimating Displacements Based on Geologic Data

Potential slide mass displacement can be estimated by evaluating past performance of the hillslope above the ISFSI site. As described below, the topographic ridge upon which the ISFSI site is located has been stable for the past 500,000 years or more (Reference 37; Reference 6, p. 2-38). A geologic analysis of slope stability, therefore, provides insights into the minimum shear strength and lateral continuity of the clay beds used in the analysis and, hence, a check on the conservatism of the assumptions used to analyze the stability of the hillslope above the ISFSI site.

Geomorphic and geologic data from mapping and trenching in the ISFSI study area show no evidence of past movements of large rock masses on the slope above the ISFSI (Reference 37). Analysis of pre-construction aerial photographs shows no features indicative of such landslides: no arcuate scarps, no vegetational lineaments indicative of filled fissures, and no textural differences in the rock exposures or slopes indicative of a broken rock mass at the ISFSI study area. Similarly, the many trenches excavated into the slope, the tower access road cuts, the extensive outcrops exposed by the 1971 borrow cut, and the many borings exposed no tension cracks or fissure fills on the hillslope (References 43, 44 and 46, Data Reports A, B, and D). Open cracks or soil-filled fissures greater than 1 foot to 2 ft in width would be easily recognized across the slope, given the extensive rock exposure provided by the borrow cut. Therefore, it is reasonable to conclude that any cumulative displacement in the slope greater than 3 ft would have produced features that would be evident in rock slope. The absence of this evidence places a maximum threshold of 3 ft on the amount of cumulative slope displacement that could have occurred in the geologic past.

The hillslope at the ISFSI site is older than at least 500,000 years, because remnants of the Q_5 (430,000 years old) marine terrace are cut into the slope west of the ISFSI site (Reference 37; Reference 6, p. 2-18). Preservation of the terrace documents that the slope has had minimal erosion (a few tens of feet) since that time. Moreover, gradual reduction of the ridge by erosion at the ISFSI site would not destroy deep tension cracks or deep disruption of the rock mass; these features would be preserved as filled fractures and fissures, even as the slope is lowered.

The topographic ridge upon which the ISFSI site is located is presumed to have experienced strong ground shaking from numerous earthquakes on the Hosgri fault zone during the past several hundred thousand years. Studies for the LTSP (Reference 6) estimated a recurrence interval of 11,350 years for a magnitude 7.2 earthquake on the Hosgri fault zone. Assuming that deep cracks from rock mass movements during the past 400,000 years would have been preserved, approximately 35 to 40 large earthquakes have occurred during the past 400,000 years without causing significant (greater than 3 ft) cumulative slope displacement.

2.6.5.1.3.7 Assessment of Conservatism in Displacement Estimates

Because a major portion of the rock mass slide surfaces analyzed is along clay beds, an approximate analysis of the slope at its pre-borrow excavation configuration was conducted to assess the degree of conservatism associated with the assumptions used in the analysis, in particular, the lateral continuity and shear strength of the clay beds. The calculation consisted of extending the potential slide surfaces 1a and 1b (located in the upper part of the slope) to the pre-excavated ground surface, and varying the undrained strength of the clay bed until a yield acceleration corresponding to a displacement of 4 inches was calculated. Ground-motion sets 1 and 5, multiplied by 1.6 and rotated through the same angle as in the previous analysis (ø = 46 degrees) were used. Several combinations of the undrained strength parameters c and ø were considered in the analysis. The results indicate that the calculated undrained clay bed shear strength is significantly greater than the undrained shear-strength parameters developed from laboratory test data. It is reasonable to conclude, therefore, that the clay bed strength properties used in the analyses are conservative (that is, the clay beds are thin, with rock-to-rock contact through some of the length of the bed that increases the strength), and that the clay beds are more limited in lateral extent than was assumed in analysis.

2.6.5.2 Stability of Cutslopes

Construction of the ISFSI will involve preparing cutslopes along the southwestern, southeastern, and northeastern margins of the site (Figure 2.6-32). The stability of these cutslopes was evaluated using kinematic, pseudostatic, and dynamic analyses.

2.6.5.2.1 Kinematic Analysis

Three potential failure modes were identified for analysis of the cutslopes along the margins of the ISFSI site (Reference 38):

- planar sliding on a single discontinuity
- wedge sliding on the intersection of two discontinuities
- toppling of blocks

2.6.5.2.1.1 Method

Kinematic analyses, based on the collected fracture data, were performed for each of the three ISFSI site cutslopes: east cutslope (northeast), back cutslope (southeast), and west cutslope (southwest), proposed to be excavated at an inclination of 70 degrees. Discontinuity data from the trenches and outcrops in the area of each cutslope (Reference 48, Data Report F) were applied in the analysis (Figures 2.6-57, 2.6-58, and 2.6-59). Data from outcrops along Reservoir Road were applied in the analyses of the slope above the road (Figures 2.6-37 and 2.6-38).

Using the Markland procedure (Reference 32), discontinuities were analyzed for three modes of rock block failure. All kinematic analyses used a friction angle (ø) of 28 degrees to represent sliding resistance along dilated joints or discontinuities in the rock mass. This friction angle value represents a conservative estimate for rock friction, and was selected on the basis of laboratory direct-shear test data on borehole core joints, and estimation of in situ shear strength using the Barton-Choubey method (Figure 2.6-52). Discontinuities generally are 2- to 4-ft long, and locally up to 14-ft long (Reference 48, Data Report F).

There is a midslope bench in the back cut that slopes to the west. The lower western part of the cutslope below the slope drainage bench will vary in inclination from about 70 degrees to 10 degrees in the westward direction. A kinematic analysis was performed for the cutslope design in calculation GEO.DCPP.01.22 (PG&E Calculation 52.27.100.732), which evaluates the general kinematic slope stability and need for cutslope support for these areas of the lower bench cuts of varying cutslope heights and inclinations. In addition, the calculation looked at varying inclinations in the east cut and west cut in the event the lower portions of the cutslopes are modified during grading.

2.6.5.2.1.2 East Cutslope

Kinematic analyses of the east cutslope are shown on Figure 2.6-57. The analysis shows low potential for toppling failure, as only a few random discontinuities plot within this failure envelope. There is a moderate to high potential for planar sliding failure, as numerous discontinuities from discontinuity set 2, as well as some random discontinuities, plot within the planar sliding failure envelope. Potential also exists for wedge sliding along the intersection lines between discontinuity sets 1 and 2, and between sets 2 and 4; though these intersections plot very close to the failure envelope, these lines represent the average orientation of the set and there is a scatter of orientations around this mean. Thus, there is a moderate to high potential for planar sliding, and a moderate to high potential for wedge sliding failures in the east cutslope.

2.6.5.2.1.3 Back Cutslope

Kinematic analyses of the back cutslope are shown on Figure 2.6-58. The analysis shows low potential for toppling failure, as only a few random discontinuities plot within this failure envelope. Planar sliding failure represents a low to moderate potential, as a few discontinuities from sets 1 and 2, as well as a number of random discontinuities, plot within the planar sliding failure envelope. Potential exists for wedge sliding along the intersection line of discontinuity sets 2 and 3, whereas another intersection (1 and 3) plots outside but relatively close to the failure envelope and should be considered a potential hazard, given that these lines represent the average orientation of the set and that there is a scatter of orientations around this mean. Thus, there is a high potential for wedge failure and minor planar sliding failure in the back cutslope.

Reducing the rock friction angle value to a value appropriate to represent the strength of the bedding-parallel clay beds results in a larger failure envelope, and introduces the possibility of planar sliding failures along the clay beds in the back cutslope and in the hill above the ISFSI site. Static and dynamic modeling of potential sliding along clay beds is presented in Sections 2.6.5.1.2 and 2.6.5.1.3.

A portion of the back cutslope will be in friable dolomite. This material does not behave as a jointed rock mass but, rather, behaves as a stiff soil. The potential exists for slumps within this material.

2.6.5.2.1.4 West Cutslope

Analyses of the west cutslope are shown on Figure 2.6-59. The west cutslope shows a high potential for topple failure. The majority of discontinuity set 2, as well as some fractures from set 1, plot within the zone of potential failure for toppling. However, analyses of planar and wedge sliding failures show low and very low potential, respectively, for these modes of failure in the west cutslope, as very few discontinuities (and none belonging to any of the defined sets) fall within the failure envelope for planar sliding, and none of the discontinuity intersections fall within the failure envelope for wedge sliding failure. Thus, the failure mode for the west cutslope is topple failure. A portion of the southwest side of the ISFSI slope will be in a fill prism; therefore, the topple failure mode would not be applicable there.

2.6.5.2.1.5 Results

None of the three potential failure modes described above pose a threat to the ISFSI, because potential displacements will be mitigated using conventional methods and appropriate setback distances from the toe of cutslopes, as discussed in Section 4.2.1.1.9. The results of the kinematic analysis are valid for the higher cutslopes that result from sloping of the bench in the back cut.

2.6.5.2.2 Pseudostatic Analyses of Potential Wedge Slides

A pseudostatic seismic analysis of the wedges identified in the kinematic analysis was conducted to assess cutslope stability under seismic loads.

2.6.5.2.2.1 Geometry and Dimensions of Wedge Blocks

The size of potential wedge block failures in the ISFSI cutslope (Figure 2.6-32) will be controlled, in part, by the spacing, continuity, and shear strength of discontinuities in the rock mass. Both the dolomite (unit Tof_{b-1}) and sandstone (unit Tof_{b-2}) bedrock at the site are jointed and faulted. Joints and faults in friable dolomite and friable sandstone are less well developed and do not control the mechanical behavior of this rock; rather, strength of the friable rock is controlled primarily by the cementation properties of the rock.

The orientation of the joint sets varies somewhat across the site; however, field measurements of the discontinuities (Reference 48, Data Report F) document two primary, steeply dipping, joint sets: a west- to northwest-striking set, and a north-northwest- to north-striking set. The joints are continuous for about 1 foot to about 14 ft, and commonly die out or terminate at subhorizontal bedding contacts. Field observations from surface exposures and trenches show that the joints commonly are slightly open or dilated in the upper 4 ft, probably due to the stress unloading from the 1971 borrow excavation and surface weathering. Dilation of the joints reduces the shear strength of the discontinuity. To be conservative, the zone of near-surface dilation was assumed to extend to a depth of 20 ft on the ISFSI cutslope.

Joints in the dolomite typically are spaced about 1 ft to 3 ft apart, and divide the rock mass into blocks having an average dimension of 1 foot to 3 ft; typical maximum dimensions are about 14 ft (Reference 48, Data Report F, Table F-6). Twenty ft was conservatively assumed to be the maximum block size in the wedge block stability analysis. This dimension would allow for multiple-block wedges to form in the cutslope.

2.6.5.2.2.2 Method

Kinematic analyses (Section 2.6.5.2.1) show that the proposed east and back cutslopes along the southeast margin of the ISFSI pads have potential for wedge slides. The back cutslope would be the highest, and also has the least stable geometry with respect to rock mass discontinuities. Pseudostatic wedge analyses of these cutslopes were performed to evaluate the potential for shallow wedge slides along joints emerging on the cut faces through the zone of stress-relieved rock (Reference 39). Analyses were performed using SWEDGE (Reference 34) a computer program for the analysis of translational slip of surface wedges in rock slopes defined by two intersecting discontinuity (joint, fault, shear, or fracture) planes, a slope face, and an optional tension crack. The program performs analyses using two techniques: probabilistic analyses (probability of failure), and deterministic analyses (factor of safety). For probabilistic analyses, variation or uncertainty in discontinuity orientation and strength values can be accounted for, resulting in safety factor distribution and prediction of failure probability. For deterministic analyses, a factor of safety is calculated for a specified wedge geometry and a set of strength parameters.

Results from the kinematic analysis show that the most critical wedges are formed by intersections between steeply dipping, northwest-trending faults and joints that intersect at a high oblique angle. These fault/joint intersections plunge steeply to the northwest, and some could daylight on the proposed back cut. These wedge geometries were specifically modeled in the SWEDGE analyses. Planar sliding along low-angle clay beds is addressed in Section 2.6.5.1.3.

Probabilistic analyses were performed to evaluate the overall susceptibility of the slope to wedge failure, and to evaluate the sensitivity of failure to variations in material strength, geometry, and water conditions. Twenty-six separate model runs were performed using the probabilistic approach. Each probabilistic model run included 1,000 Monte Carlo iterations of input parameter variations to generate a probability distribution. After completing the probabilistic analyses, deterministic analyses were performed for the most critical modeled conditions in terms of probability of failure, and size and weight of wedge. Sixteen separate deterministic models were run that included variations in slope height and inclination, wedge geometry (with and without tension cracks), and degree of water saturation.

2.6.5.2.2.3 Rock Wedge Strength Parameters

Strength values derived from the Barton-Choubey method (Reference 27) (Figure 2.6-52) were used for the analyses of potential shallow rock wedge failures of rock blocks along existing discontinuities within the stress-relieved outermost rock zone directly behind the cutslope face. Cohesion was neglected. The friction angles selected and used in the probabilistic analyses ranged from 16 degrees (clay-coated faults) to 46 degrees (clay-free joints), and from about 26 degrees to 31 degrees, respectively, for the deterministic analyses.

2.6.5.2.2.4 Assumptions

The following assumptions and parameters were used for the pseudostatic probabilistic analysis:

- A horizontal midslope bench for the back cut
- Three 70-degree cutslope geometrics were analyzed for the back cutslope: (1) 20.5-ft-high, (2) 31.8-ft-high, and (3) 52.3-ft-high (inclined at 47 degrees) cutslopes were used to model potential failures. (Figure 2.6-60)

The following assumptions and parameters were used for the pseudostatic deterministic analysis:

- The bounding design wedge is shown in Figure 2.6-61. It occurs in the upper cutslope riser. The wedge is 46.5 ft high (the maximum height of the cutslope, plus a 6-ft buffer) and 20 ft thick. The tension crack is located at an uphill drainage trench.
- The maximum friction angle of the joint discontinuity sets for the cutslopes are conservatively taken at 26.5 degrees (the median friction angle for faults at the site). The value presumes that joints intersecting faults may have undergone sympathetic slip and developed fault-like characteristics.
- Each slope was evaluated with and without tension cracks, for example, in the case of the back cutslope, tension cracks were located at distances of 1.6 ft and 23 ft back from the crest of the slope. These distances are reasonable for a slope model, because the fractures at the ISFSI site have spacings of up to several feet, and the cutslope bench is 25 ft wide. One set of tension cracks (at 23 ft) specifically models the potential for tension cracks to develop along a backfilled trench for a drainage pipe at the back of the intermediate bench.
- Analyses were performed for each cutslope configuration using: (1) a horizontal (out-of-slope) pseudostatic seismic coefficient of 0.5 g, and (2) dry and partially saturated rock mass (water levels at one-half the height of the slope). The value of 0.5 g (Reference 68) was derived using the procedure described by Ashford and Sitar (Reference 35), and is approximately two-thirds of the peak horizontal acceleration of 0.83 g from the LTSP spectra shown in Figure 2.6-43. This level of reduction has been shown to be appropriate for pseudostatic analyses of slopes (Reference 35).

2.6.5.2.2.5 Results

The results of the pseudostatic probabilistic SWEDGE analysis for the back cutslope and the east cutslope are presented in Tables 2.6-6 and 2.6-7, respectively. Results of the deterministic analyses for these cutslopes are presented in Table 2.6-8. The probabilistic analyses show that rock wedges in the modeled cutslopes (Figure 2.6-60) have a low probability of failure in a dry condition. The probability of failure increases significantly with partial saturation of the slope and the addition of seismic force. The largest predicted wedge, with a factor of safety less than 1.0, weighs 4,475 kips and has an estimated face area of 2,649 square ft (Table 2.6-6).

Deterministic analyses were performed to calculate support forces required to restrain the wedges and achieve a factor of safety of 1.3 under seismic loading conditions (Table 2.6-8). The calculated total support force to stabilize the largest predicted wedge to a factor of safety of 1.3 is 1,881 kips. For an assumed anchor spacing of 5 ft by 5 ft, this force translates to 32 kips per anchor (Table 2.6-8). The design of slope reinforcement to prevent wedges from displacing is described in Section 4.2.1.1.9 (Reference 69).

2.6.5.3 Slope Stability at CTF Site

In a previous submittal examining the stability of the slope behind DCPP Unit 1 (Reference 9, p. 30-36), it was shown that displacements along the interface between colluvial and terrace deposits within the underlying bedrock would be limited. The results of this analysis also indicate that the farthest extent of these estimated displacements is at the uppermost edge of the colluvium/bedrock interface, which is more than 100 ft west of the CTF (Figure 2.6-7), and similar to the relationship shown in cross sections B-B''' and L-L' (Figures 2.6-11 and 2.6-19). Therefore, slope-related displacements at the CTF site are estimated to be nil.

2.6.5.4 Slope Stability Along the Transport Route

2.6.5.4.1 Static Stability

As discussed in Section 2.6.1.12.3, the Patton Cove landslide is more than 100 ft from the transport route, and it is not likely to encroach headward to where it would affect the route.

Small debris flows (up to 3 ft deep on the road) could impact the roadway as they issue from the swales on the steep slopes above the road (Section 2.6.1.12.3). These debris flows occur infrequently during or shortly following severe rainstorms (Reference 7), and are relatively easy to clear from the road.

Kinematic analyses of the stability of the slope above the transport route are shown on Figures 2.6-37 and 2.6-38. (Reference 38, Figures 7 and 8). The north-trending slope (station 43+00 to 46+00) shows moderate potential for toppling failure, as a large portion of set 1 plots within this failure envelope. There is low potential of planar sliding failure, and very low potential for wedge sliding failure. Due to the very low inclination of the northwest-trending slope (station 35+00 to 43+00), this slope shows low potential for all three failure modes. Thus, the only potentially significant failure mode is for small topple failures along the transport route cutslopes.

Reference 76 provided an additional static stability assessment for portions of the transport route on rock. Where the transport route is founded on rock begins at approximately Stations 34+50 and continues uphill to the CTF at station 53+50. Along the southern part of this section (station 34+50 to 46+10), the rock beddings dip into the slope and thus makes it kinematically unlikely for slope movements to daylight at the slope face. Along the northern section (station 46+10 to 53+50), the rock beddings show a gentle out-of-slope dip and thus it is kinematically feasible to have out-of-slope movement along possible clay beds that parallel the bedding.

Reference 76 (PG&E Response to NRC Request 5) concluded the following:

The northern section of the transport route on rock, between stations 46+10 and 53+50, has bedding that dips gently out of the slope. Potential rock mass slide models on clay bed or rock discontinuities are kinematically feasible. Potential slide mass models were developed based on conservative interpretations of geological information from surfacing mapping, trenching, and exploratory boring as shown on Section M-M’. Static slope stability analysis for the two rock slide models indicates that the minimum static factor of safety is 2.07, which is higher than the 1.5 that is typically required for static slope stability. This demonstrates the slope has ample safety factors against static slope failure.

The southern section of this alignment, between stations 34+50 and 46+10, has rock bedding that dips into the hillslope making slope failure on bedding not possible. Kinematic analysis of joints along the transport route also shows that failure of the bedrock below the transport route is not possible.

Based on the above evaluation and documentation provided in References 74 and 76 (Attachment 5-1), it is concluded that the portions of transport route on rock have adequate static factor of safety against rock mass sliding on clay beds.

2.6.5.4.2 Dynamic Stability and Displacements

Stability analyses using the ILP ground motions (Section 2.6.2.5) were performed on the hillslope behind Unit 2 using cross section L-L' (Figure 2.6-19). Borings drilled during investigations for the power block along the slope provided data for modeling the slope. Reference 76 (PG&E Response to NRC Request 6) provided additional information on material properties used for the transport route stability evaluations and PG&E submitted Revision 3 of Reference 74 that provides technical bases for the material properties. The results of these analyses indicate the bedrock slope and the transport route that crosses it are expected to undergo only minor displacements of about 1.0 ft or less during the possible occurrence of the ILP ground motions (References 50, 51, and 52).

An additional location, shown on cross section D-D' (Figure 2.6-16), along the transport route also was modeled, and the responses to the ILP ground-motions were assessed in a similar manner. Results from this analysis show that this location also is expected to undergo only minor displacements of about l.0 ft or less.

PG&E has revised the dynamic slope stability calculations (References 74, 42, and 71) for the transport route incorporating the inertial mass of the transporter, a new section of the transport route with two slide mass models, and revised seismic coefficient time histories for the three slide masses in sections of the transport route underlain by surficial deposits. The transport route slopes' estimated displacement magnitude of 1.5 ft is smaller than that computed for the ISFSI pad's slope (Reference 41) and is not indicative of an unstable slope. Thus, the transport route slope remains stable during and after a design basis earthquake. Further details of the revised stability analyses for the transport route are provided in Reference 76 (PG&E Response to NRC Request 7).

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TABLE 2.1-1

POPULATION TRENDS OF THE STATE OF CALIFORNIA
AND OF SAN LUIS OBISPO AND SANTA BARBARA COUNTIES

Year	State of California	San Luis Obispo County	Santa Barbara County	Notes
1940	6,907,387	33,246	70,555	(a)
1950	10,586,233	51,417	98,220	(a)
1960	15,717,204	81,044	168,962	(a)
1970	19,953,134	105,690	264,324	(a)
1980	23,668,562	155,345	298,660	(a)
1990	29,760,021	217,162	369,608	(a)
2000	33,871,648	246,681	399,347	(a)
2005	40,262,400	323,100	467,700	(b)
2025	48,626,052	426,812	603,966	(c)

Notes:
(a) U.S. Bureau of the Census
(b) State of California Department of Finance (June 2001)
(c) State of California Department of Finance Data Files (March 16, 2000)

TABLE 2.1-2

GROWTH OF PRINCIPAL COMMUNITIES WITHIN
50 MILES OF ISFSI SITE

Community	Population (1960 Census)	Population (1970 Census)	Population (1980 Census)	Population (1990 Census)	Population 2000 Census
Arroyo Grande	3,291	7,454	10,350	21,992	15,851
Atascadero	5,983	10,290	15,930	27,720	26,411
Grover Beach	5,210	5,939	8,827	11,790	13,067
Guadalupe	2,614	3,145	3,629	6,464	5,659
Lompoc	14,415	25,284	26,267	49,960	41,103
Morro Bay	3,692	7,109	9,064	10,457	10,350
Paso Robles	6,617	7,168	9,163	29,255	24,297
Pismo/Shell Beach	1,762	4,043	5,364	7,474	8,551
San Luis Obispo	20,437	28,036	34,253	56,614	44,174
Santa Maria	20,027	32,749	39,685	60,187	77,423

TABLE 2.1-3

POPULATION CENTERS OF 1,000 OR MORE
WITHIN 50 MILES OF ISFSI SITE

Community	County	Distance and Direction From the Site	Population (1970 Census)	Population (1980 Census)	Population (1990 Census)	Population (2000 Census)
Baywood-Los Osos	San Luis Obispo	8 miles N	3,487	10,933	15,290	14,351
Morro Bay	San Luis Obispo	10 miles N	7,109	9,064	12,949	10,350
San Luis Obispo	San Luis Obispo	12 miles ENE	28,036	34,253	51,173	44,174
Pismo Beach	San Luis Obispo	13 miles ESE	4,043	5,364	7,699	8,551
Grover City	San Luis Obispo	14 miles ESE	5,939	8,827	11,656	13,067
Oceano	San Luis Obispo	15 miles ESE	2,564	4,478	6,169	7,260
Arroyo Grande	San Luis Obispo	17 miles ESE	7,454	10,350	14,378	15,851
Cayucos	San Luis Obispo	17 miles N	1,772	2,301	2,960	2,943
Atascadero	San Luis Obispo	21 miles NNE	10,290	15,930	23,138	26,411
Guadalupe	Santa Barbara	23 miles SE	3,145	3,629	5,479	5,659
Nipomo	San Luis Obispo	24 miles ESE	3,642	5,247	7,109	12,626
Cambria	San Luis Obispo	28 miles NNW	1,716	3,061	5,382	6,232
Santa Maria	Santa Barbara	29 miles SE	32,749	39,685	61,284	77,423
Paso Robles	San Luis Obispo	30 miles NNE	7,168	9,163	18,583	24,297
Orcutt	Santa Barbara	33 miles SE	8,500	1,469	----	28,830
Vandenberg	Santa Barbara	35 miles SSE	13,193	13,975	----	11,953
Lompoc	Santa Barbara	45 miles SSE	25,284	26,267	37,649	41,103
		Total	180,793	203,996	280,898	351,081

TABLE 2.1-4

TRANSIENT POPULATION AT RECREATION AREAS WITHIN 50 MILES OF ISFSI SITE

Name	Visitor Days
State Parks [a]	
Cayucos State Beach	698,000
Hearst San Simeon State Historical Monument	795,000
Montana de Oro State Park	683,000
Morro Bay State Park	1,129,000
Morro Strand State Beach	129,000
Pismo State Beach	1,297,000
San Simeon State Park	696,000
W. R. Hearst Memorial State Beach	213,000
County and Local Parks [b]	
Lake Nacimiento	345,000
San Antonio Reservoir	361,000
Avila Beach	800,000
Cambria	15,000
Cayucos Beach	918,000
Cuesta	67,000
Lopez Recreation Area	379,000
Nipomo	168,000
Oceano	95,000
San Miguel	54,000
Santa Margarita Lake	169,000
Shamel	130,000
Templeton	99,000
Los Alamos Park	45,000
Miguelito Park	36,000
Ocean Park	105,000
Rancho Guadalupe Dunes Park	48,000
Waller	450,000
Atascadero Lake	300,000

Name	Visitor Days
Los Padres National Forest [c]	
Agua Escondido	700
American Canyon	800
Balm of Gilead	200
Brookshire Springs	1,600
Buckeye	200
Cerro Alto	15,600
French	200
Frus	700
Hi Mountain	4,800
Horseshoe Springs	1,400
Indians	600
Kerry Canyon	300
La Panza	4,400
Lazy Camp	500
Miranda Pine	2,300
Navajo	2,800
Pine Flat	300
Pine Springs	400
Plowshare Springs	300
Queen Bee	2,200
Stony Creek	1,100
Sulphur Pot	1,000
Upper Lopez	600
Wagon Flat	2,200

(a) California Department of Parks and Recreation (July 1998 through June 1999).
(b) County Park Departments:
Monterey and Santa Barbara Counties (July 1999 through June 2000).
San Luis Obispo County (July 1998 through June 1999).
(c) Los Padres National Forest (July 1971 through June 1972. Current data is no longer compiled).

TABLE 2.3-1

MONTHLY AVERAGE TEMPERATURES
FOR SAN LUIS OBISPO
(1948 to 2000) [a]

Month	Temp (ºF)	Month	Temp (ºF)
Jan	52.2	Jul	65.2
Feb	54.1	Aug	66.0
Mar	54.6	Sep	65.8
Apr	56.9	Oct	63.3
May	58.9	Nov	58.2
Jun	62.4	Dec	53.3

[a] Information from Western Regional Climate Center

TABLE 2.6-1

SOIL AND ROCK TEST PROGRAM

Type of Properties	Soil		Rock	
	Tests Conducted	Reference	Tests Conducted	Reference
Basic properties	Classification, identification, unit weight, saturation	References 44 and 49, Data Report B and Data Report G	Classification, identification, JRC, (mi)	References 44 and 50, Data Report B and Data Report H
Strength, deformation (static)	Drained, undrained triaxial strength, direct shear	Reference 49 Data Report G	Drained and undrained triaxial strength, unconfined compression, direct shear, Poisson's ratio, Young's modulus	Reference 51 Data Report I
Strength, deformation (dynamic)	Triaxial, drained, undrained strain vs. damping, strain vs. shear modulus	Reference 49 Data Report G		
Field and in situ properties			Field discontinuity, shear wave and compression wave velocity, Poisson's ratio, Young's modulus	References 48 and 45 Data Report F and Data Report C

TABLE 2.6-2

SUMMARY OF SLOPE STABILITY ANALYSES

SLOPE	TYPE OF ANALYSIS						
	Kinematic	**Static**	**Pseudo-Static**	**Seismic Displacement**	**Displacement Historical/ Back Calculation**	**Results**	**Action**
Hillslope		X	X	X	X	• Stable under static and dynamic conditions • Less than 3 feet seismic displacement	— —
Pad Cutslope	X	X	X	X		• Potential for small wedge slippage under partial saturation and/or seismic loading	Mitigation
CTF Slope	X	X		X		• Stable under static and dynamic conditions	—
Transport Route Slope		X		X		• Stable under static and dynamic conditions • Small seismic displacement (less than 1 foot)	— —

TABLE 2.6-3

FACTORS OF SAFETY AND YIELD ACCELERATIONS COMPUTED FOR POTENTIAL SLIDING MASSES

Slide Mass Analyzed	Static Factor of Safety	Yield Acceleration, k_y (g)
1a	2.55	0.28
1b	1.62	0.20
2a	2.55	0.31
2b	2.16	0.24
2c	2.18	0.19
3a	2.86	0.44
3b	2.70	0.39
3c	2.26	0.25
3c-1	2.38	0.28
3c-2	2.28	0.23

TABLE 2.6-4

DOWN SLOPE DISPLACEMENT CALCULATED BASED ON ROTATED INPUT MOTIONS ALONG CROSS SECTION I-I'
(DISPLACEMENT UNIT: FEET; YIELD ACCELERATION: 0.2 G)

Set No.	Description	Polarity	K_y = 0.20		
			$I\text{-}I_{36}$	$I\text{-}I_{46}$	$I\text{-}I_{26}$
Set 1	Lucerne	FN-	2.9	3.3	2.5
		FN+	1.4	1.4	1.5
Set 2a	Yarimca	FN-	2.4	2.8	1.8
		FN+	1.2	1.4	1.1
Set 3	LGPC	FN-	2.5	2.8	2.3
		FN+	1.3	1.2	1.4
Set 5	El Centro	FN-	2.2	2.6	1.8
		FN+	2.4	2.8	2.1
Set 6	Saratoga	FN-	0.9	1.1	0.8
		FN+	0.9	1.0	0.8

TABLE 2.6-5

COMPUTED DOWN-SLOPE DISPLACEMENTS
USING SET 1 AND SET 5 INPUT MOTIONS

Sliding Mass Location	Input Motion	Yield Acceleration, k_y, (g)	Peak Seismic Coefficient, k_{max}, (g)	Down-Slope Displacement (feet)
1b	Set 1	0.20	0.98	3.1
2c	Set 1	0.19	0.89	3.1
3c	Set 1	0.25	0.81	1.4
3c-1	Set 1	0.28	0.80	1.2
3c-2	Set 1	0.23	0.81	2.0
1b	Set 5	0.20	0.75	2.4
2c	Set 5	0.19	0.68	2.3
3c	Set 5	0.25	0.61	0.6
3c-1	Set 5	0.28	0.61	0.4
3c-2	Set 5	0.23	0.62	0.8

TABLE 2.6-6

PSEUDOSTATIC PROBABILISTIC SWEDGE ANALYSES OF ISFSI BACK CUTSLOPE

Run	Cut[1] Height (ft)	Discontinuity[2] A	Discontinuity[2] B	Mean[3] Friction Angle	Tension[4] Crack Distance (ft)	Seismic[5] Force (g)	Water[6] Unit Weight (kips*/ft^3)	Probability of Failure	Factor of Safety	Wedge Weight (kips*)	Wedge Face Area (ft^2)
Back cut P1R	31.8	69/220 (2)	88/12 (3)	26.5 (A/B)	None	None	None	0.036	1.39	40.1	101.8
Back cut P2R	31.8	69/220 (2)	88/12 (3)	26.5 (A/B)	3.3	None	None	0.007	1.39	25.1	101.8
Back cut P3R	31.8	69/220 (2)	88/12 (3)	26.5 (A/B)	None	None	0.031	0.978	0.27	40.1	101.8
Back cut P4R	31.8	69/220 (2)	88/12 (3)	26.5 (A/B)	None	0.50	None	1.0	0.49	40.1	101.8
Back cut P5R	31.8	69/220 (2)	88/12 (3)	26.5 (A/B)	None	0.50	0.031	1.0	0	40.1	101.8
Back cut P6R	31.8	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	None	None	None	0	2.74	11,991.8	1059.9
Back cut P7R	31.8	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	11.5	None	None	0	2.74	915.9	1059.9
Back cut P8R	31.8	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	23.0	None	None	0	2.74	1783.8	1059.9
Back cut P9R	31.8	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	23.0	None	0.031	0	1.44	1783.8	1059.9
Back cut P10R	31.8	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	23.0	0.50	None	0.90	0.92	1783.8	1059.9
Back cut P11R-R	31.8	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	23.0	0.50	0.031	1.0	0.62	1783.8	1059.9
Back cut P12R	31.8	75/12 (3)	75/261 (1)	26.5 (A) 30.5 (B)	None	None	None	1.0	0.43	10.8	77.5
Back cut P13R	31.8	75/12 (3)	75/261 (1)	26.5 (A) 30.5 (B)	None	0.50	0.031	1.0	0	10.8	77.5
Back cut P14R	52.3	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	None	None	None	0	2.74	21,836.2	2649.1
Back cut P15R	52.3	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	11.5	None	None	0	2.74	3243.9	2649.1
Back cut P16R	52.3	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	23.0	None	None	0	2.74	4474.6	2649.1
Back cut P17R	52.3	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	23.0	0.50	0.031	1.0	0.63	4474.6	2649.1
Back cut P18R	20.5	69/220 (2)	88/12 (3)	26.5 (A/B)	4.9	0.50	0.031	1.0	0.42	9.7	42.2
Back cut P19R	20.5	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	4.9	0.50	0.031	1.0	0.71	1,751.6	1,059.9

TABLE 2.6-6

*1 kip = 1000 pounds

[1]Cut height geometry from PG&E/Enercon Drawing PGE-009-SK-001, 9/12/01

[2]Mean dip and dip direction of intersecting joints (set number indicated in parentheses) that were identified by kinematic analyses in Reference 38, as forming potential wedges. Geometry of discontinuity is defined by the dip/dip direction convention. (Reference 39, Table 23-1). Numbers in brackets refer to Joint Set identification (Table 23-1).

[3]Mean rock discontinuity friction angle determined by Barton-Choubey method.

[4]Tension crack distance is the distance between the top of the wedge block crest and tension crack location, measured along strike of discontinuity A. Wedges modeled in Runs P6-P11 and P14-P17 consisted of unrealistically long, narrow wedges when tension cracks were not included. Final runs, therefore, include a tension crack at 23 ft behind slope face.

[(]Seismic force recommended for pseudostatic wedge analyses.

[6]Water unit weight of 0.031 kips/ft^3 approximates a condition in which water collects halfway up wedge-bounding discontinuities.

TABLE 2.6-7

PSEUDOSTATIC PROBABILISTIC SWEDGE ANALYSES OF ISFSI EAST CUTSLOPE

Run	Cut Height[1] (ft)	Discontinuity[2] A	Discontinuity[2] B	Mean Friction Angle[3] (ø)	Tension Crack Distances[4] (ft)	Seismic Force[5] (g)	Water Unit Weight[6] (kips/ft^3)*	Probability of Failure	Factor of Safety	Wedge Weight (kips)*	Wedge Face Area (ft^2)
East cut P1	23.3	76/08 (4)	67/239 (2)	35.0 (A) 36.0 (B)	None	None	None	0.20	1.08	33.96	446.0
East cut P2	23.3	76/08 (4)	67/239 (2)	35.0 (A) 36.0 (B)	1.64	None	None	0.12	1.08	33.96	446.0
East cut P3	23.3	76/08 (4)	67/239 (2)	35.0 (A) 36.0 (B)	None	None	0.031	0.31	1.02	33.96	446.0
East cut P4	23.3	76/08 (4)	67/239 (2)	35.0 (A) 36.0 (B)	None	0.50	None	1.0	0.65	33.96	446.0
East cut P5	23.3	76/08 (4)	67/239 (2)	35.0 (A) 36.0 (B)	None	0.50	0.031	1.0	0.54	33.96	446.0
East cut P6	23.3	88/98 (1)	67/239 (2)	36.0 (A) 36.0 (B)	None	None	None	0.97	0.31	23.81	469.8
East cut P7	23.3	88/98 (1)	67/239 (2)	36.0 (A) 36.0 (B)	None	0.50	0.031	0.99	0	23.81	469.8

*1 kip = 1000 pounds

1 Cut height geometry from PG&E/Enercon drawing, PGE-009-SK-001, 9/12/01.

2 Mean dip and dip direction of intersecting joints (set number indicated in parentheses) that were identified by kinematic analyses in Reference 38, as forming potential wedges. Geometry of discontinuity is defined by the dip/dip convention. (Reference 39, Table 23-1). Numbers in brackets refer to Joint Set identification (Table 23-1).

3 Mean rock discontinuity friction angle determined by Barton-Choubey method.

4 Tension crack distance is the distance between the top of the wedge block crest and tension crack location, measured along strike of discontinuity A.

5 Seismic force recommended for pseudostatic wedge analyses.

6 Water pressure of 0.031 kips/ft^3 approximates a condition in which water collects halfway up wedge-bounding discontinuities.

TABLE 2.6-8

PSEUDOSTATIC DETERMINISTIC SWEDGE ANALYSES OF ISFSI BACK CUTSLOPE AND EAST CUTSLOPE

Run	Cut[1] Height (ft)	Discontinuity[2] A	Discontinuity[2] B	Mean[3] Friction Angle	Tension[4] Crack Distance (ft)	Seismic[5] Force (g)	Water[6] Unit Weight (kips*/ft^3)	Bolt[7] Force (kips*)	Factor of Safety	Wedge Weight (kips*)	Wedge |Face Area (ft^2)	Penetration[8] Anchor Length (ft)	Per Anchor[9] Force (kips*)
Back cut D1RR	31.8	69/220 (2)	88/12 (3)	26.5 (A/B)	None	0.5	0.031	None	0	40.1	101.8		
Back cut D2R	31.8	69/220 (2)	88/12 (3)	26.5 (A/B)	None	0.5	0.031	41.8	1.39	40.1	101.8	7	21.8
Back cut D3R	31.8	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	23.0	0.5	0.031	None	0.62	1783.8	1059.9		
Back cut D4R	31.8	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	23.0	0.5	0.031	796.4	1.30	1783.8	1059.9	13	40.0
Back cut D4R, Rev. 2-1	46.5	88/12 (3)	24/232 (4)	16 (A) 26.5 (B)	37.4	0.5	0.062	None	0.071	6189.6	2263.3		
Back cut D4R, Rev. 2-2	46.5	88/12 (3)	24/232 (4)	16 (A) 26.5 (B)	37.4	0.5	0.062	2136.0	1.30	6189.6	2263.6	20	50
Back cut D7R	20.5	69/220 (2)	88/12 (3)	26.5 (A/B)	4.92	0.5	0.031	-	0.27	10.12	41.94		
Back cut D8R	20.5	69/220 (2)	88/12 (3)	26.5 (A/B)	4.92	0.5	0.031	8.8	1.67	10.12	41.94	4	14.0
Back cut D9R	20.5	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	20.0	0.5	0.031	-	0.76	596.2	440.1		
Back cut D10R	20.5	88/12 (3)	24/232 (4)	26.5 (A) 30.5 (B)	20.0	0.5	0.031	189.2	1.31	596.2	440.1	16	22.9
East cut D1R	23.3	76/08 (4)	67/239 (2)	35.0 (A) 36.0 (B)	None	None	None	None	1.08	33.96	446.0		
East cut D2R	23.3	76/08 (4)	67/239 (2)	35.0 (A) 36.0 (B)	None	0.5	0.031	None	0.54	33.97	446.0		
East cut D3R	23.3	76/08 (4)	67/239 (2)	35.0 (A) 36.0 (B)	None	0.5	0.031	81.6	1.34	33.98	446.0	3	9.7
East cut D3R Rev. 2	23.3	76/08 (4)	67/239 (2)	16.0 (A) 26.5 (B)	None	0.5	0.062	16.2	1.34	87.8	1176	2	4.0
East cut D4R	23.3	88/98 (1)	67/239 (2)	36.0 (A/B)	None	None	None	None	0.31	23.81	469.8		
East cut D5R	23.3	88/98 (1)	67/239 (2)	36.0 (A/B)	None	0.5	0.031	None	0	23.81	469.8		
East cut D6R	23.3	88/98 (1)	67/239 (2)	36.0 (A/B)	None	0.5	0.032	83.8	1.43	23.81	469.8	3	9.5

*1 kip = 1000 pounds

[1] Cut heights are from Calculation GEO.DCPP.01.23.

[2] Mean dip and dip direction of intersecting joints (set number indicated in parentheses) that were identified by kinematic analyses in Reference 38 as forming potential wedge. Geometry of discontinuity is defined by the dip/dip direction convention. Refer to Table 23-1. Numbers in brackets refer to Joint Set identification in Table 23-1, Reference 38.

[3] Mean rock discontinuity friction angle determined by Barton Equation as developed in Reference 64.

TABLE 2.6-8

4 Tension crack distance is the distance between the top of the wedge block crest and tension crack location measured along strike of discontinuity A. Wedges modeled in runs D3-D6 were unrealistically long and narrow when tension cracks were not included. Runs therefore include tension cracks specified in Calculation GEO.DCPP.01.23.

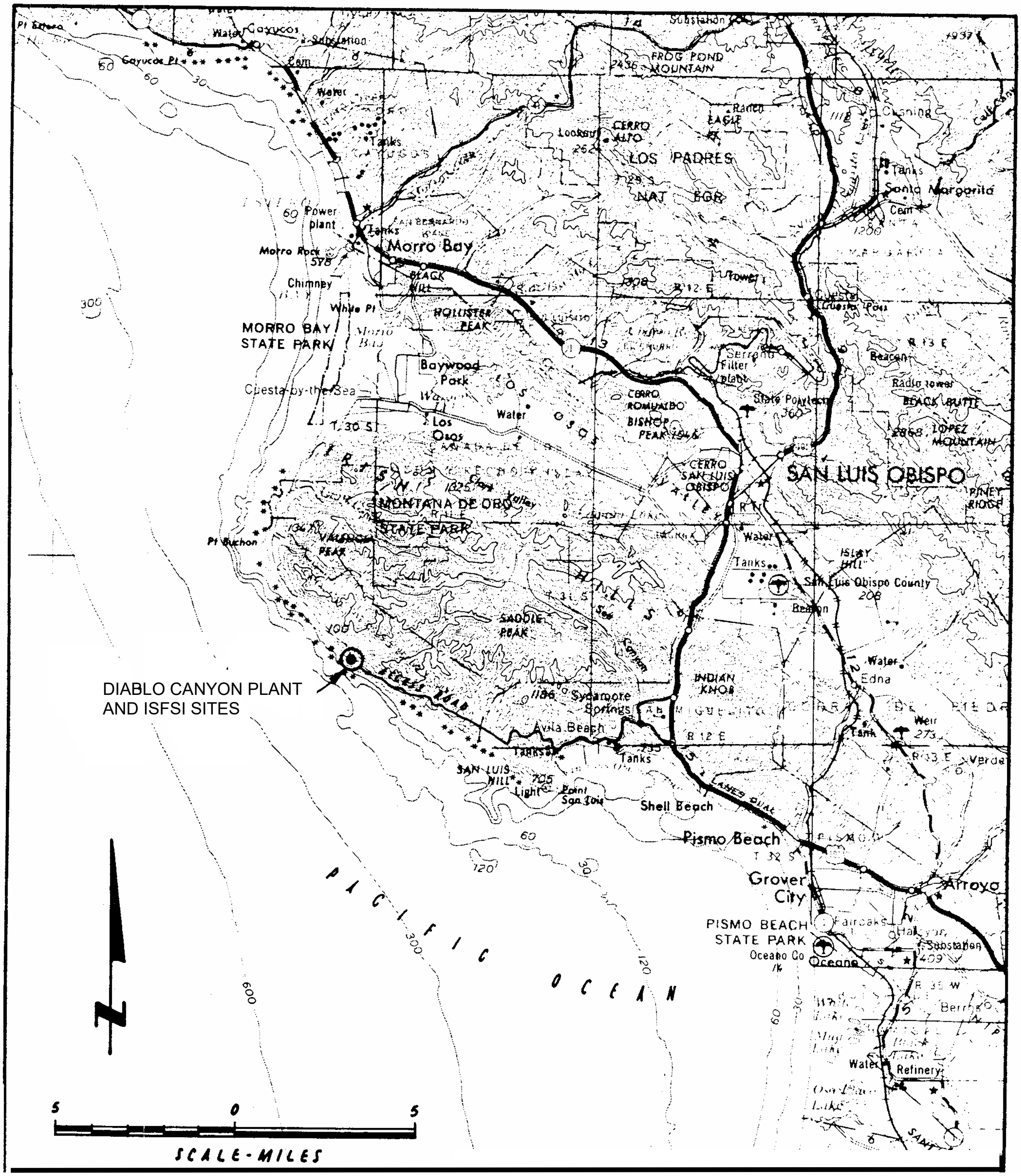
5 Seismic force recommended for pseudostatic wedge analyses as defined in Reference 68.

6 Water pressure input using SWEDGE v3.06 assumes a fissures filled completely with water (a fluid unit weight of 0.031 kips/ft^3). Water pressure input using SWEDGE v4.078 assumes a fissures filled halfway with water (fluid unit weight of 0.062 kips/ft^3).

7 Total force required to stabilize block to the listed factor of safety.

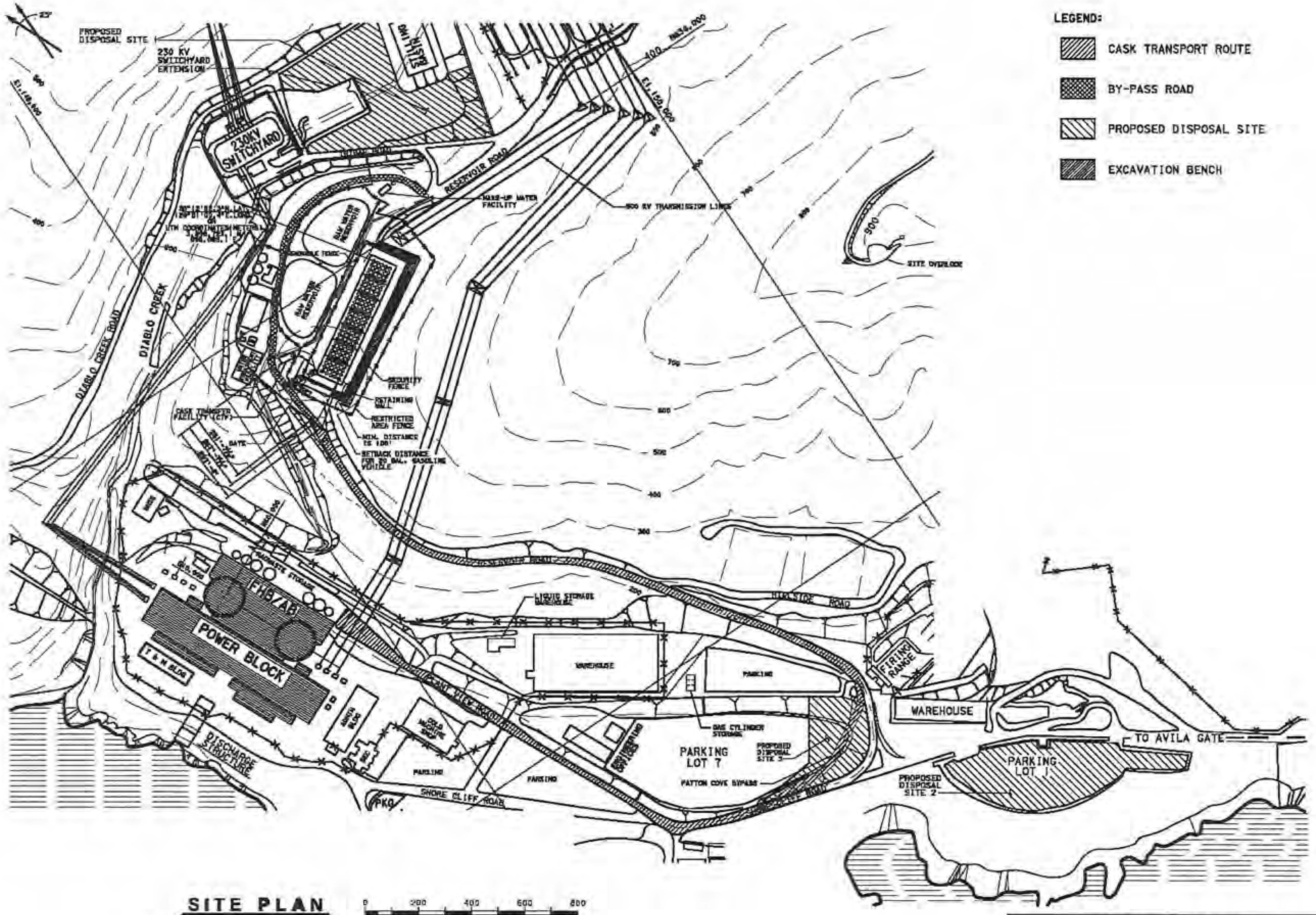
8 Length of anchor in meters required to penetrate modeled wedge sliding plane (free-stressing length), assuming an anchor inclination of 15º below horizontal, and plunge direction perpendicular to slope face. Additional length is required to provide anchor anchorage and capacity in sound rock behind the failure wedge, and total bolt lengths therefore are greater.

9 Per anchor force calculated by dividing wedge face area by 50% to account for wedge margins that are not suitable for providing anchor, and then dividing this value by the required anchor force, and assuming one anchor per 22.6 ft^2, which represents an anchor pattern spacing of 5.0 feet, vertical and horizontal, projected onto a 70-degree cutslope face.



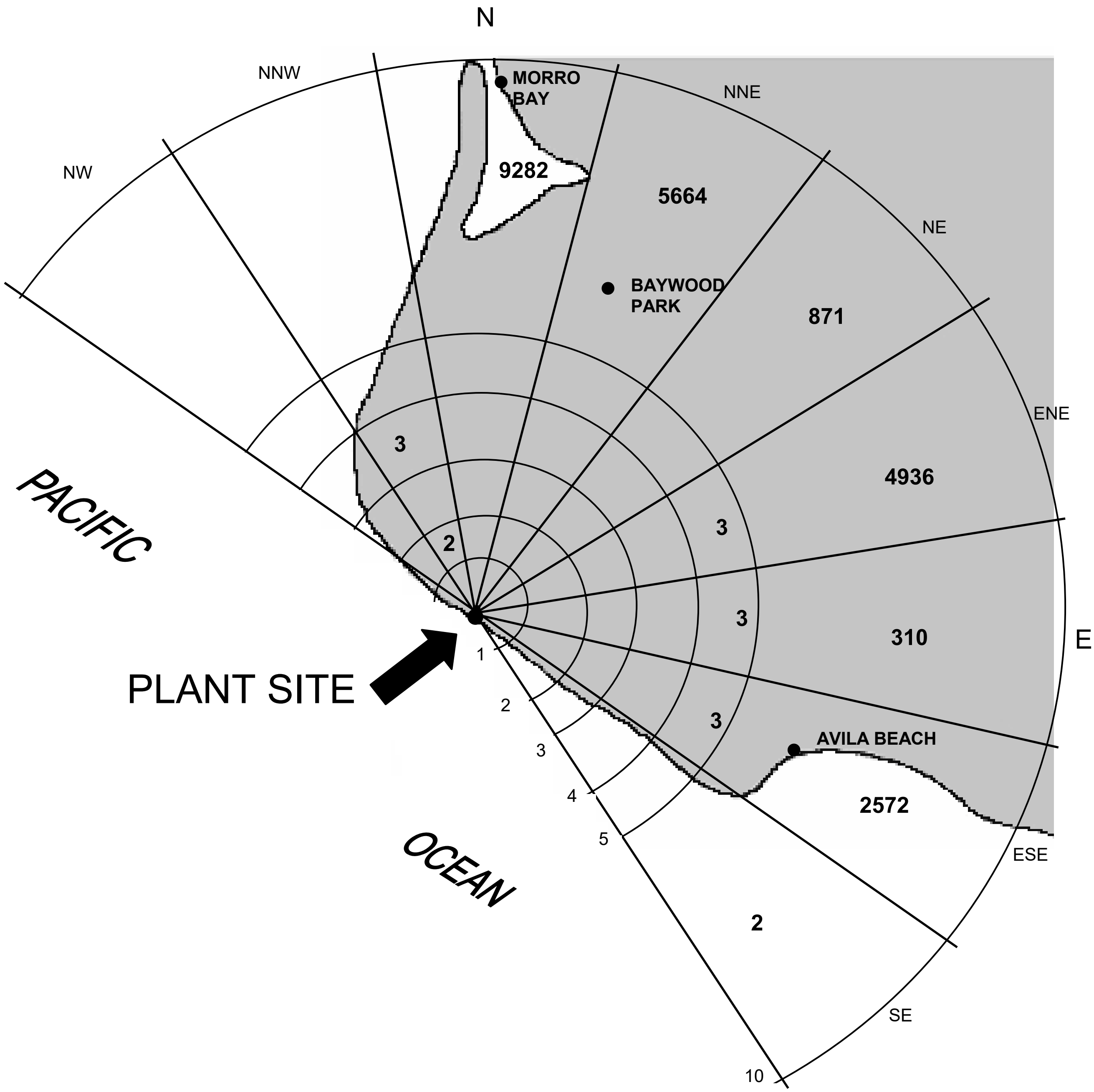
FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.1-1 SITE LOCATION MAP

Revision 0 June 2004



FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.1-2 **SITE PLAN**

Revision 2 June 2008



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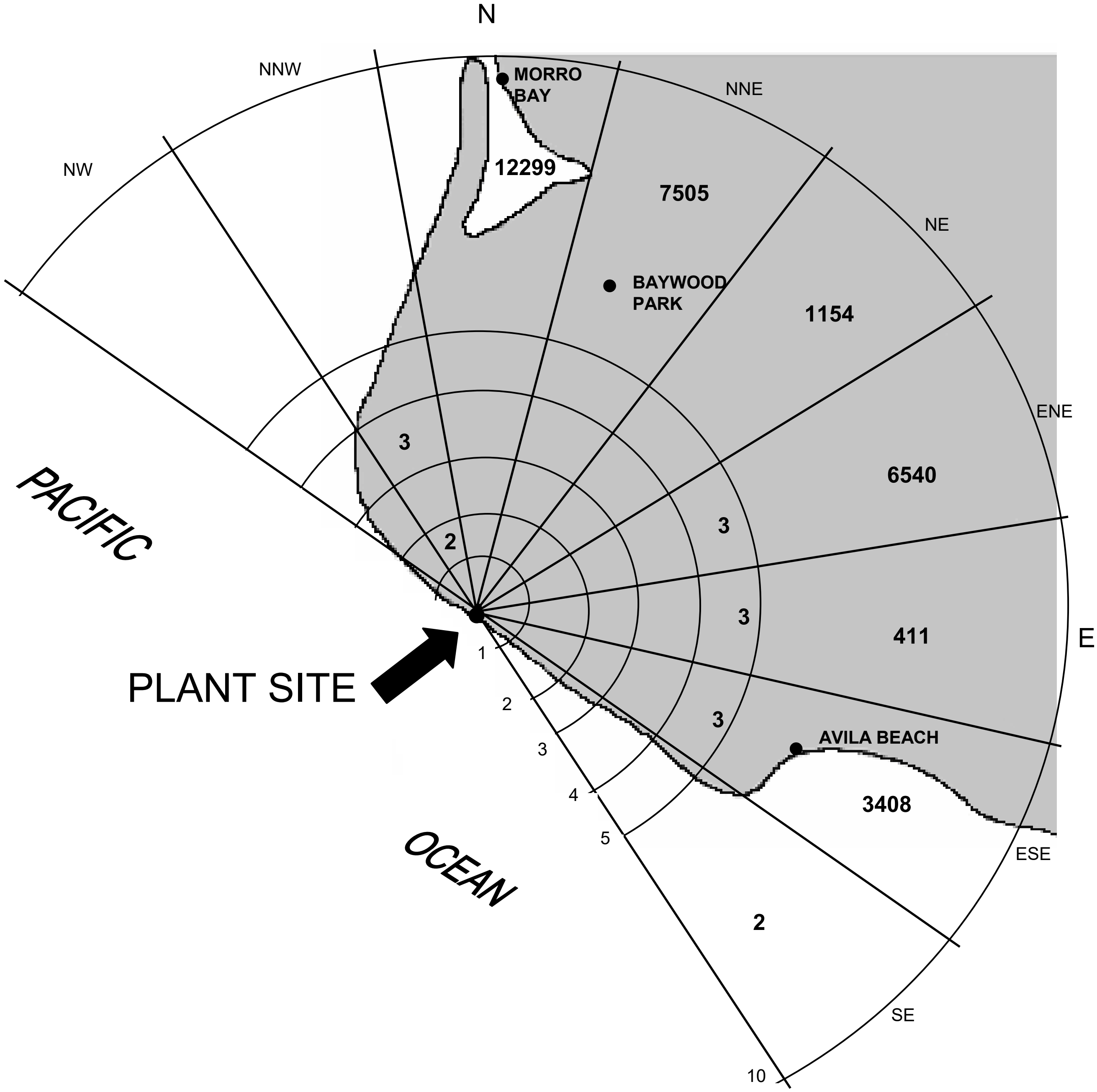
DIABLO CANYON ISFSI
FIGURE 2.1-3 POPULATION DISTRIBUTION 0 TO 10 MILES 2000 CENSUS



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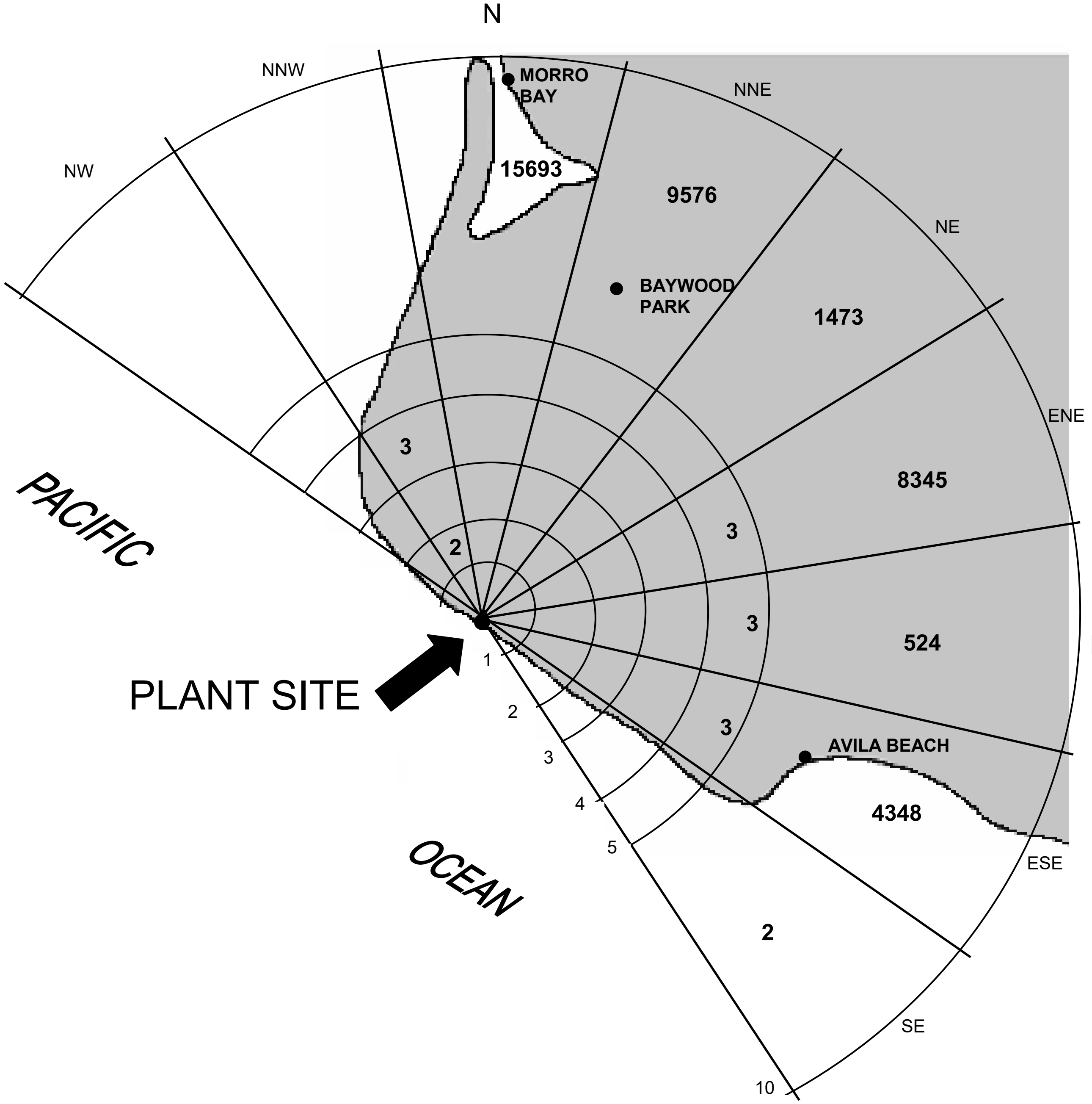
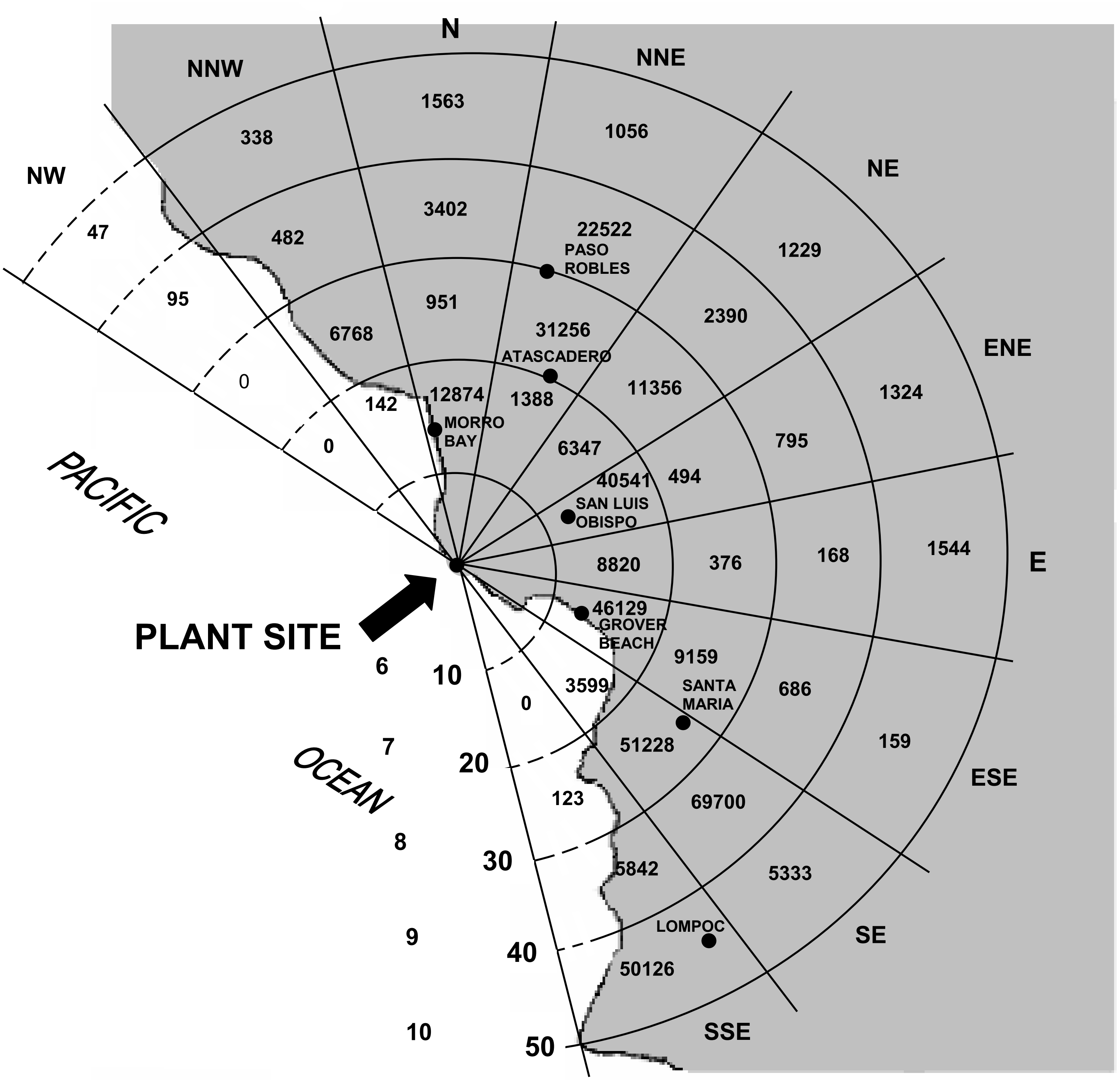
DIABLO CANYON ISFSI
FIGURE 2.1-4 POPULATION DISTRIBUTION 0 TO 10 MILES 2010 PROJECTED

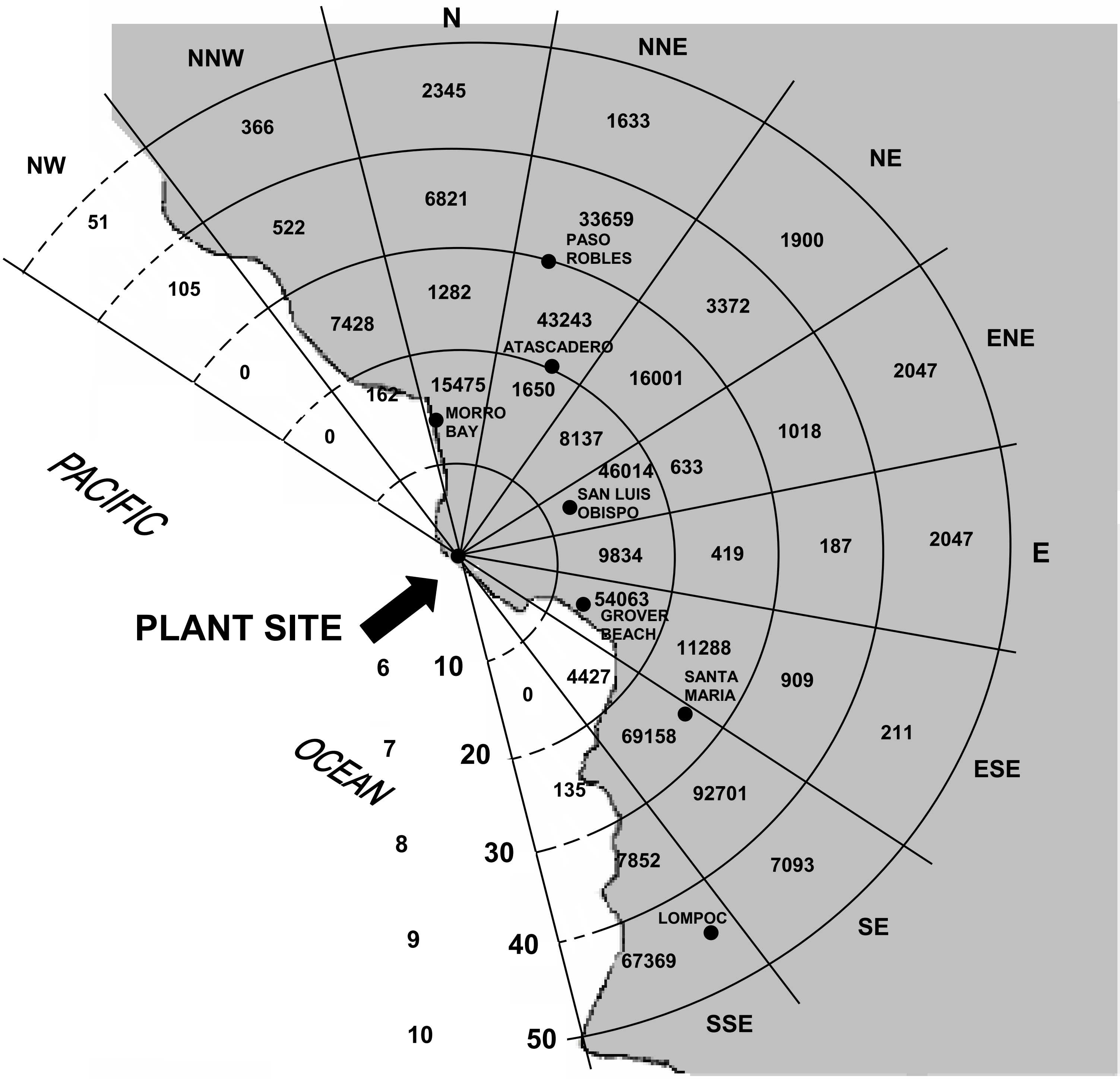
Revision 0 June 2004



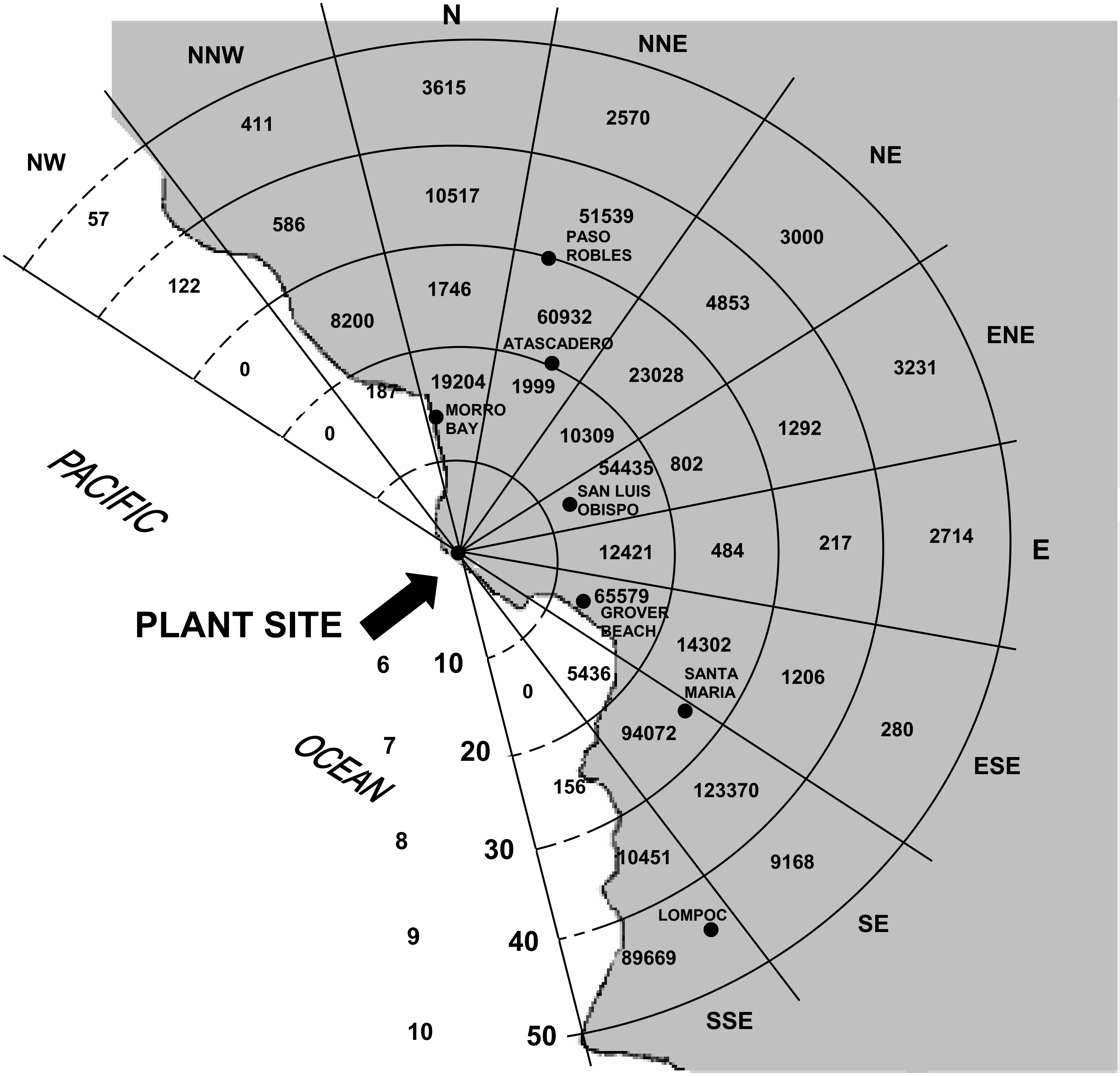
FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.1-5 POPULATION DISTRIBUTION 0 TO 10 MILES 2025 PROJECTED



FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.1-6 POPULATION DISTRIBUTION 10 TO 50 MILES 2000 CENSUS



FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.1-7 POPULATION DISTRIBUTION 10 TO 50 MILES 2010 PROJECTED

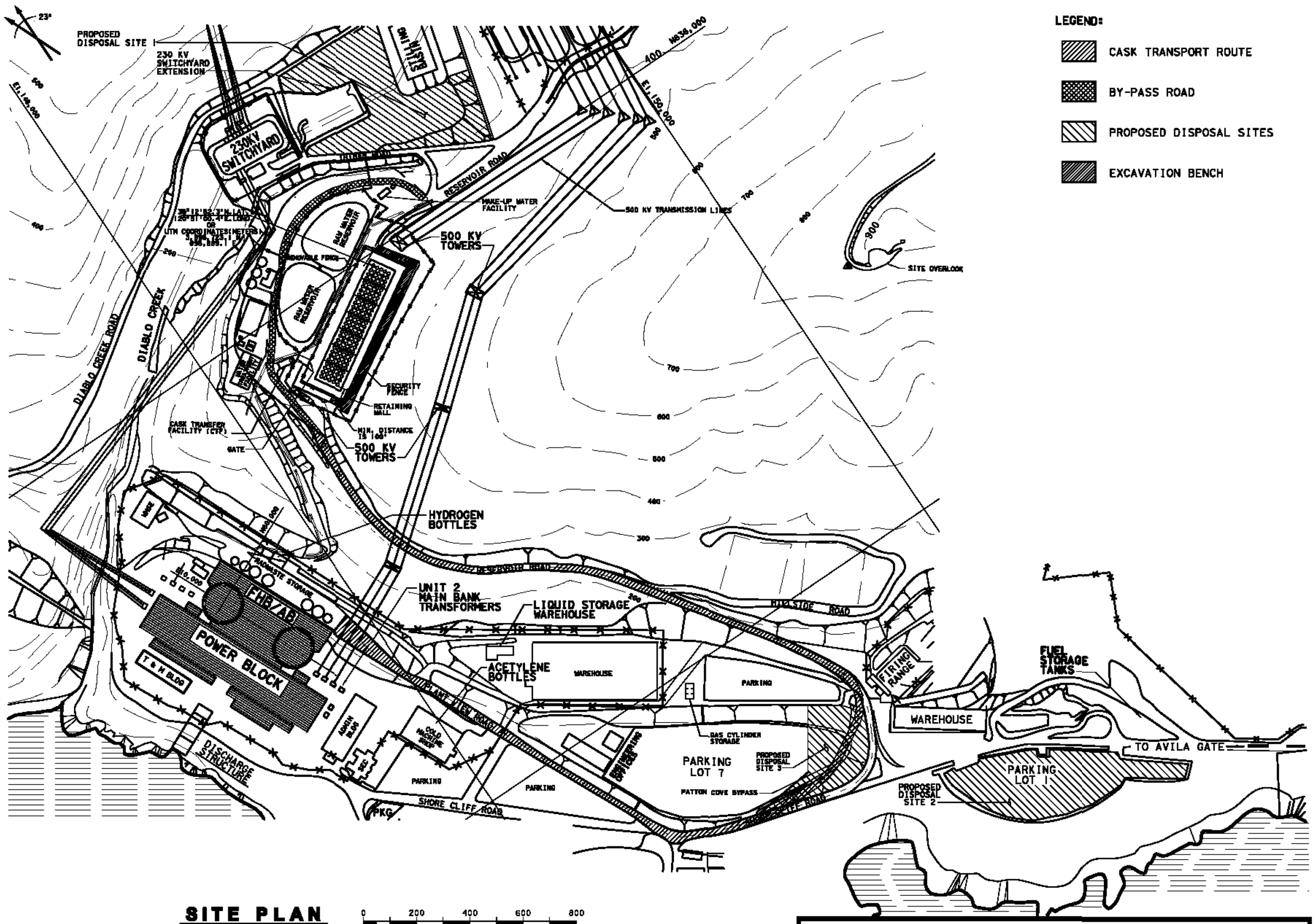


FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.1-8
POPULATION DISTRIBUTION
10 TO 50 MILES
2025 PROJECTED

Revision 0 June 2004



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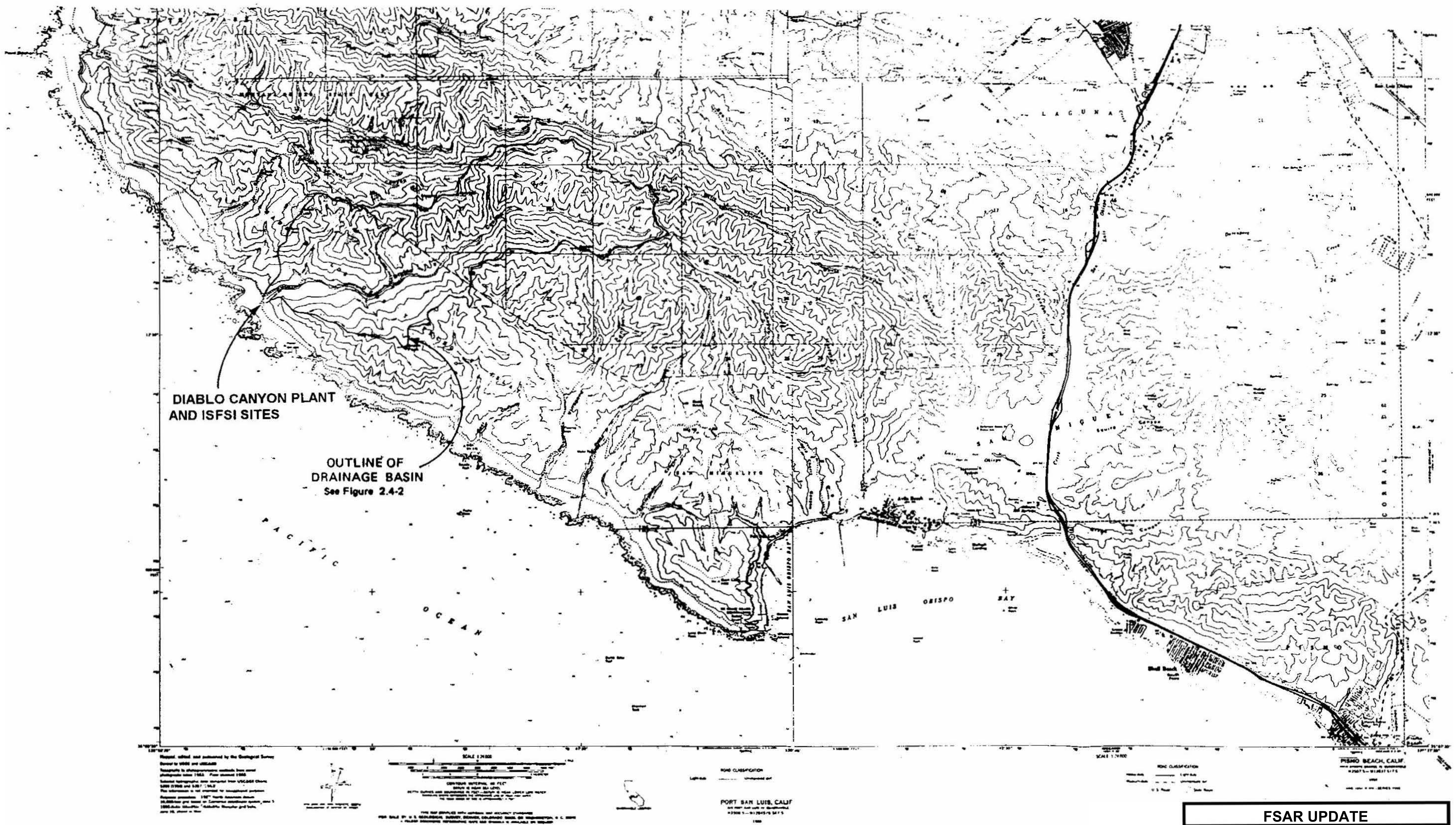
DIABLO CANYON ISFSI

FIGURE 2.2-1
STATIONARY ONSITE POTENTIAL HAZARDS



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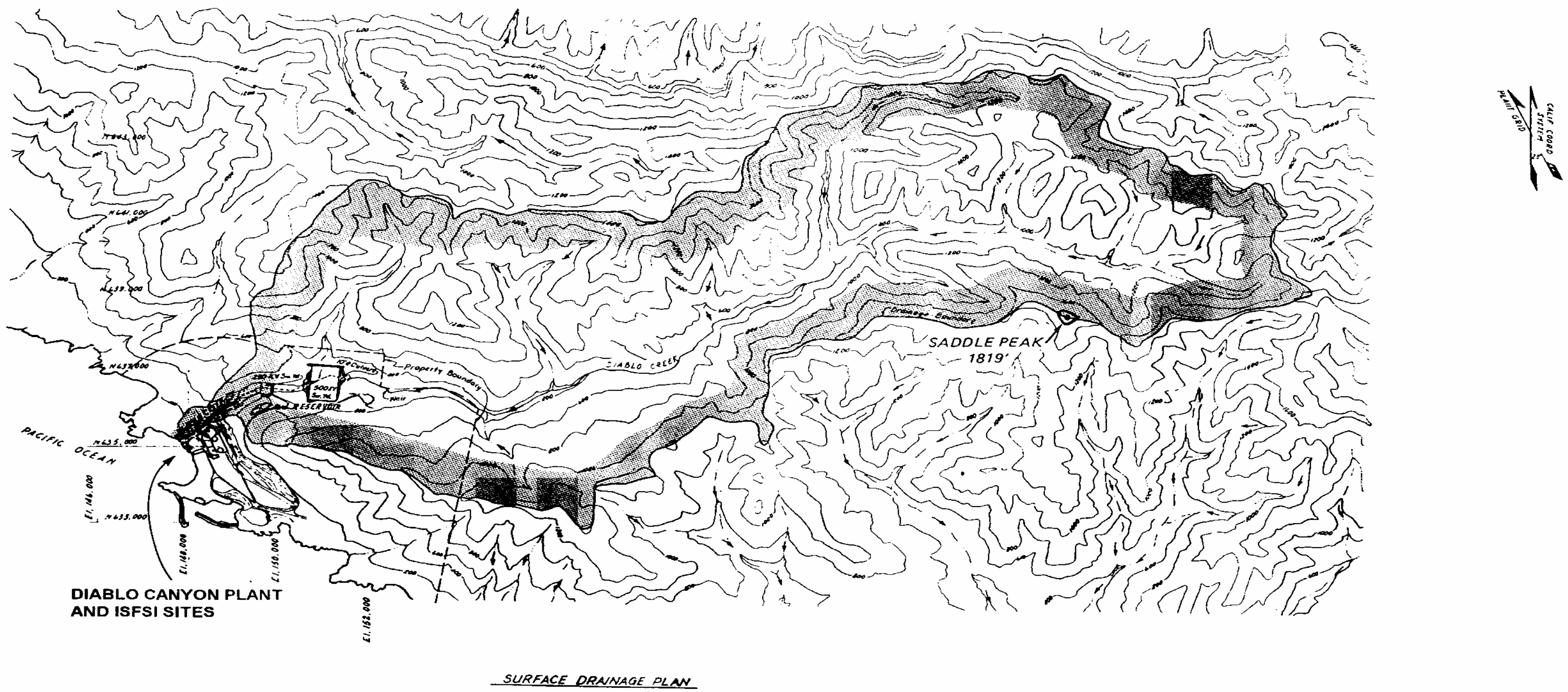
DIABLO CANYON ISFSI

FIGURE 2.4-1
PLANT LOCATION DRAINAGE
AND TOPOGRAPHY



NOTES:

1 GRID COORDINATES SHOWN ARE BASED ON CALIF STATE COORDINATE SYSTEM
2 DATUM FOR ELEVATIONS SHOWN IS MEAN SEA LEVEL EL 0.0'
3 ARROW (←) INDICATES DIRECTION OF RUN OFF
4 TOPOGRAPHY PHOTOGRAMMETRICALLY REPRODUCED FROM USGS PORT SAN LUIS 7.5 MINUTE QUADRANGLE, 1965.

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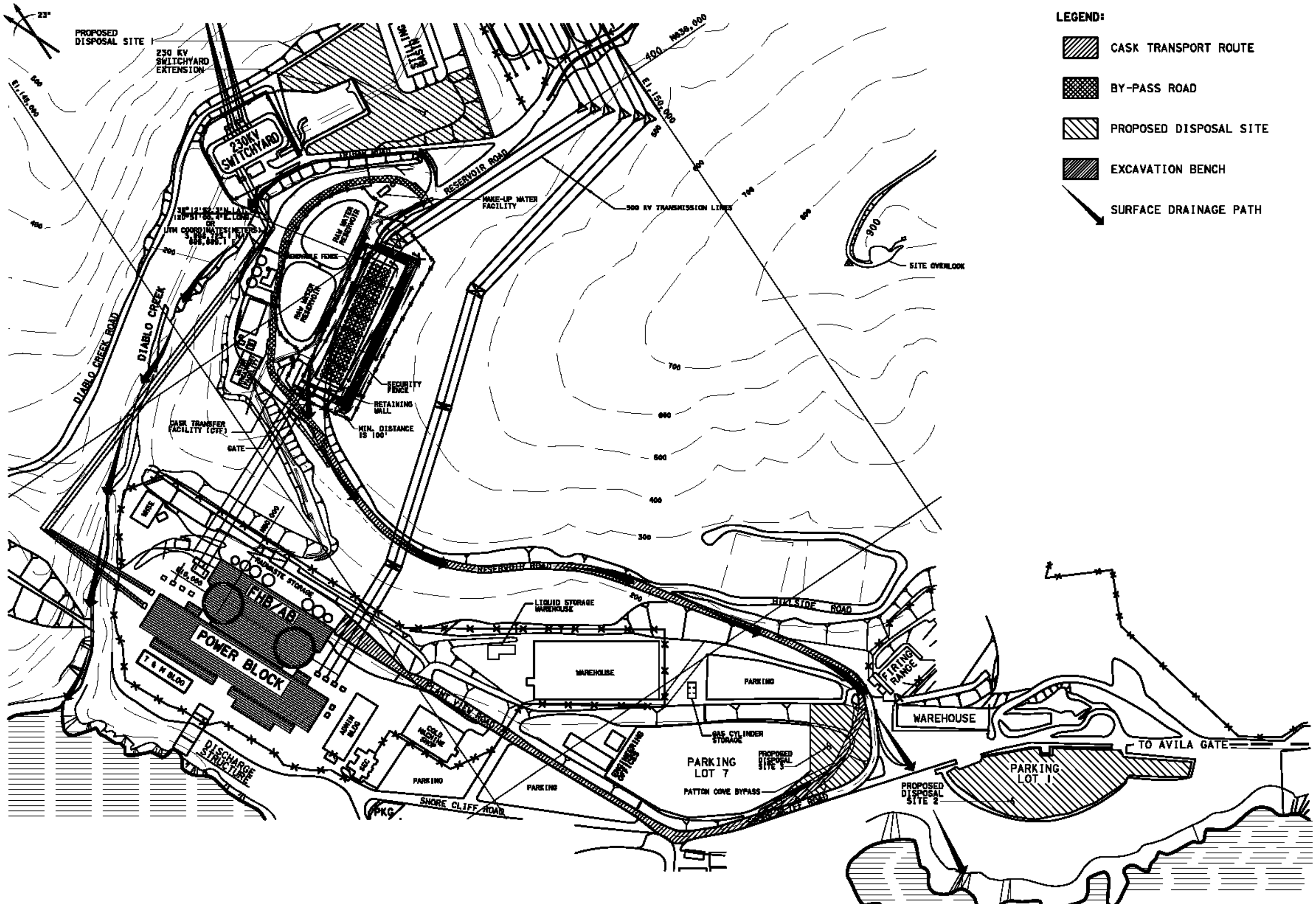
DIABLO CANYON ISFSI
FIGURE 2.4-2 **SURFACE DRAINAGE PLAN**



SITE PLAN

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.4-3
ISFSI SURFACE DRAINAGE

Security-Related Information Figure Withheld Under 10 CFR 2.390.

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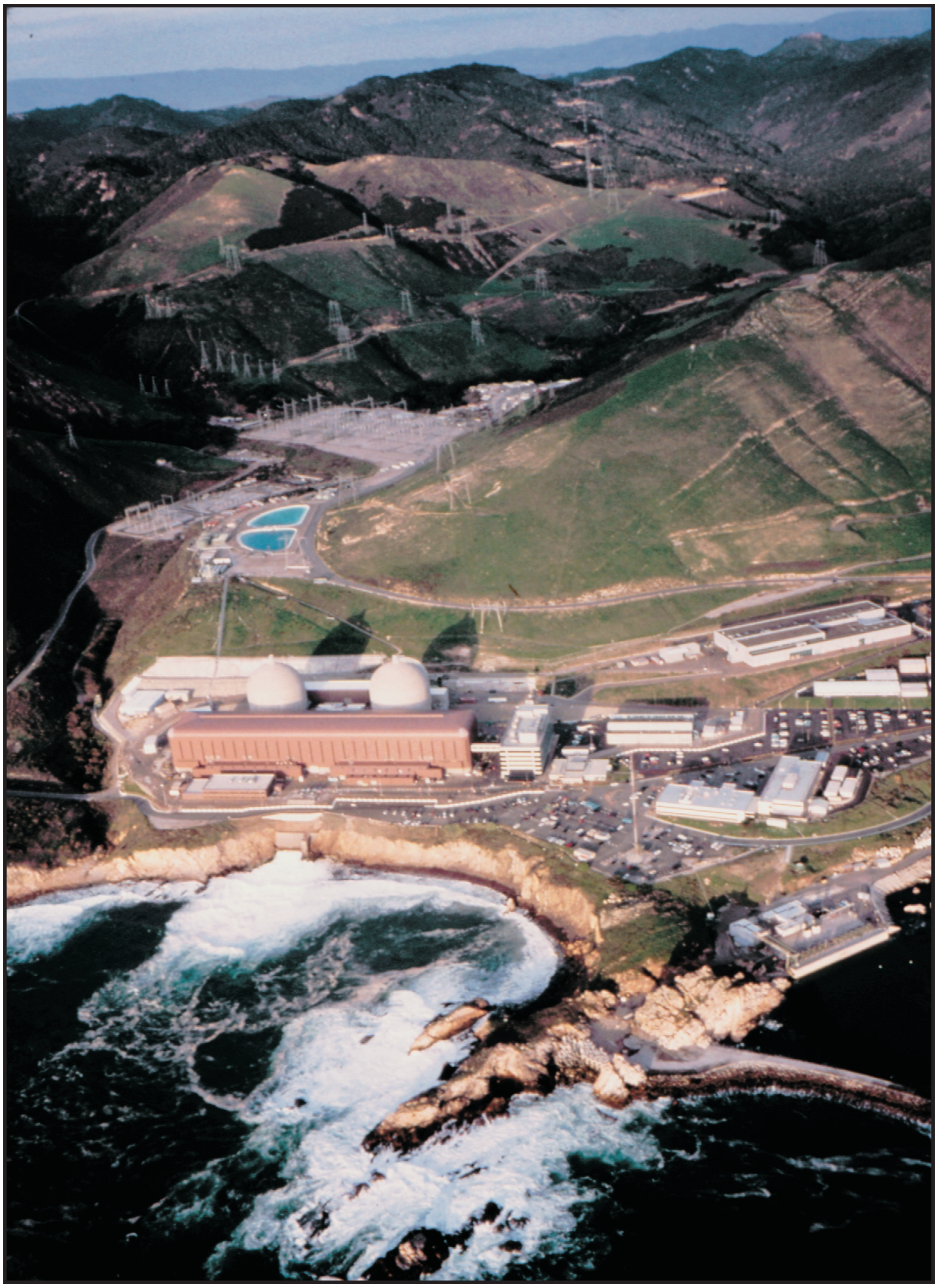
DIABLO CANYON ISFSI
FIGURE 2.6-1 LOCATION AND BOUNDARIES OF GEOLOGIC STUDY AREAS

Revision 0 June 2004

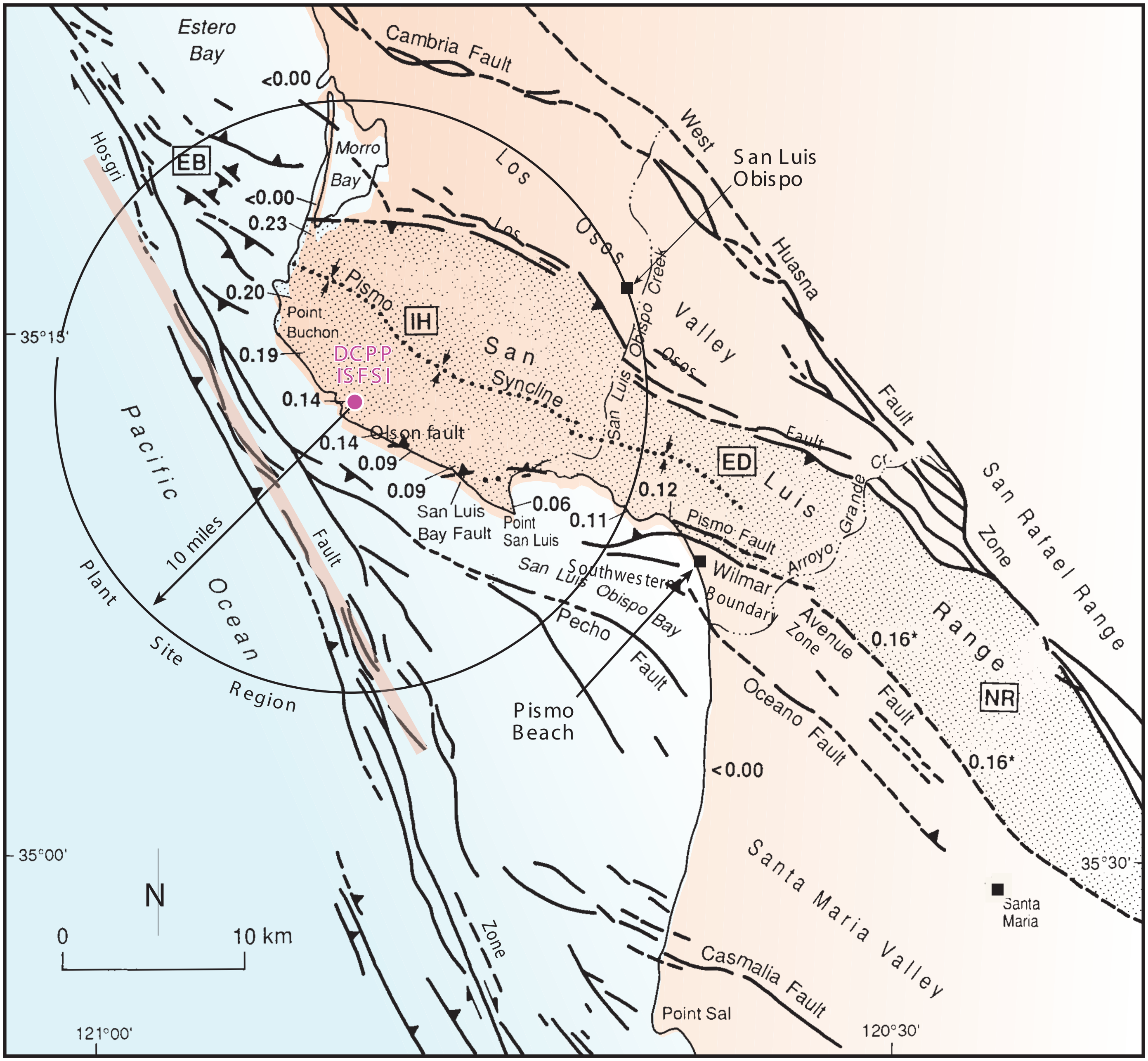


Northeast view of Diablo Canyon Power Plant and the ISFSI and CTF sites. The ISFSI is at the base of the slope to the right of the raw water reservoir. The CTF is directly southwest of the reservoirs. The extent of the 1971 borrow area excavation is indicated by the rocky area on the slope above the reservoir. The power plant and adjacent facilities are constructed on a marine terrace that is covered by Quaternary fan deposits. Photo roll WDP-1.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-2 DIABLO CANYON POWER PLANT AND THE ISFSI AND CTF SITES

Security-Related Information Figure Withheld Under 10 CFR 2.390.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-3 SOUTHWARD VIEW OF THE ISFSI AND CTF SITES AND TRANSPORT ROUTE



(from PG&E, 1988)

EB

IH

ED

NR

0.14

0.16*

Average fault trend (338) used f or ground motion analyses

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.6-4
REGIONAL STRUCTURE MAP

Photo of Obispo Formation dolomite and sandstone strata exposed on the hillslope above the transport route on Reservoir Road. The ISFSI site is to the right of the raw water reservoir. Bedding dips into the hillslope on the west limb of the regional Pismo syncline and extends beneath the power block (off photo to lower left). A small parasitic syncline is manifest as the U-shaped strata directly below the ridge crest in the middle of the photo. Several debris-flow chutes (↓) form the gullies on the slope above Reservoir Road. Photo roll JLB-2.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-5 OBISPO FORMATION DOLOMITE AND SANDSTONE ON HILLSLOPE ABOVE RESERVOIR ROAD

Security-Related Information Figure Withheld Under 10 CFR 2.390.

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FIGURE 2.6-6 GEOLOGIC MAP OF BEDROCK AND LANDSLIDES IN THE PLANT SITE AREA

Revision 0 June 2004

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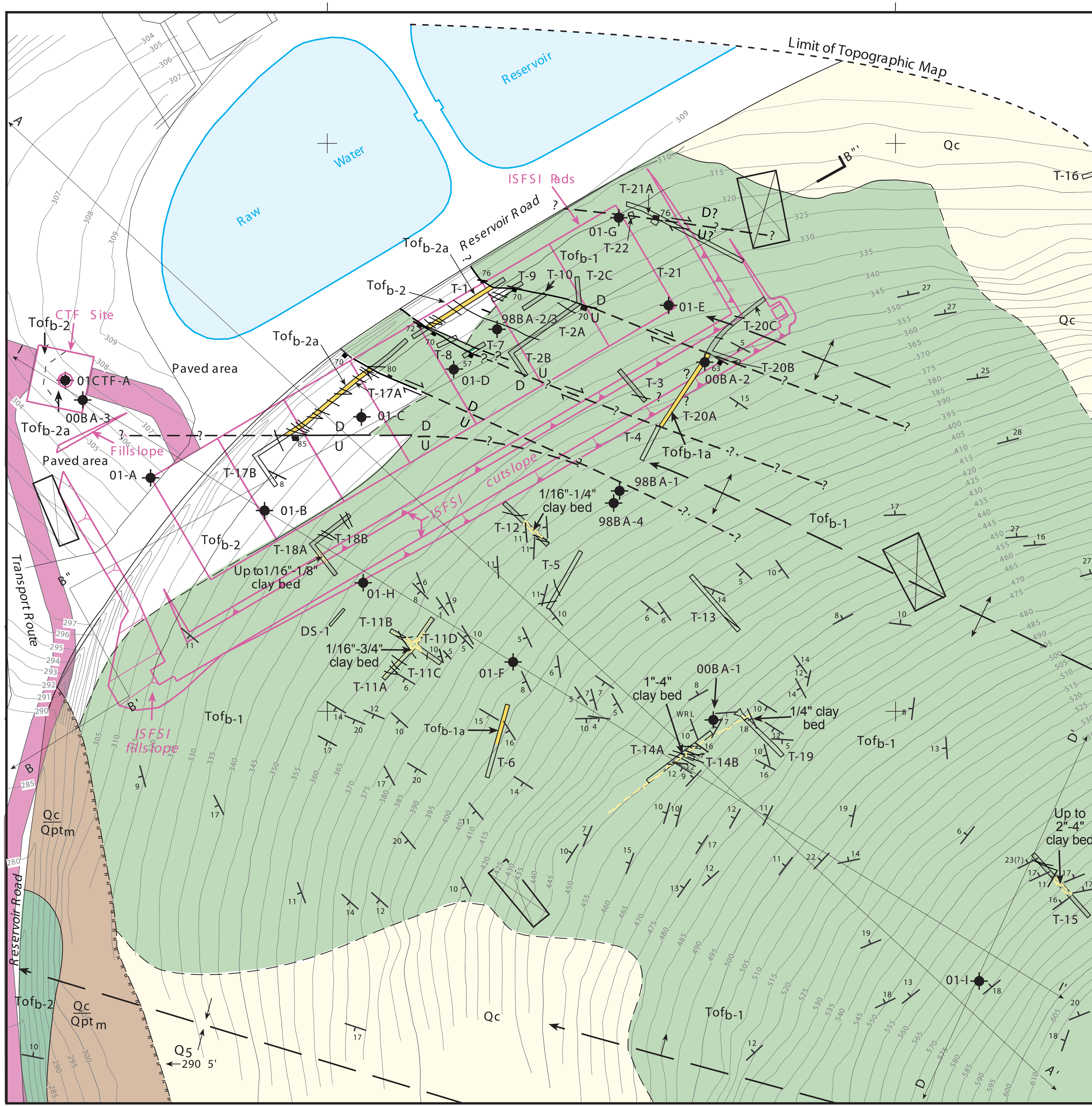
FIGURE 2.6-7 GEOLOGIC MAP OF THE ISFSI STUDY AREA AND TRANSPORT ROUTE VICINITY



Explanation

Qc	Colluvium
Qptm	Marine terrace deposit (overlain by Qc)

Obispo Formation (lower and middle Miocene)

DOLOMITE UNIT

Tofb-1	Dolomite, clayey dolomite, dolomitic siltstone to fine-grained dolomitic sandstone, and limestone. The unit contains occasional discontinuous to continuous (tens to hundreds of feet) clay beds that are generally 1/32- to 1/2-inch thick, but locally are thicker. Rocks in this unit are moderately to well cemented, moderately hard to hard, moderately to slightly weathered, brittle and typically medium strong.
Tofb-1a	Friable dolomite and dolomitic siltstone of unit Tofb-1. These rocks typically have low hardness, are very weak to weak, and occur as discontinuous zones where weathering and/or alteration has been concentrated. Inferred lateral extent of friable zones is schematic.

SANDSTONE UNIT

Tofb-2	Fine to coarse-grained dolomitic sandstone and sandstone (arkosic to arenitic) with lesser dolomite beds. Detrital clasts are composed primarily of dolomitized feldspars, marine fossil fragments, and volcanic rock fragments. Discontinuous clay beds that are generally less than 1/2-inch thick occur locally within the unit. The rocks are of low to medium hardness, moderately to well cemented and typically medium strong.
Tofb-2a	Friable sandstone of unit Tofb-2. These rocks typically are of low hardness are very weak to weak, and occur as discontinuous zones where weathering and/or alteration has been concentrated. Inferred lateral extent of friable zones is schematic.

6	Strike and dip of bedding
U D 80	Minor fault, dip indicated, dashed where inferred, queried where uncertain, arrows show relative sense of displacement, U-upthrown, D-downthrown
	Small, secondary faults exposed in trench
1/16"	Clay bed, thickness indicated
	Geologic contact, solid line where well-defined, dashed where approximate
01-C	Boring for ISFSI, number indicated (initial number is year drilled, e.g. 01 was drilled in 2001)
A A'	Geologic cross section, arrows indicate end of line is off the map area
T-4	Exploratory trench, number indicated
DS-1	Discontinuity survey line in bulldozer cut
	Footprint of 500-kV tower
	Cutslope above and fill prism west of ISFSI pads
	Axis of anticline, larger arrow shows plunge, dashed where approximate
	Axis of syncline, larger arrow shows plunge, dashed where approximate
	Axis of monocline, larger arrow shows plunge, dashed where approximate
Q5 290 5'	Buried shoreline angle of marine terrace wave-cut platform, number and elevation indicated

Notes:

ISFSI geometry is based on PG&E Enercon Drwg. Base map from No. PGE-009-sk-001 dated 9/27/01.

Geology not shown in paved area and reservoir area.

N

0 50 100 150 200

Contour interval = 5 feet

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FIGURE 2.6-8

GEOLOGIC MAP OF ISFSI AND CTF SITES

Revision 0 June 2004

Explanation

Geology

Quaternary

- **af** Artificial fill (engineered)
- **Qal, Qdf, Qc, Qls, Qhf** Quaternary deposits - alluvium (Qal), debris flow (Qdf), colluvium (Qc), landslide (Qls), Holocene colluvial fan (Qhf)
- **Qpf** Pleistocene colluvial fan
- **Qptm** Pleistocene marine terrace deposit (inferred)

Tertiary

- **Tvr** Volcanic rock (middle Miocene), diabase intrusive sills and dikes

Obispo Formation (lower and middle Miocene)

- **Tofb** Member Tof, unit b - dolomite, dolomitic siltstone, dolomitic sandstone, and sandstone: medium to thick bedding
- **Tofc** Member Tof, unit c - shale, claystone and siltstone, thin to medium bedding, extensively sheared
- **Tor** Member Tor - volcanic rock, zeolitized and silicified tuff

DOLOMITE SUBUNIT

- **Tofb-1** Dolomite, clayey dolomite, dolomitic siltstone to fine-grained dolomitic sandstone, and limestone. The unit contains occasional discontinuous to continuous (tens to hundreds of feet) clay beds that are generally 1/32- to 1/2-inch thick, but locally are thicker. Rocks in this unit are moderately to well cemented, moderately hard to hard, moderately to slightly weathered, brittle and typically medium strong.
- **Tofb-1a** Friable dolomite and dolomitic siltstone of unit Tofb-1. These rocks typically have low hardness, are very weak to weak, and occur as discontinuous zones where weathering and/or alteration has been concentrated. Inferred lateral extent of friable zones is schematic.

SANDSTONE SUBUNIT

- **Tofb-2** Fine to coarse-grained dolomitic sandstone and sandstone (arkosic to arenitic) with lesser dolomite beds. Detrital clasts are composed primarily of dolomitized feldspars, marine fossil fragments, and volcanic rock fragments. Discontinuous clay beds that are generally less than 1/2-inch thick occur locally within the unit. The rocks are of low to medium hardness, moderately to well cemented and typically medium strong.
- **Tofb-2a** Friable sandstone of unit Tofb-2. These rocks typically are of low hardness, are very weak to weak, and occur as discontinuous zones where weathering and/or alteration has been concentrated. Inferred lateral extent of friable zones is schematic.

Symbols

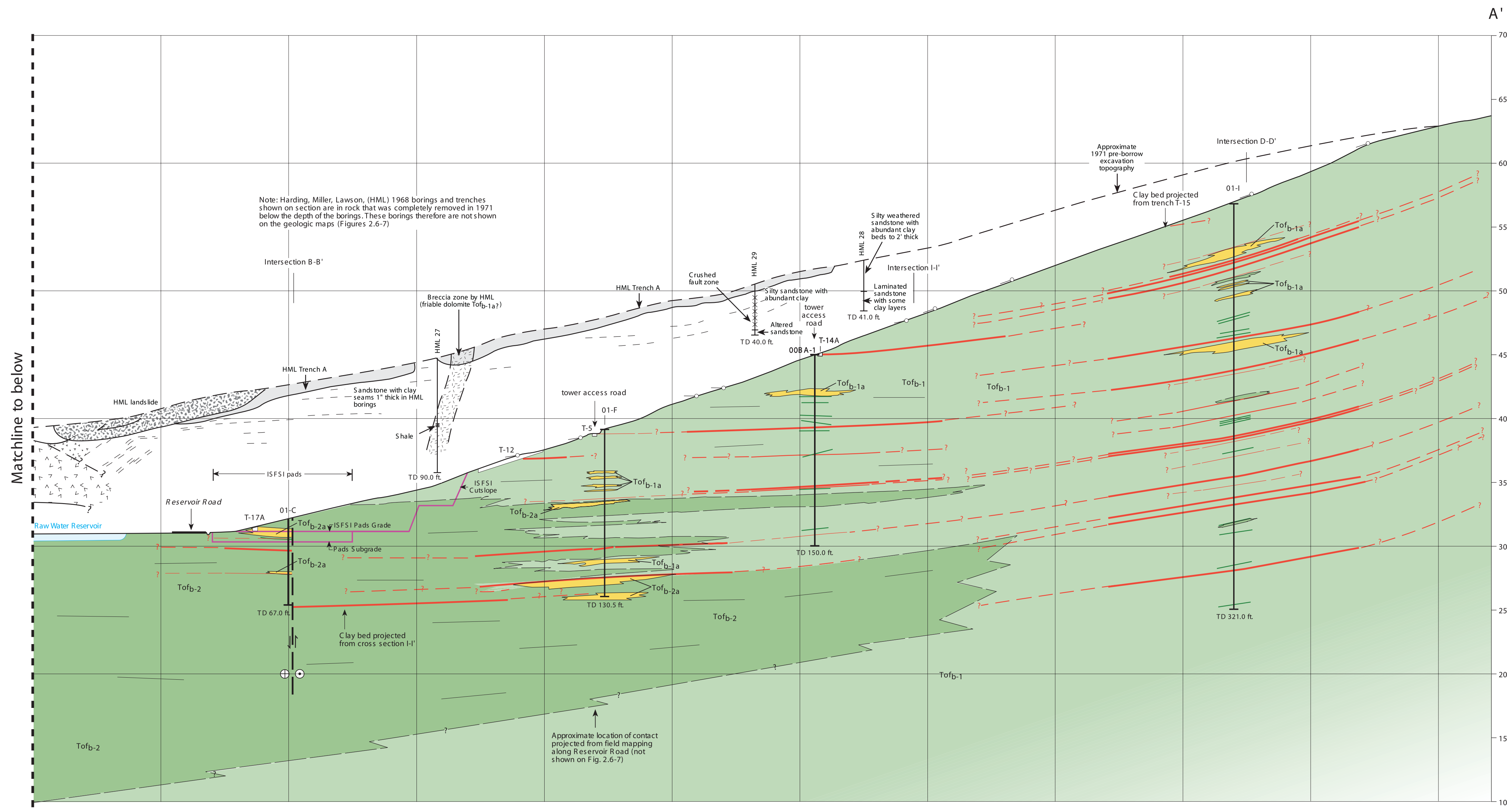
- Apparent dip of bedding
- Apparent dip of bedding measured in boring
- Bedding shown schematically and inferred from projection of nearby surface outcrops or borehole surveys
- ? Lithologic contact, inferred except where encountered in boreholes and surface mapping, queried where uncertain
- Fault - location based on projection of trench or surface exposure. Vertical projection is inferred. Dip of fault assumed to be vertical unless well constrained by multiple surface attitudes. Queried where uncertain. Sense of vertical separation is indicated by arrows and inferred horizontal right-slip displacement by ⊕ (moving away from viewer) and ⊙ (moving toward viewer).
- Clay, silty and sandy, tan to brown, some angular fragments within clay matrix, locally contains foraminifera, generally thin (1/16 to 1/2 inch). Clay beds more common and more laterally continuous in dolomite unit than in the sandstone unit.
- <1/8", 1/8"-1/4", >1/4" Clay bed - weight of solid line indicates bed thickness where encountered in borings or trenches, dashed line where inferred (no thickness indicated); queried where uncertain, clay beds projected beyond borings or trenches based on thickness (>1/4-inch thick beds projected 100 feet as solid line and 100 feet as dashed line, 1/8-inch to 1/4-inch thick beds projected 50 feet solid and 50 feet dashed, <1/8-inch thick beds projected 25 feet solid and 25 feet dashed) unless constrained to be shorter on adjoining boreholes and trenches.
- 01-H, TD 30.5 ft. Exploration boring, number (01-H, initial number indicates year drilled) and depth (TD 30.5 feet) indicated
- T-18 Exploration trench, number indicated; trenches are shown open if they cross section line, filled if projected into section line
- Groundwater elevation and date of measurement indicated

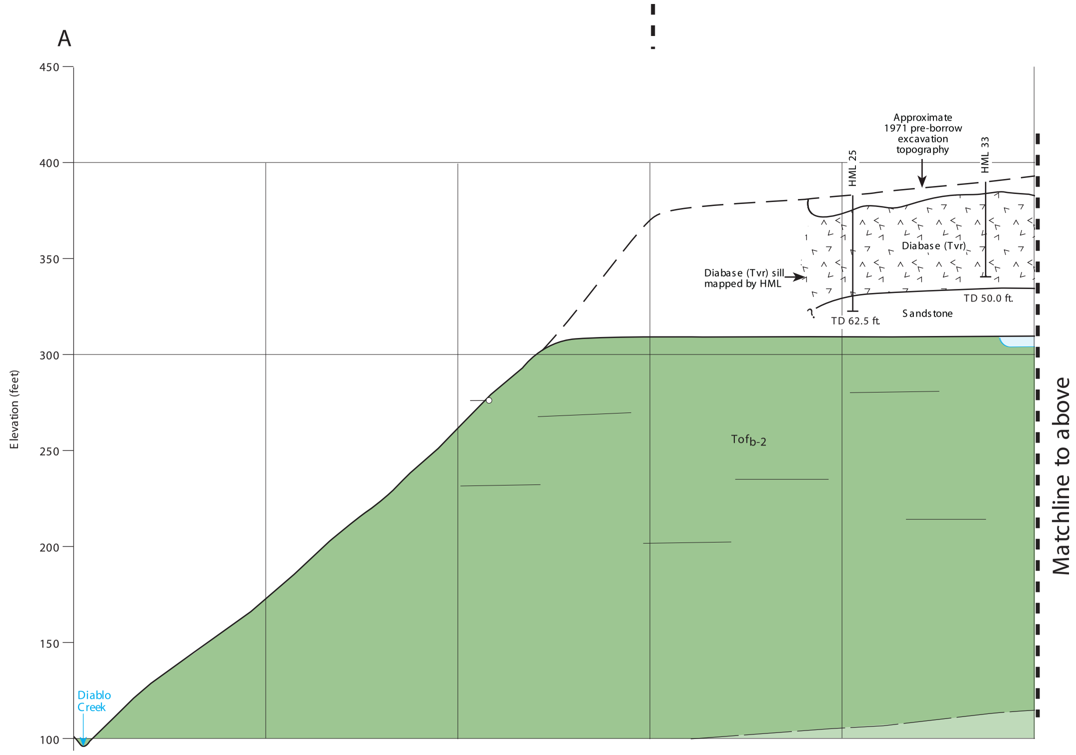
Pleistocene Marine Terraces Designation*

- Q_1 Oxygen Isotope Stage 5a marine terrace (80,000 years old)
- Q_2 Oxygen Isotope Stage 5e marine terrace (120,000 years old)
- Q_3 Oxygen Isotope Stage 7 marine terrace (210,000 years old)
- Q_4 Oxygen Isotope Stage 9 marine terrace (330,000 years old)
- Q_5 Oxygen Isotope Stage 11 marine terrace (430,000 years old)

* *Ages and correlation of marine terraces based on K.L., Hansen, J.R. Wesling, W.R. Lettis, K.I. Kelson and L. Mezger, 1994. Correlation, ages, and uplift rates of Quaternary marine terraces, in Alterman, I.B., McMullen, R.B., Cluff, L.S., and Slemmons, D.B., eds., Seismotectonics of the Central California Coast Ranges: Boulder, Colorado, Geological Society of America Special Paper 292.*

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-9
EXPLANATION FOR CROSS SECTIONS





Notes
1. Location of cross section shown on Figures 2.6-7 and 2.6-8. Nearby borings are projected to cross section.
2. See Figure 2.6-9 for explanation of geologic units.
3. Horizontal scale = vertical scale.

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FIGURE 2.6-10 **CROSS SECTION A-A'**

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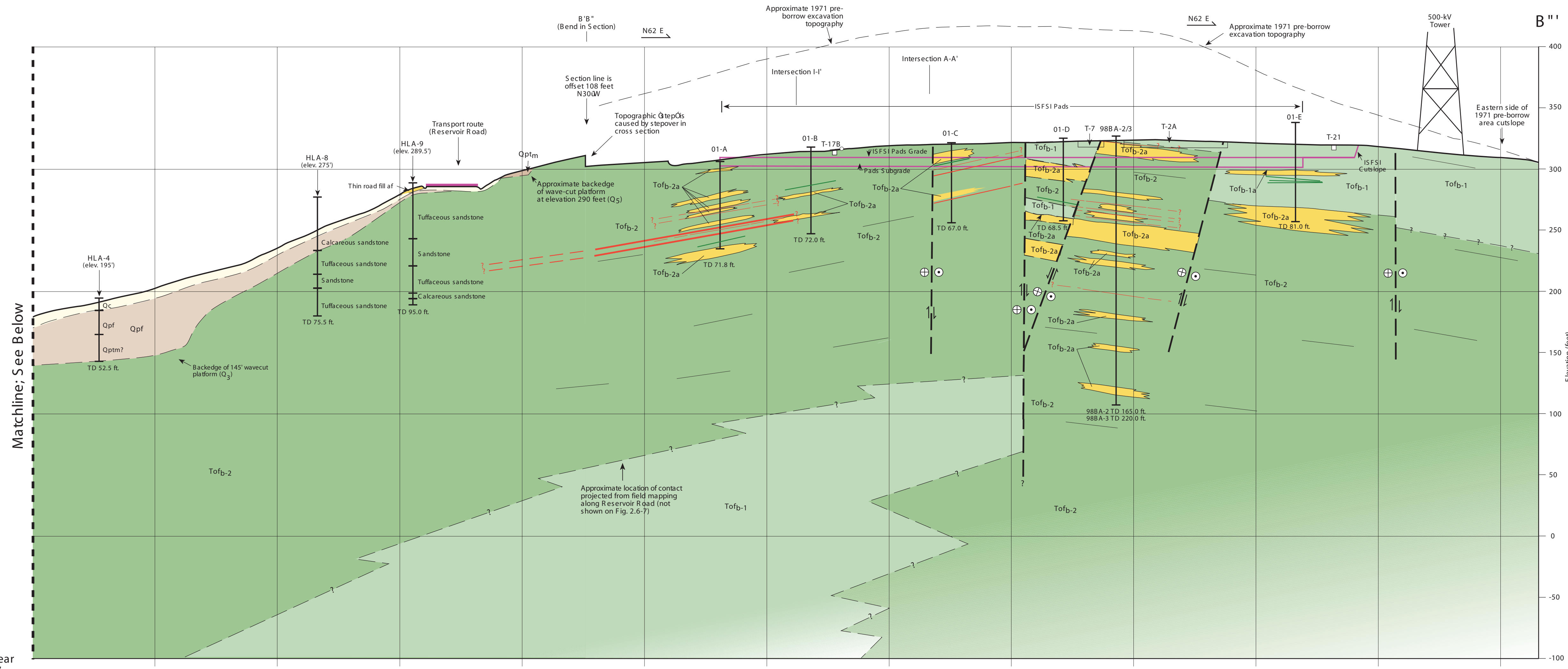

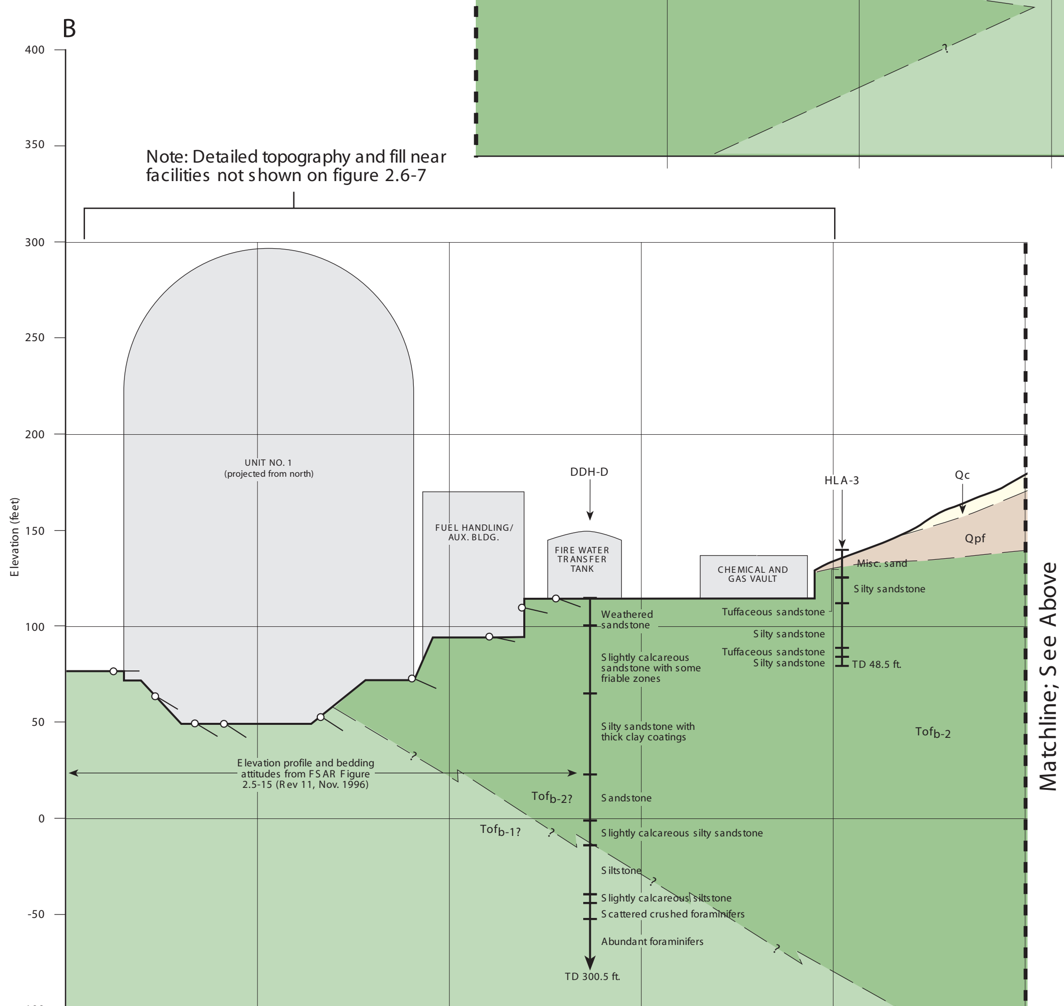

Notes
1. Location of cross section shown on Figures 2.6-7 and 2.6-8. Nearby borings are projected to cross section.
2. See Figure 2.6-9 for explanation of geologic units.
3. Hortizontal scale = vertical scale.

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FIGURE 2.6-11
CROSS SECTION B-B'

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Southward view of the ISFSI site, above the raw water reservoir. The 1971 borrow area cutslope is indicated by areas of bedrock exposure and brown grass. Trenches excavated for the ISFSI investigations are shown (trenches backfilled in April 2001). Trench T-16 is located to the left of the photo. Photo roll AR 3-25.

FSAR UPDATE
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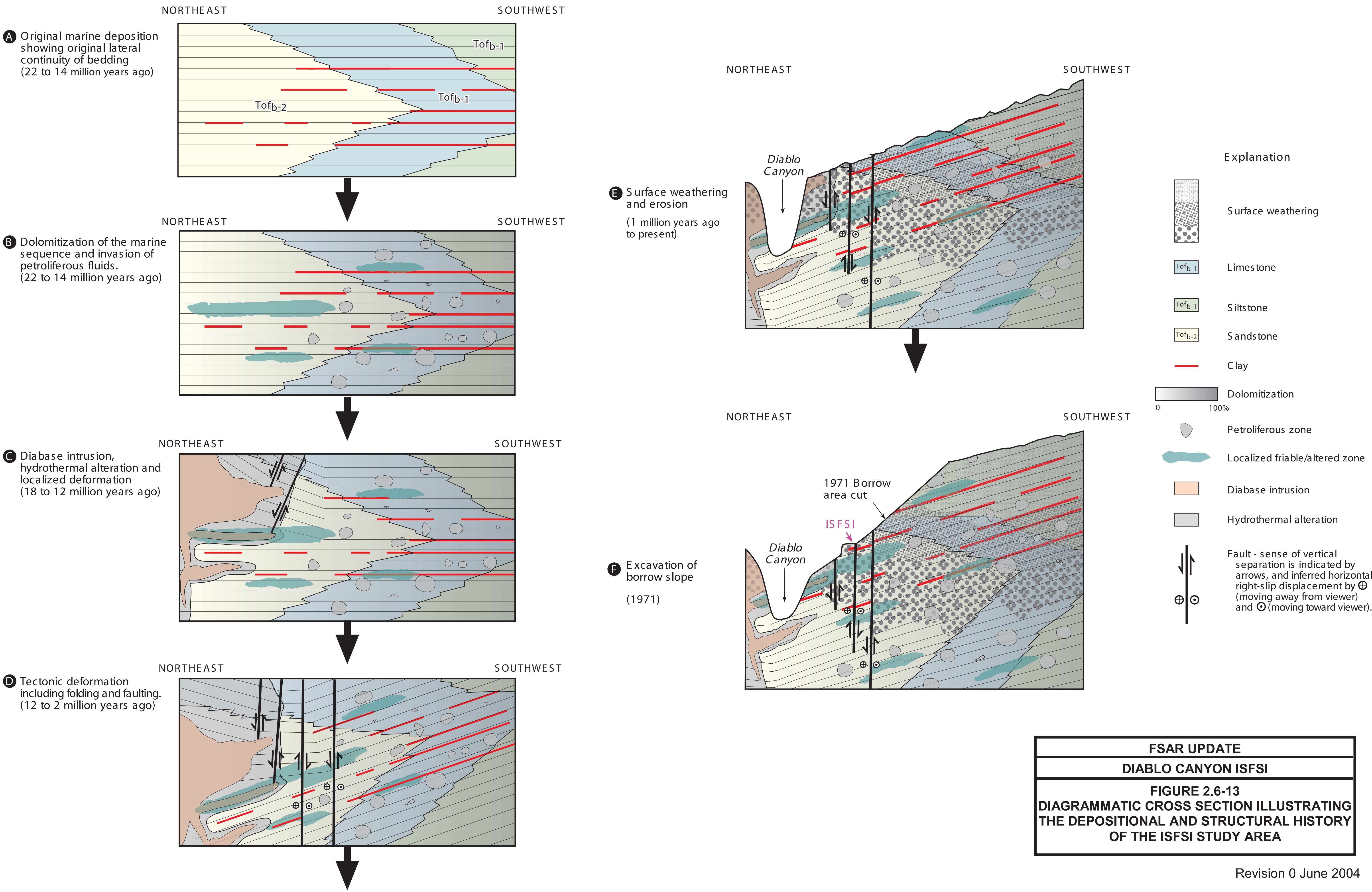
FIGURE 2.6-12 SOUTHWARD VIEW OF ISFSI STUDY AREA



FSAR UPDATE

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FIGURE 2.6-13
DIAGRAMMATIC CROSS SECTION ILLUSTRATING THE DEPOSITIONAL AND STRUCTURAL HISTORY OF THE ISFSI STUDY AREA

Revision 0 June 2004

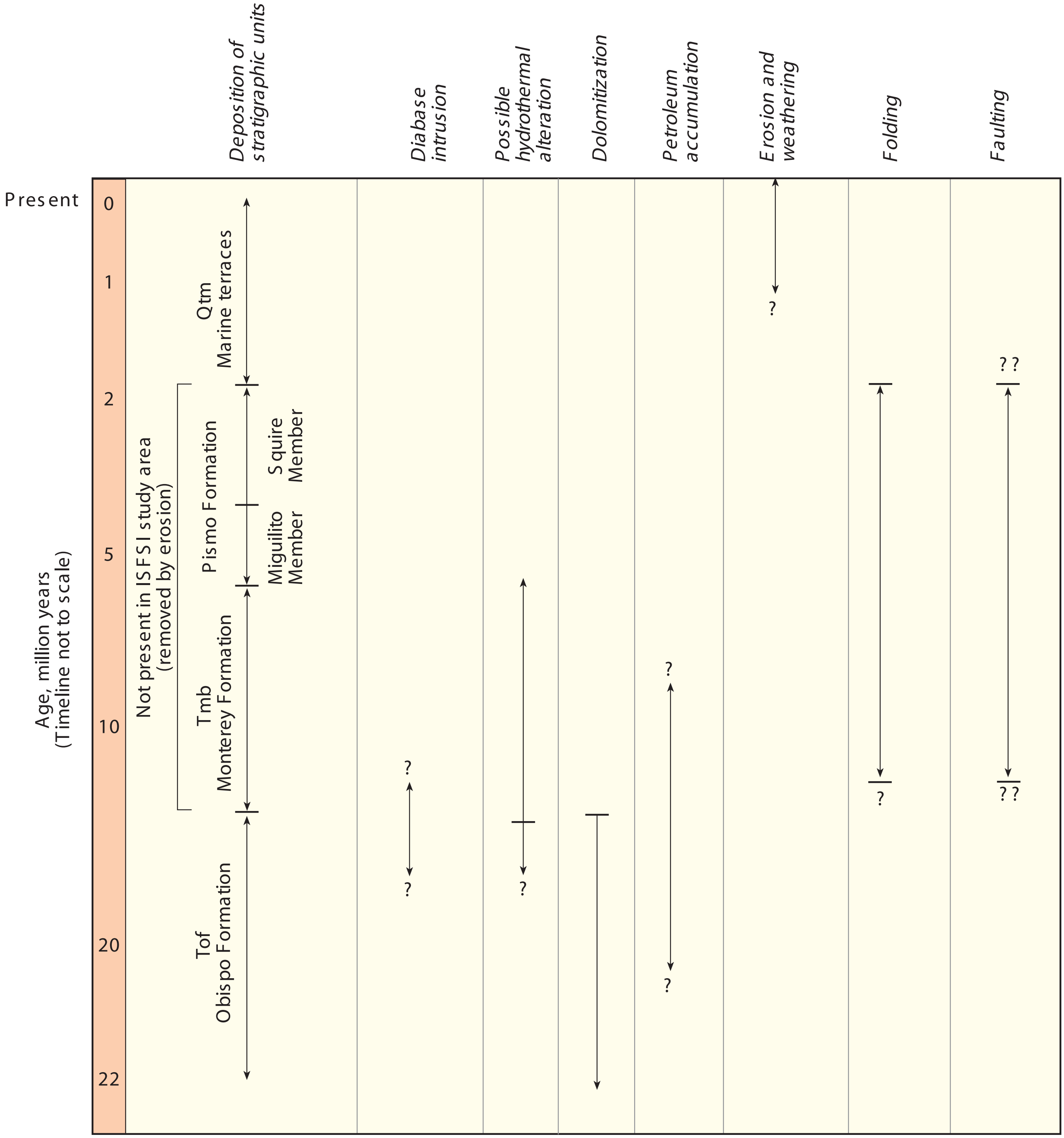

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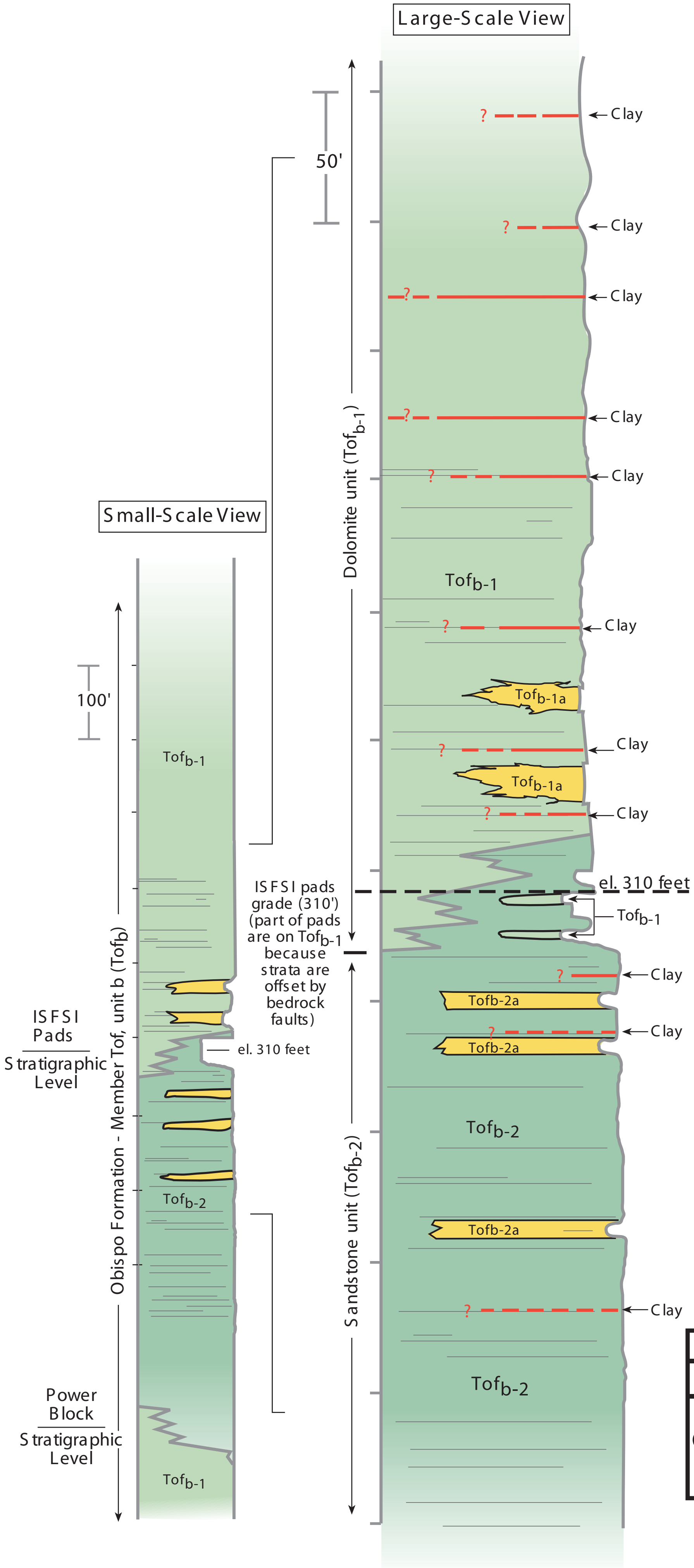
FIGURE 2.6-14
CHRONOLOGY OF STRATIGRAPHY AND GEOLOGIC PROCESSES AT THE ISFSI STUDY AREA



Explanation

(Informal subunits of unit Tof_b of Obispo Formation)

DOLOMITE SUBUNIT

Tof_{b-1} Dolomite, clayey dolomite, dolomitic siltstone to fine-grained dolomitic sandstone, and limestone. The unit contains occasional discontinuous to continuous (tens to hundreds of feet) clay beds that are generally 1/32-to 1/2-inch thick, but locally are thicker. Rocks in this unit are moderately to well cemented, moderately hard to hard, moderately to slightly weathered, brittle and typically medium strong.

Tof_{b-1a} Friable dolomite and dolomitic siltstone of unit Tof_{b-1}. These rocks typically have low hardness, are very weak to weak, and occur as discontinuous zones where weathering and/or alteration has been concentrated. Inferred lateral extent of friable zones is schematic.

SANDSTONE SUBUNIT

Tof_{b-2} Fine to coarse-grained dolomitic sandstone and sandstone (arkosic to arenitic) with lesser dolomite beds. Detrital clasts are composed primarily of dolomitized feldspars, marine fossil fragments, and volcanic rock fragments. Discontinuous clay beds that are generally less than 1/2-inch thick occur locally within the unit. The rocks are of low to medium hardness, moderately to well cemented and typically medium strong.

Tof_{b-2a} Friable sandstone of unit Tof_{b-2}. These rocks typically are of low hardness and are very weak to weak, and occur as discontinuous zones where weathering and/or alteration has been concentrated. Inferred lateral extent of friable zones is schematic.

? — — Clay bed; silty and sandy, tan to brown, some angular fragments within clay matrix, locally contains foraminifera, generally thin (1/16 to 1/2 inch). Clay beds more common and more laterally continuous in dolomite unit than in the sandstone unit

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FIGURE 2.6-15
GENERALIZED STRATIGRAPHIC COLUMN AT THE ISFSI AND POWER BLOCK SITES

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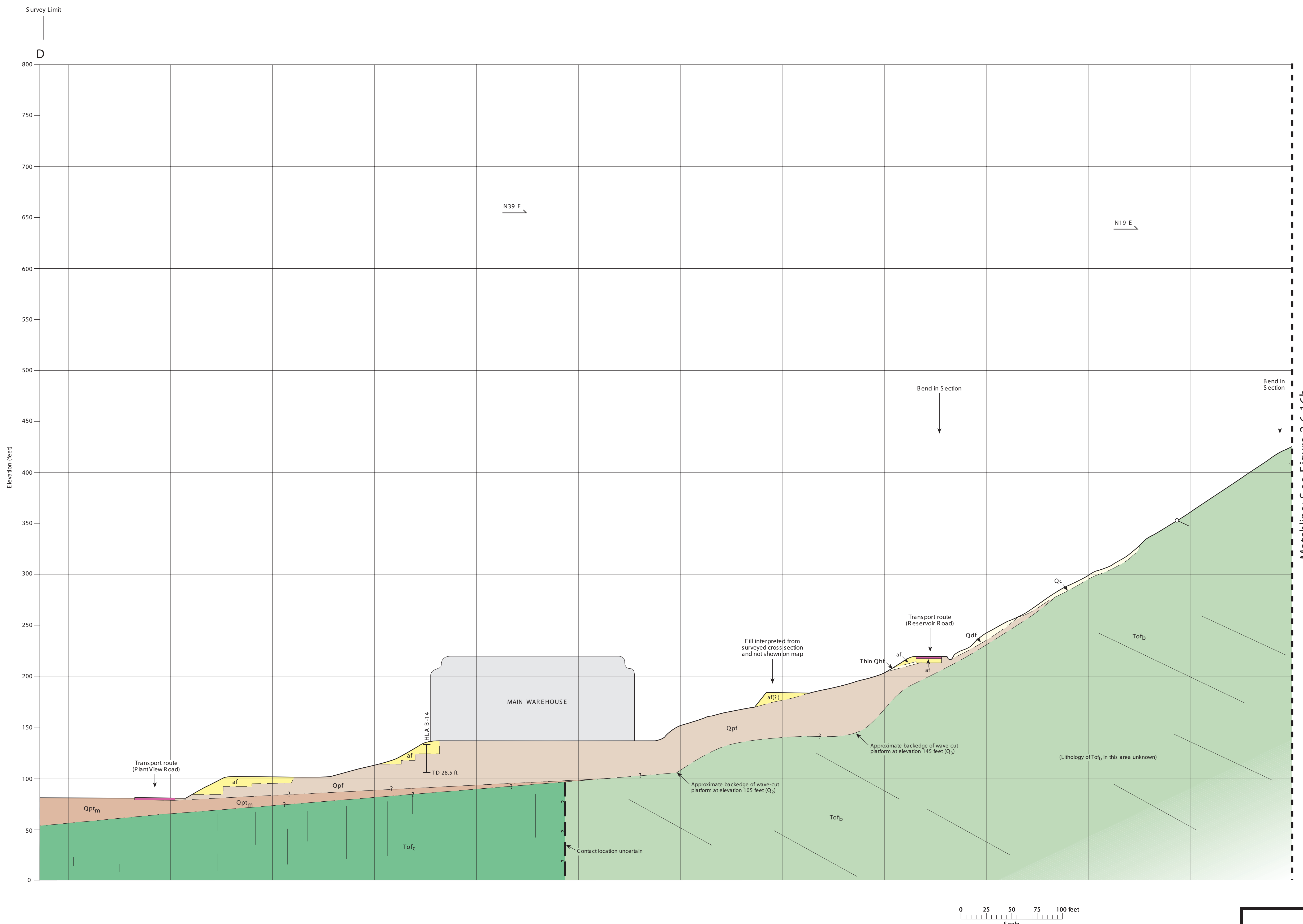

Notes
1. Location of cross section shown on Figures 2.6-7 and 2.6-8. Nearby borings are projected to cross section.
2. See Figure 2.6-9 for explanation of geologic units.
3. Horizontal scale = vertical scale.

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FIGURE 2.6-16a
CROSS SECTION D-D’

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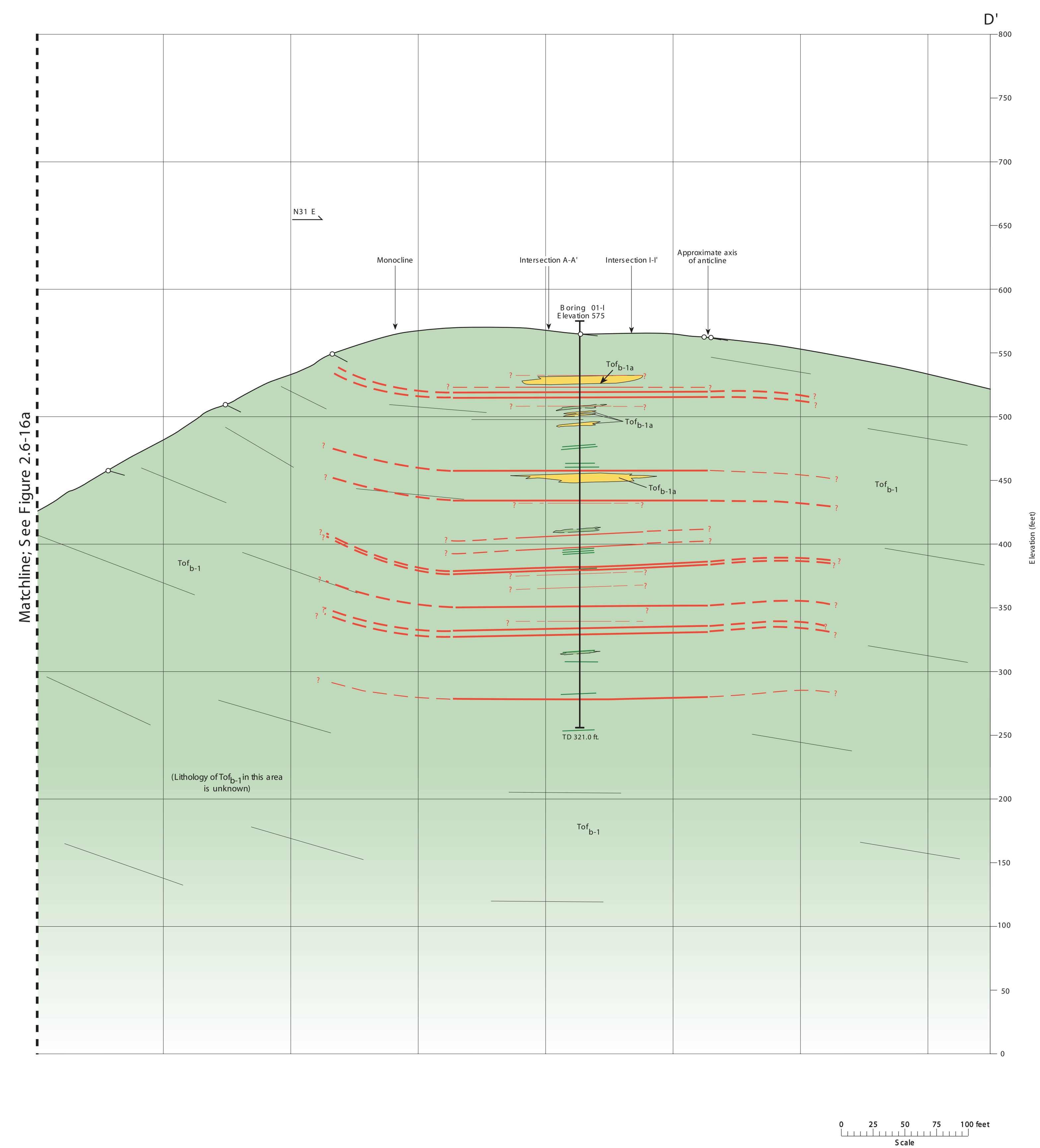

Notes
1. Location of cross section shown on Figures 2.6-7 and 2.6-8. Nearby borings are projected to cross section.
2. See Figure 2.6-9 for explanation of geologic units.
3. Horizontal scale = vertical scale.

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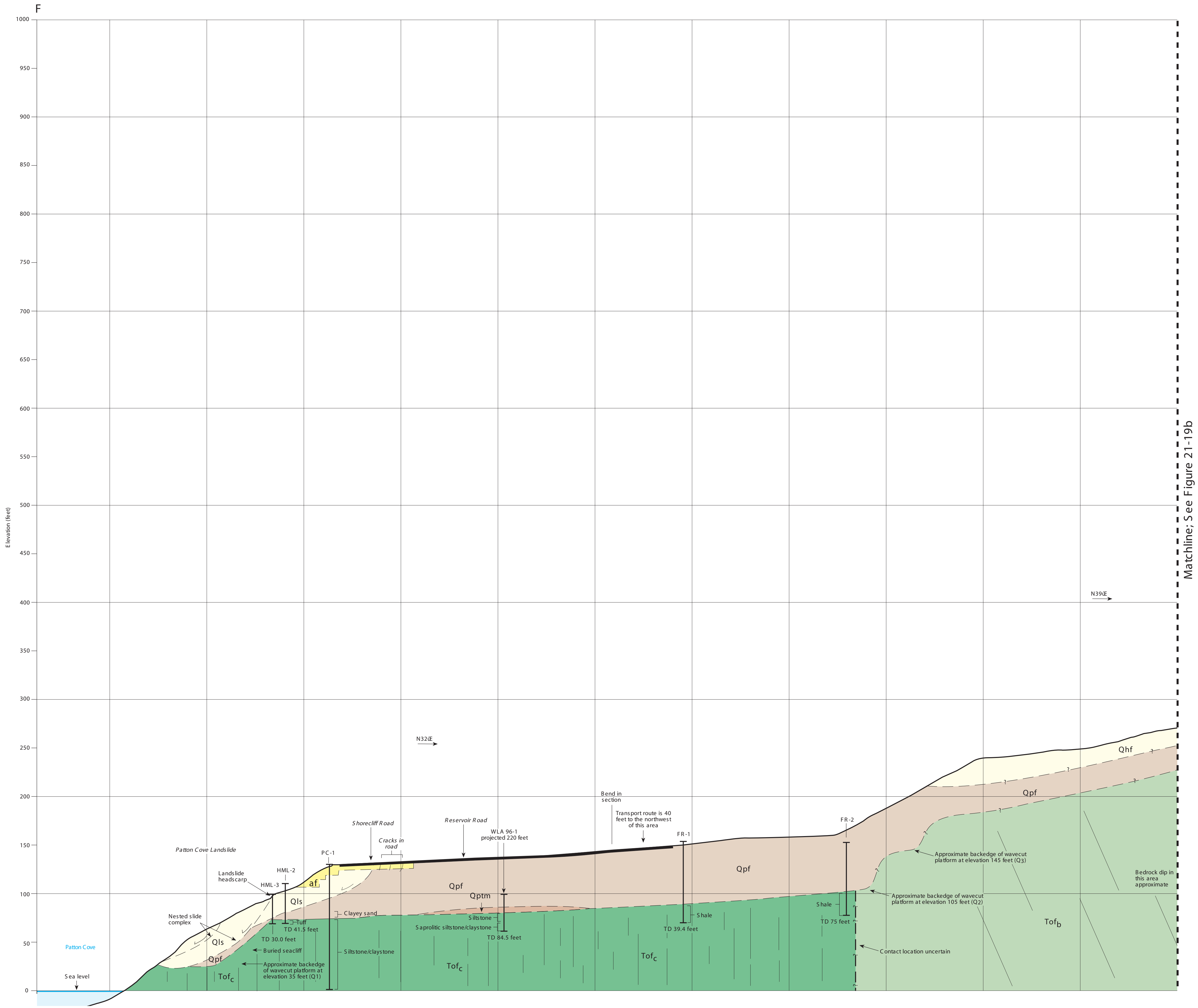
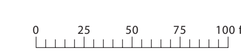
FIGURE 2.6-16b
CROSS SECTION D-D'

Revision 0 June 2004



Notes
1. Location of cross section shown on Figures 21-3 and 21-4. Nearby borings are projected to cross section.
2. See Figure 21-13 for explanation of geologic units.
3. Horizontal scale = Vertical scale.



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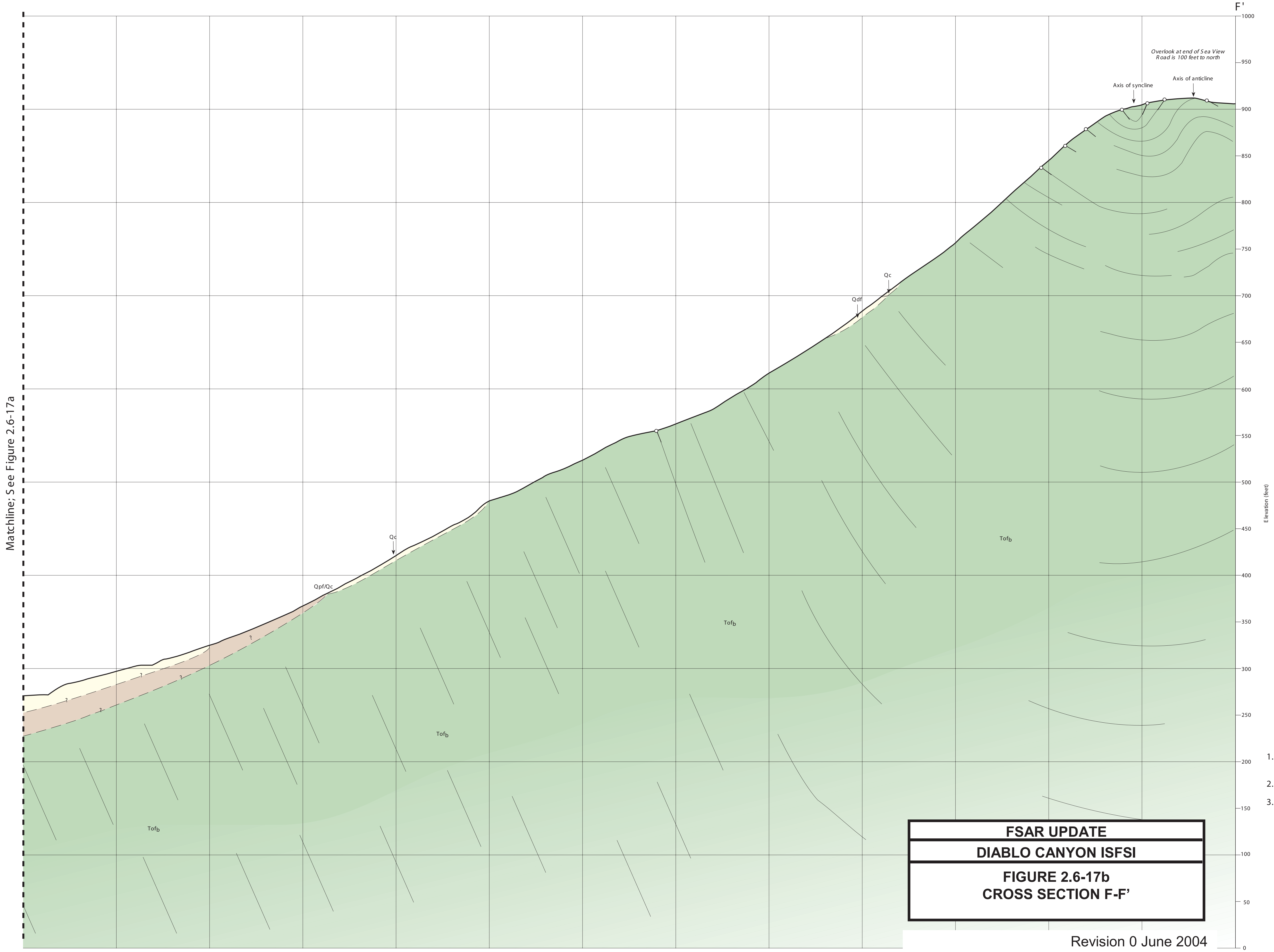
DIABLO CANYON ISFSI
FIGURE 2.6-17a
CROSS SECTION F-F'

Revision 0 June 2004



Notes
1. Location of cross section shown on Figures 21-3 and 21-4. Nearby borings are projected to cross section.
2. See Figure 21-13 for explanation of geologic units.
3. Horizontal scale = Vertical scale.

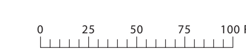


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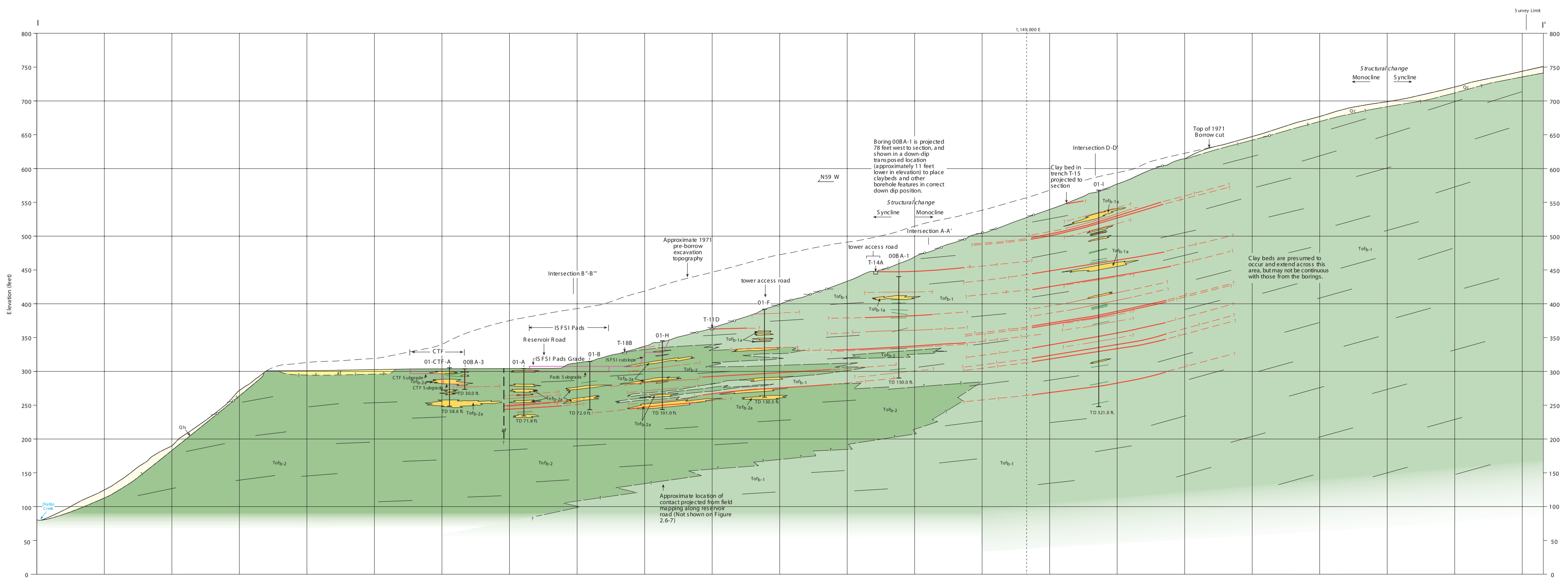
FIGURE 2.6-17b
CROSS SECTION F-F'



Notes

1. Location of cross section shown on Figures 2.6-7 and 2.6-8. Nearby borings are projected to cross section.

2. See Figure 2.6-9 for explanation of geologic units.

3. Horizontal scale = vertical scale.

0 25 50 75 100 feet
Scale

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FIGURE 2.6-18
CROSS SECTION I-I'

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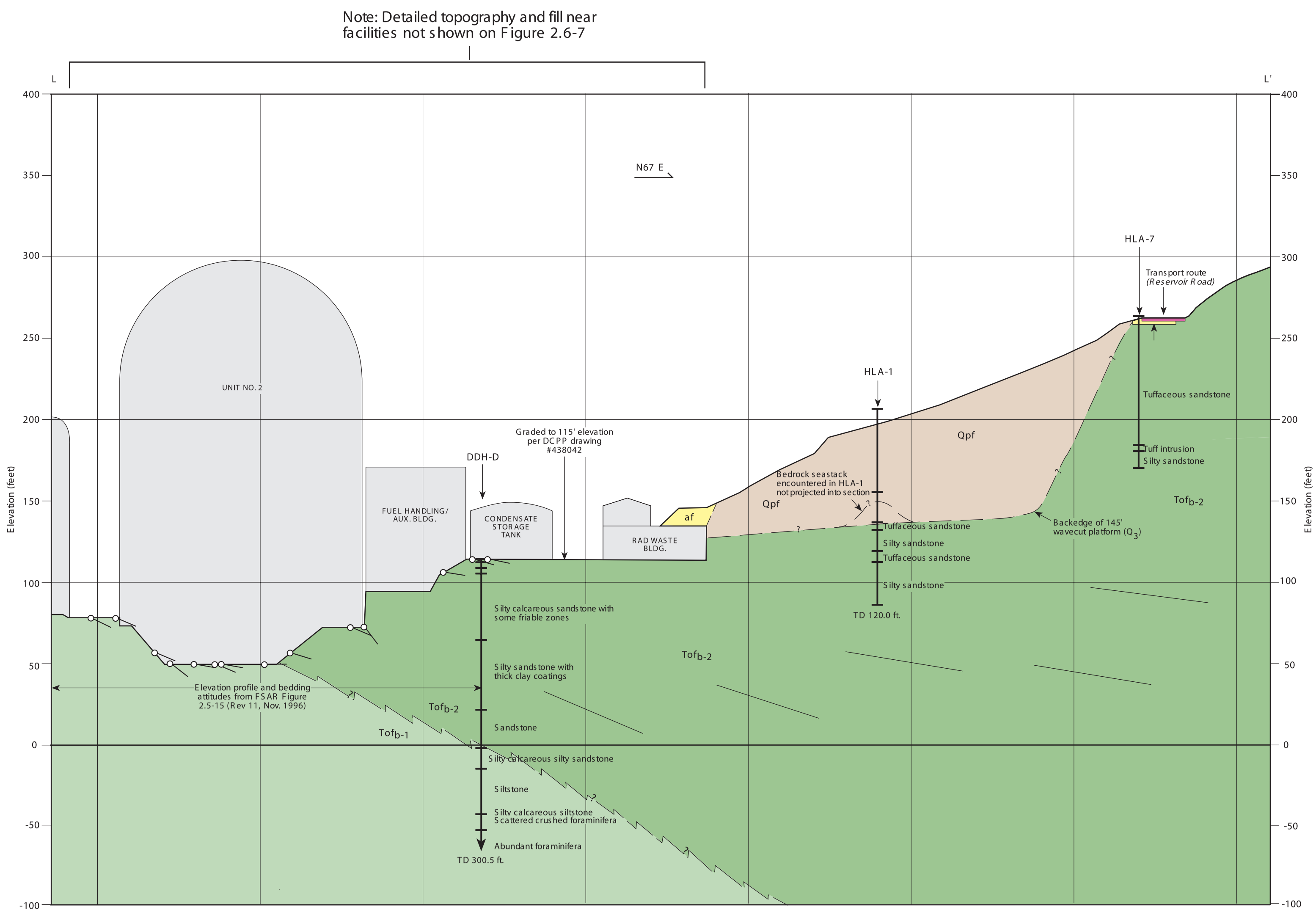

Notes
1. Location of cross section shown on Figures 2.6-7. Nearby borings are projected to cross section.
2. See Figure 2.6-9 for explanation of geologic units.
3. Horizontal scale = vertical scale.

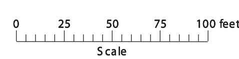

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FIGURE 2.6-19
CROSS SECTION L-L'

Revision 0 June 2004

Dolomite (Tof_{b-1}) exposed along Reservoir Road above parking lot 8. Exposure illustrates well-bedded strata. Some joints terminate at bedding planes (e.g., in left middle). Gray is unweathered, and brown is weathered rock. Photo roll JLB-4.

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FIGURE 2.6-20 CLOSE-UP VIEW OF WELL-BEDDED DOLOMITE ALONG RESERVOIR ROAD

Outcrop of thick to massive bedded, weathered sandstone of unit (Tof_{b-2}), directly west of the ISFSI. Photo roll JLB OLD-2.

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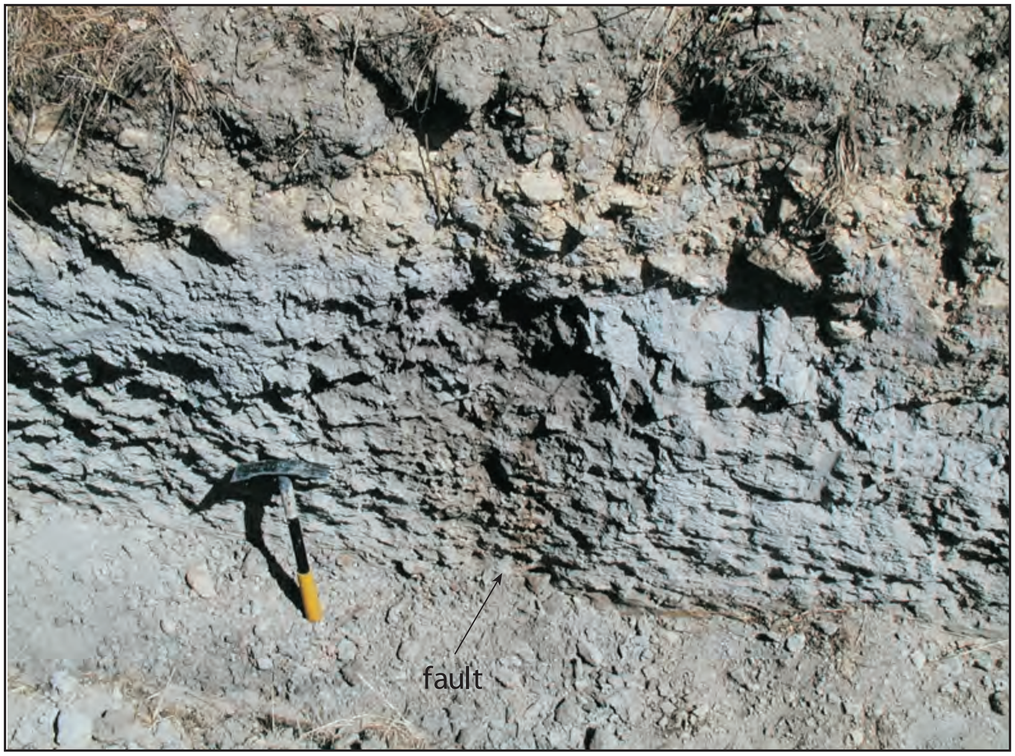
FIGURE 2.6-21 SANDSTONE OUTCROP IN THE ISFSI STUDY AREA



Friable sandstone (Tof_{b-2a}) in trench T-1. The friable sandstone generally is weakly bedded and jointed. A small near-vertical fault is indicated by oxidized clay stringers in the sandstone. Photo roll JLB-3.

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FIGURE 2.6-22 FRIABLE SANDSTONE IN TRENCH T-1

Clay bed within dolomite (Tof_{b-1}) with sample tube in trench T-14B. Photo roll JLB-8.

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FIGURE 2.6-23
CLAY BED IN TRENCH T-14B

Typical dolomite Tof_{b-1} and thin clay beds exposed in trench T-11C. Clay beds are subhorizontal and define bedding. Photo roll 01JLB-1.

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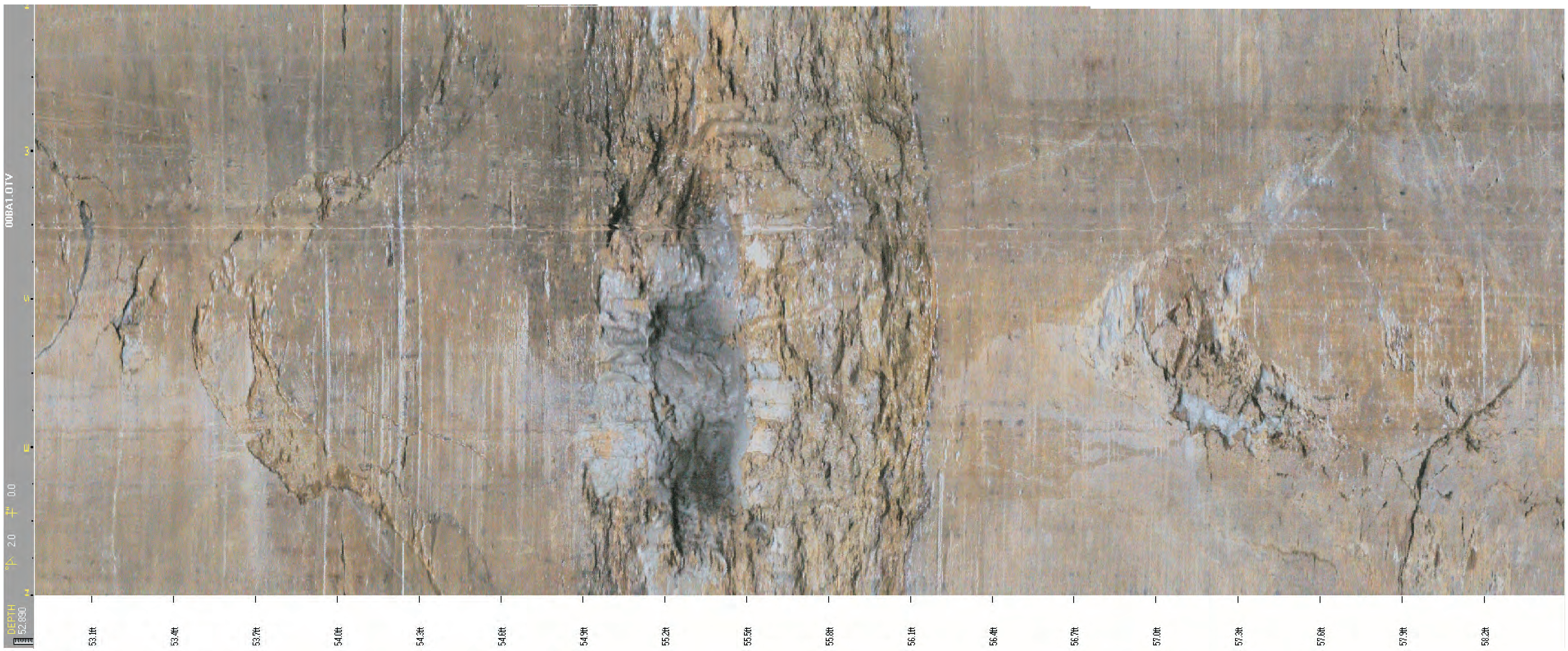
FIGURE 2.6-24 CLAY BEDS AND DOLOMITE IN TRENCH T-11C

Right margin



Left margin

Optical televiewer image of 50.0 to 58.5 feet, boring 00BA-1. Image "unwraps" the round borehole and displays picture as flat; right and left margins are the same vertical line in the boring. Clay bed at 55 feet (center of photo between 54.9 and 56.1 feet) is within the dolomite (Tof_{b-1}). Fractures in the dolomite that intercept the boring at a steep angle appear as sinusoidal shapes. The fracture between 53.7 and 54.6 feet dips 65 degrees and is partly filled with clay where it breaks into two or more joints. Three intersecting joints are present between 57.0 and 58.3 feet; drilling has broken parts of rock between the subparallel joints near 57.3 feet. (12/5/00)

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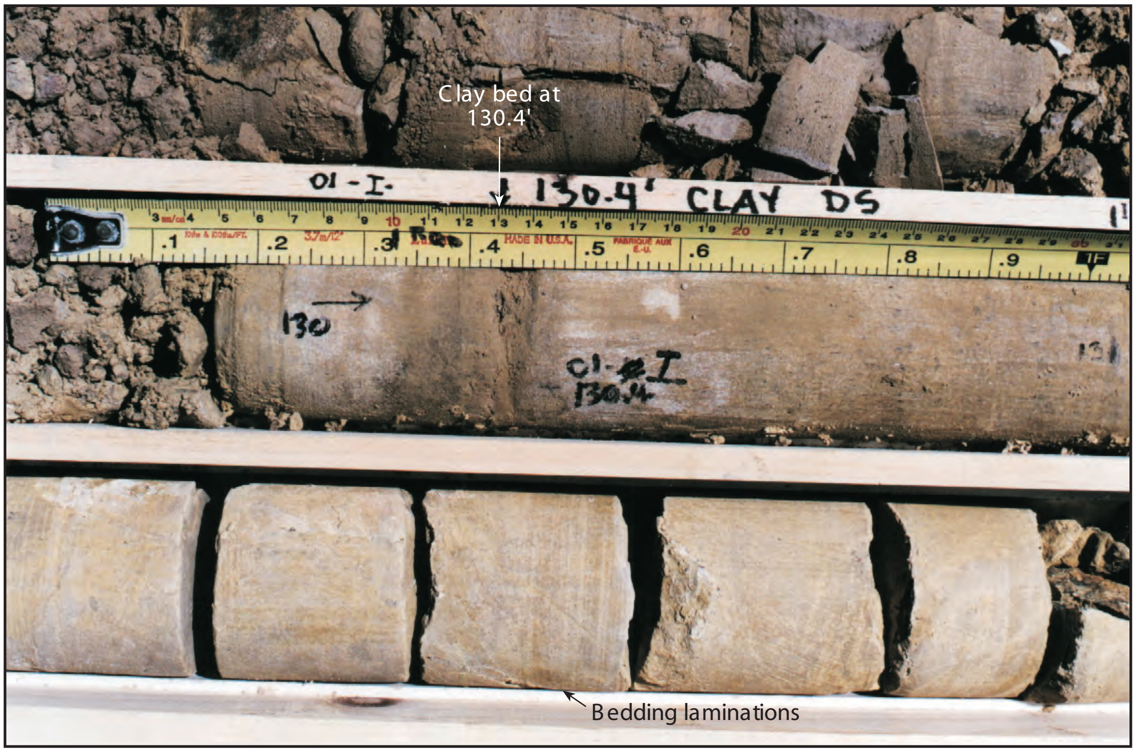
FIGURE 2.6-25 TELEVIEWER IMAGE OF CLAY BED AT 55 FEET, BORING 00BA-1



Typical appearance of clay bed and bedding laminations in a section of core at 130 feet from boring 01-I, south of the ISFSI. Clay bed occurs within Tof_{b-1}.
Photo roll 01JLB-ba.

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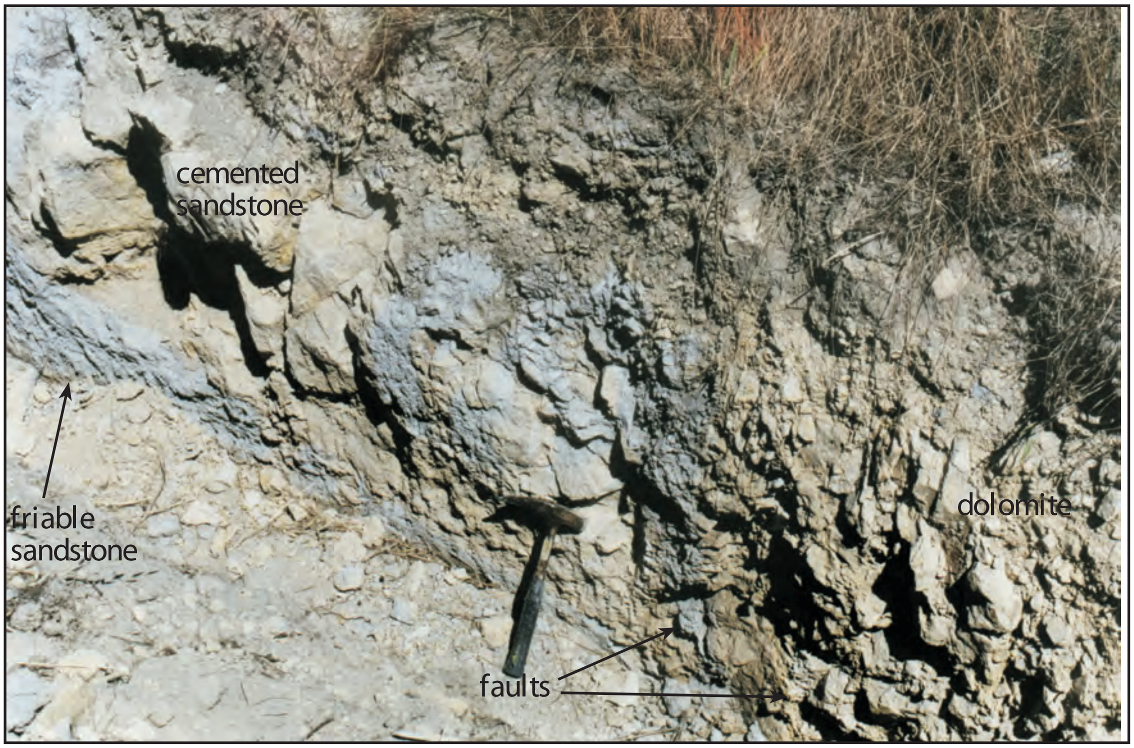
FIGURE 2.6-26 CLAY BED AT 130 FEET IN BORING 01-1



Typical small bedrock faults in trench T-1. The faults juxtapose friable sandstone (Tof_{b-2a}) on left against fractured dolomite (Tof_{b-1}) on right. A remnant of unaltered, cemented sandstone (Tof_{b-2}) remains in upper left. Photo roll JLB-4.

FSAR UPDATE
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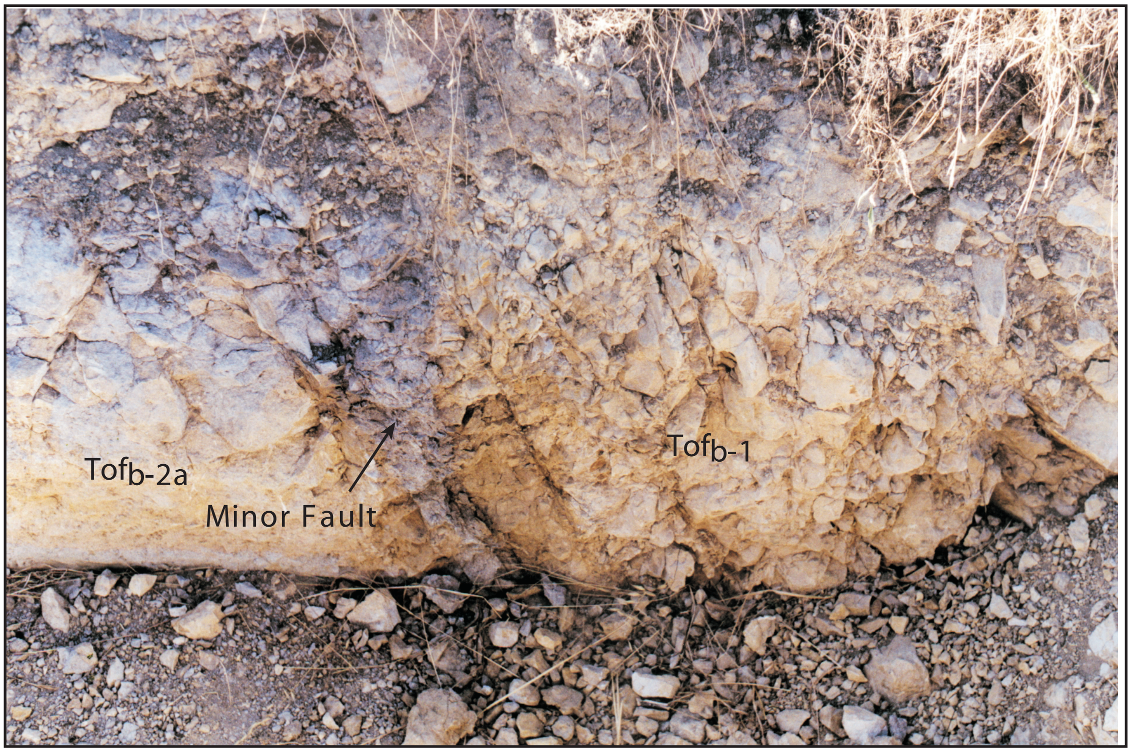
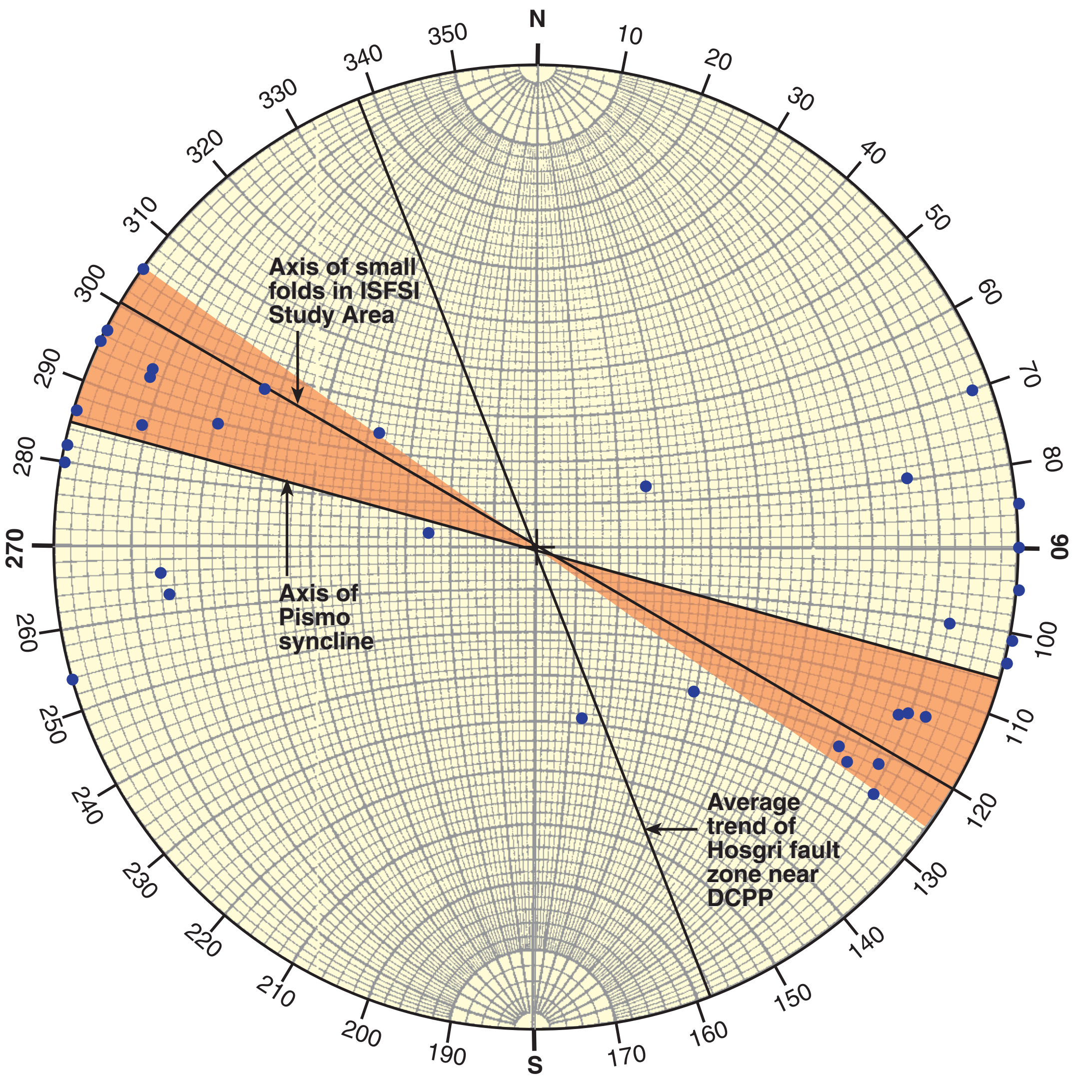
FIGURE 2.6-27 TYPICAL SMALL BEDROCK FAULTS IN TRENCH T-1



Minor fault in trench T-1 juxtaposing friable sandstone (Tof$_{b-2a}$) on left against dolomite (Tof$_{b-1}$). Photo roll JLB-2.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-28 MINOR FAULT IN TRENCH T-1



Explanation

General range in strike of zone of minor faults.

Rake of slickenside on fault plane of minor faults

Equal-angle lower hemisphere plot.

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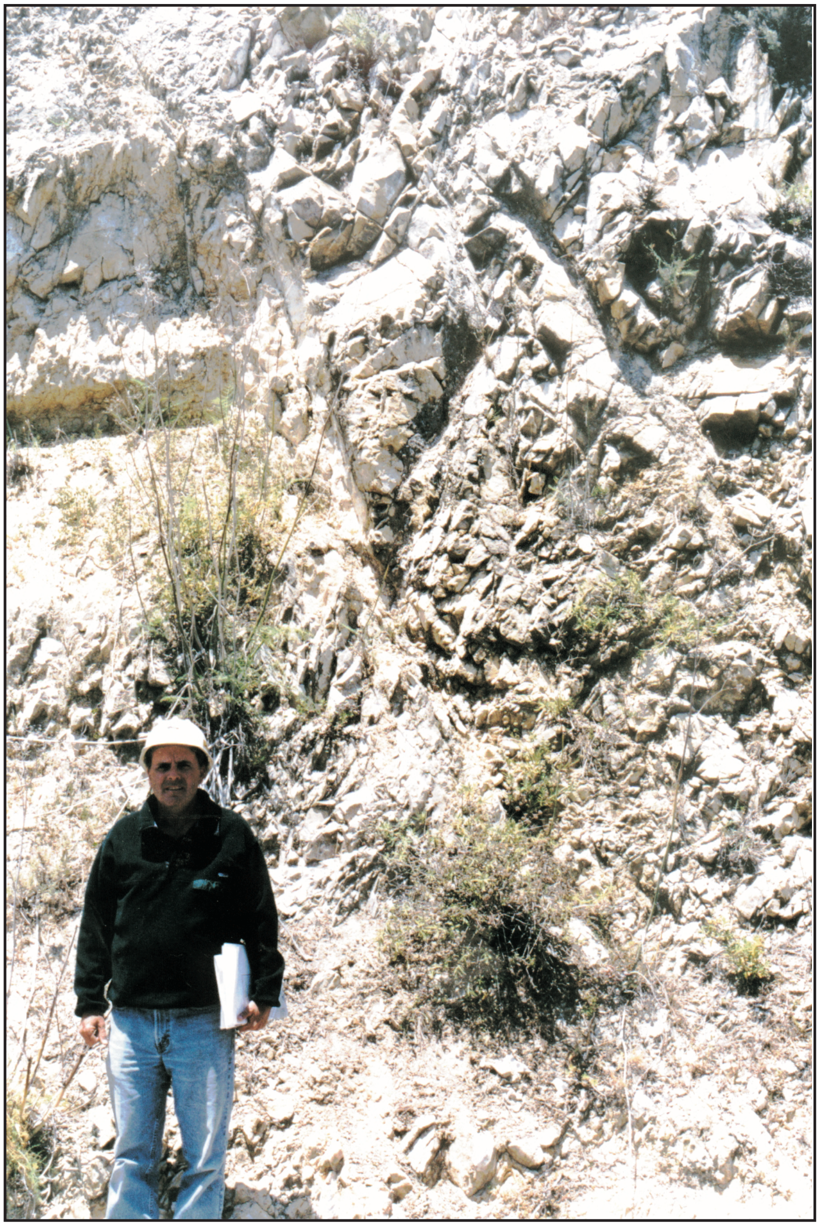
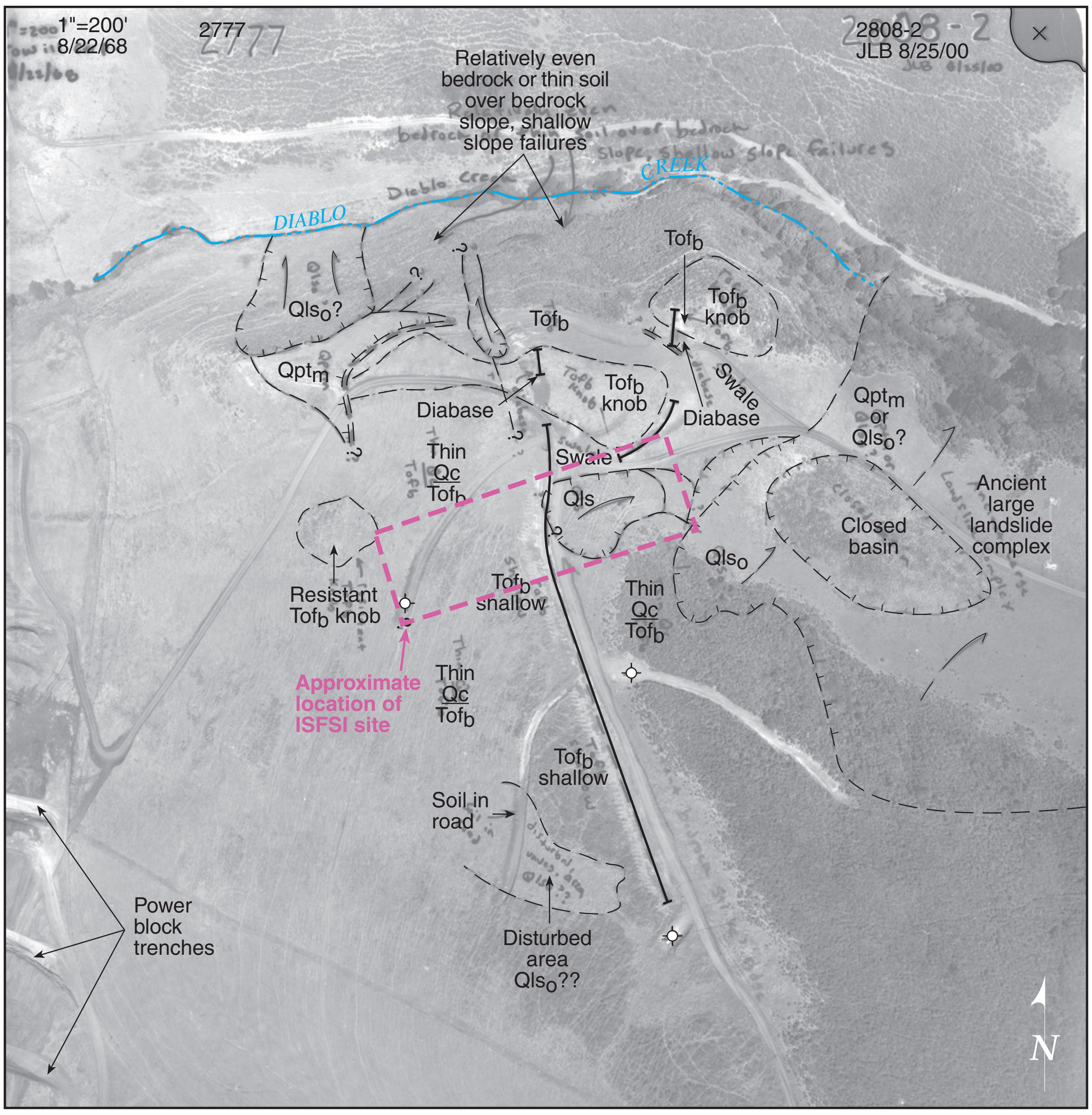
DIABLO CANYON ISFSI
FIGURE 2.6-29 COMPARISON OF ORIENTATIONS OF MINOR FAULTS AND FOLDS IN THE ISFSI STUDY AREA WITH OTHER STRUCTURES

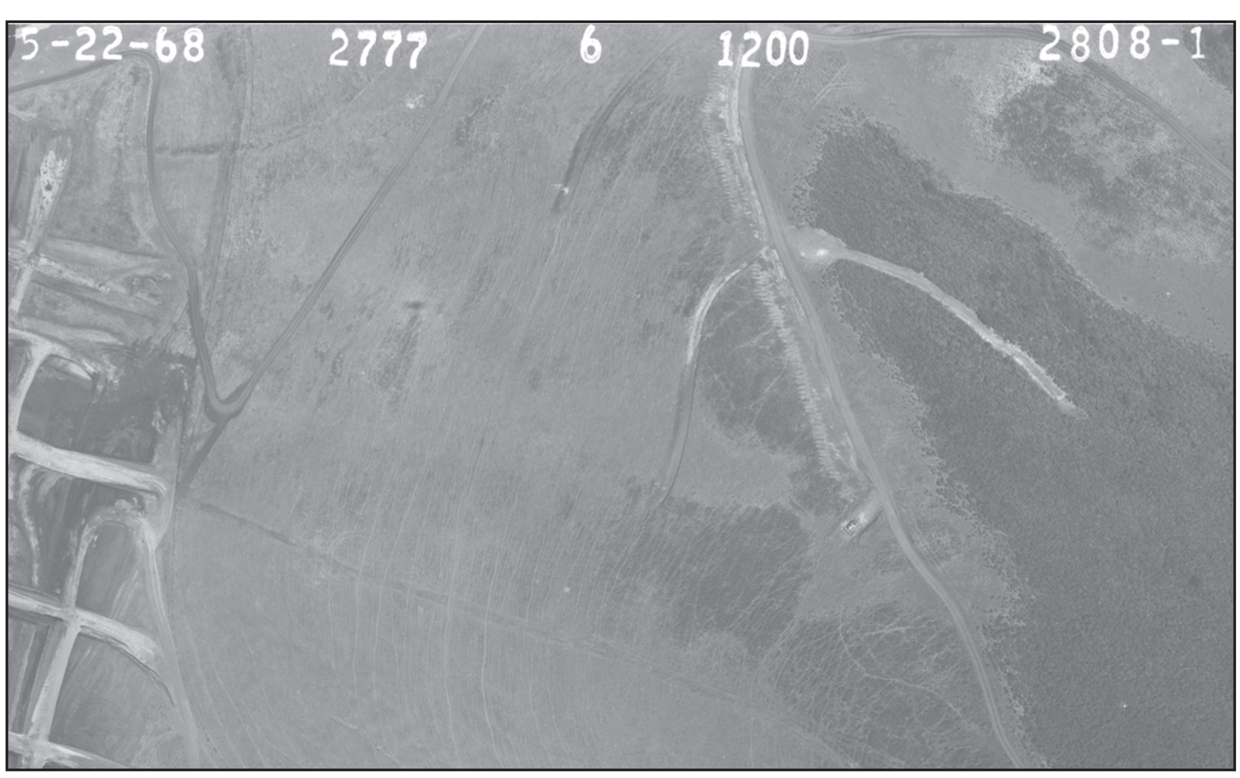


Northward view of Diablo Creek Road cut showing steeply dipping minor faults in dolomite of unit Tof_{b-1}. Slickensides and mullions on the fault plane indicate primarily strike-slip displacement, but bedding also suggests a component of down-to-the- east vertical separation of approximately 3 to 6 feet. These faults are located along projection of faults exposed in trenches at the ISFSI, approximately 800 feet to the southeast, that have similar strike and slickenside/mullion rakes. Photo roll JLB 5/16-1.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-30 MINOR FAULTS ALONG DIABLO CREEK ROAD



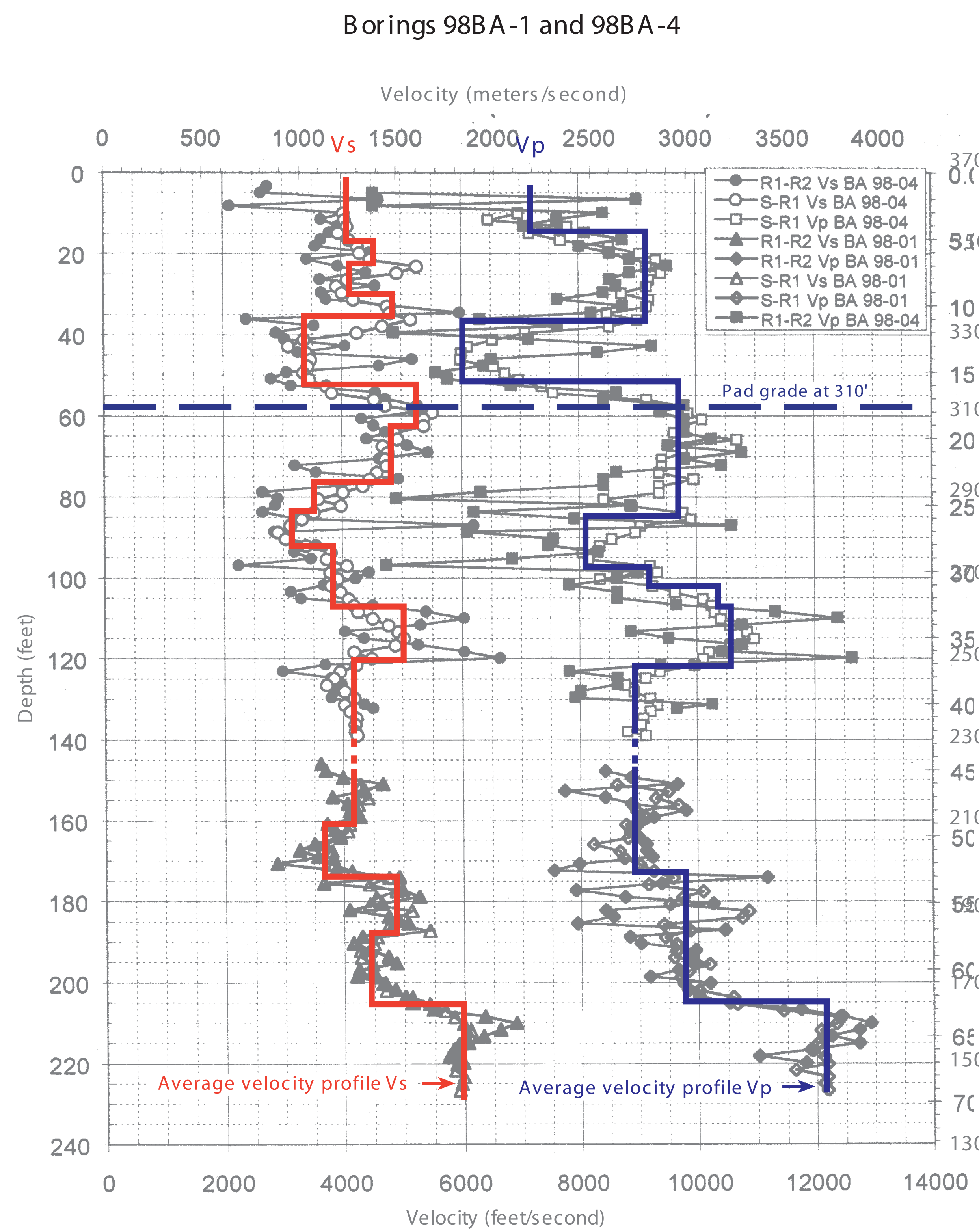
1968 stereo air photos (2777; 2808-1 and 2808-2) of ISFSI study area prior to the 1971 excavation of the borrow site. Diablo Creek traverses the upper (northern) part of the photo. Trenches for the power block are evident in the lower left. The road that follows the ridge crest in center of photo was removed during 1971 excavation of the borrow area. No features indicating deep seated landslides are present at the site; large landslides are evident to the east, however. The small landslide south of the word "swale" is shallow and was removed in the 1971 excavations. See Figure 2.6-7 for unit descriptions. To view with a stereoscope, fold and adjust the photos as necessary.

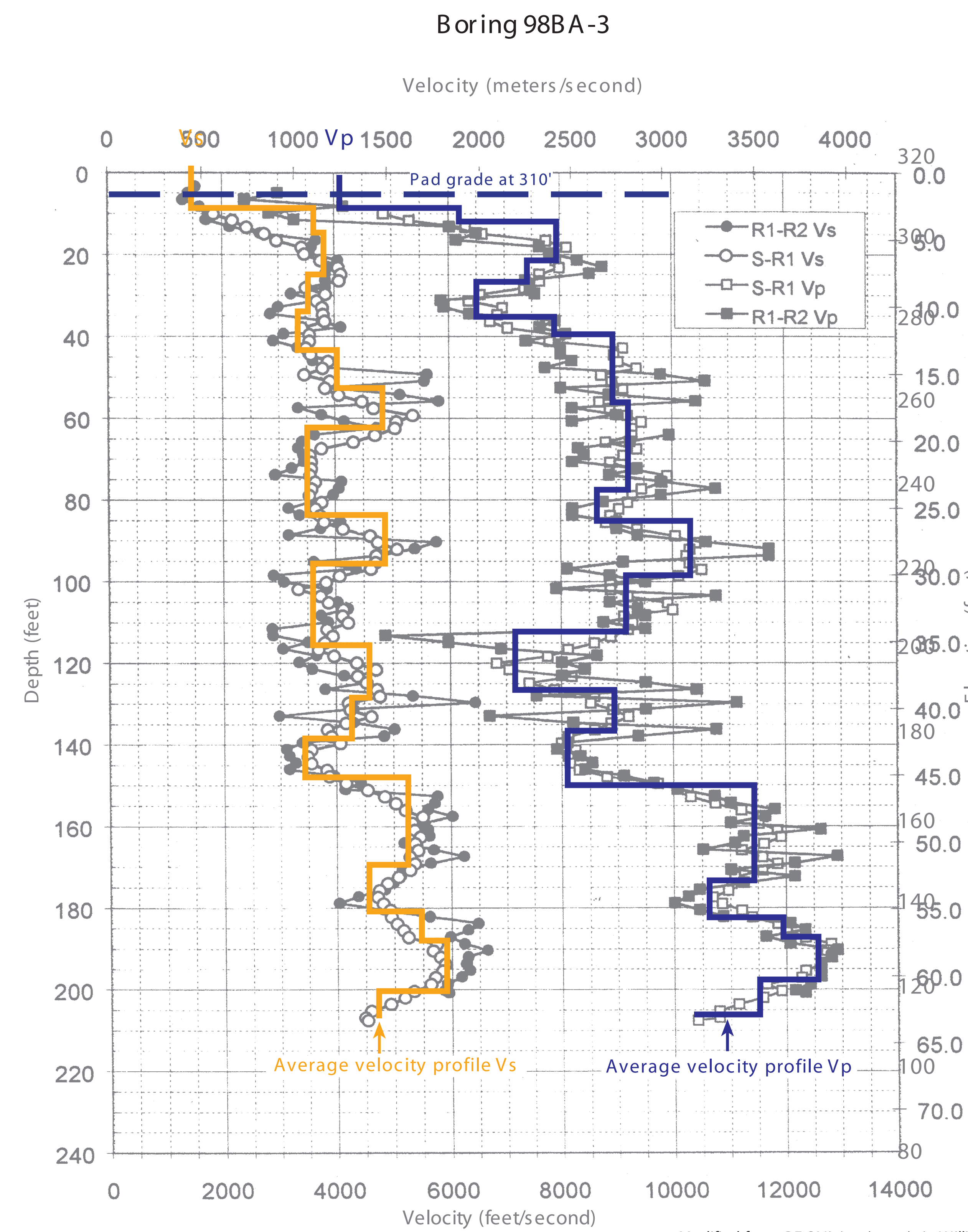


FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-31 1968 AERIAL STEREO PHOTOGRAPHY OF ISFSI STUDY AREA

Security-Related Information Figure Withheld Under 10 CFR 2.390.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-32 GEOLOGY OF ISFSI AND CTF SITES AT PROPOSED FINAL GRADES





Note: Average velocity profiles interpreted from data

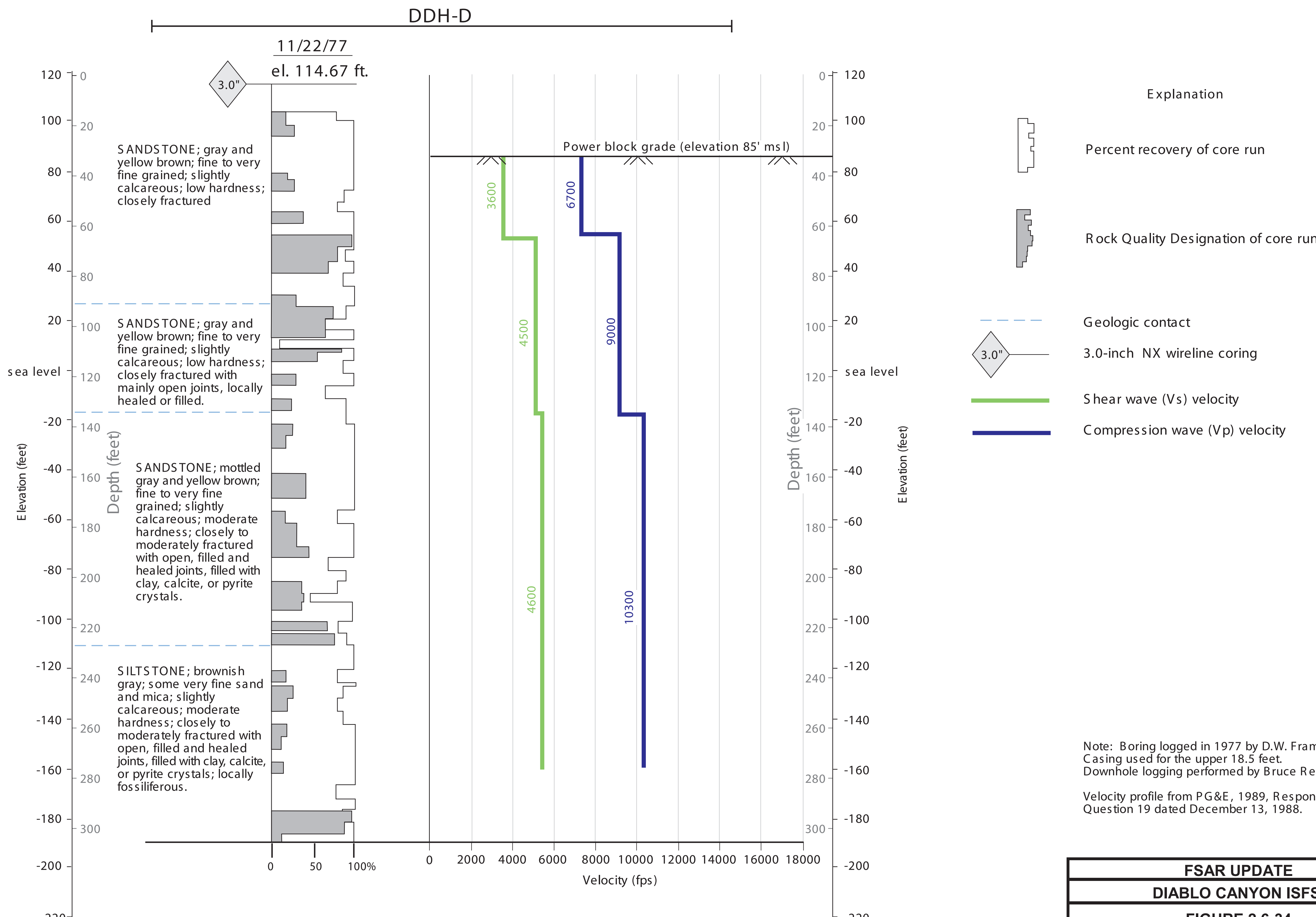
R1 - R2 = Receiver-to-receiver velocity (3.3-foot spacing)
S-R1 = Source-to-receiver velocity (10.3-foot spacing)

Modified from GEOVision (1998), in William Lettis & Assoc. Inc., 2001, DCPP ISFSI Data Report C.

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.6-33
ISFSI SITE SUSPENSION LOGS AND INTERPRETED AVERAGE SEISMIC VELOCITIES



Note: Boring logged in 1977 by D.W. Frames.
Casing used for the upper 18.5 feet.
Downhole logging performed by Bruce Redpath.

Velocity profile from PG&E, 1989, Response to NRC Question 19 dated December 13, 1988.

FSAR UPDATE

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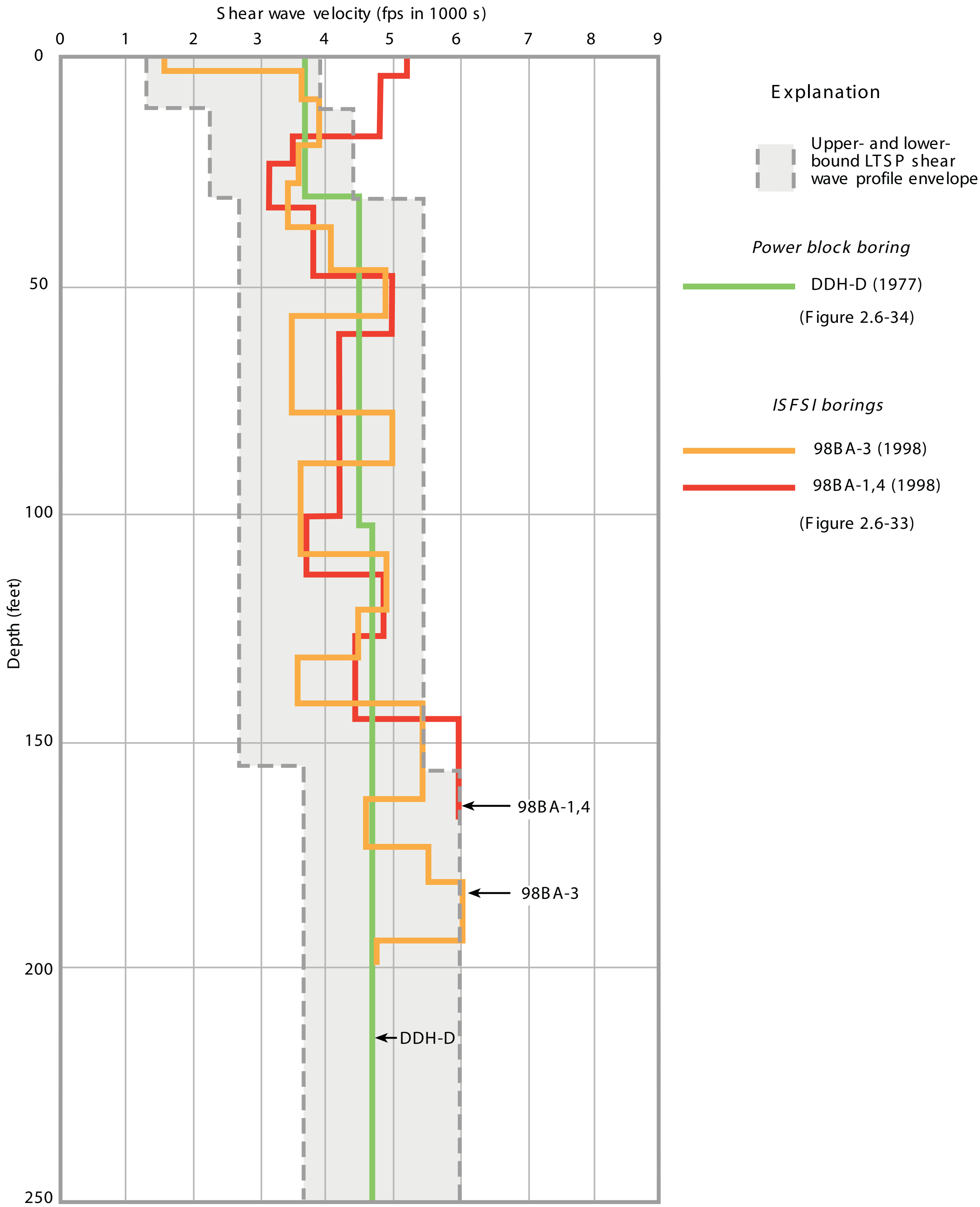
FIGURE 2.6-34
SUMMARY LOG OF
1977 POWER BLOCK BORING DDH-D

Revision 0 June 2004



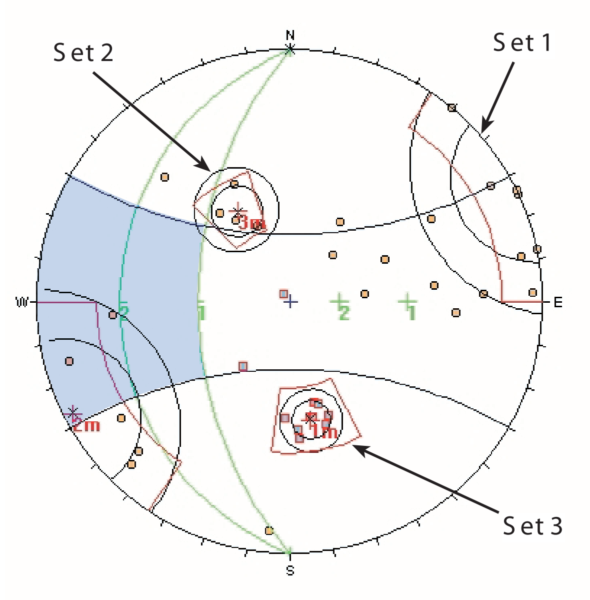
FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.6-35
COMPARISON OF SEISMIC SHEAR-WAVE VELOCITIES AT THE POWER BLOCK AND ISFSI SITES

Security-Related Information Figure Withheld Under 10 CFR 2.390.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-36 TRANSPORT ROUTE NEAR PATTON COVE LANDSLIDE



A. Topple hazard (moderate hazard)

Explanation

Poles

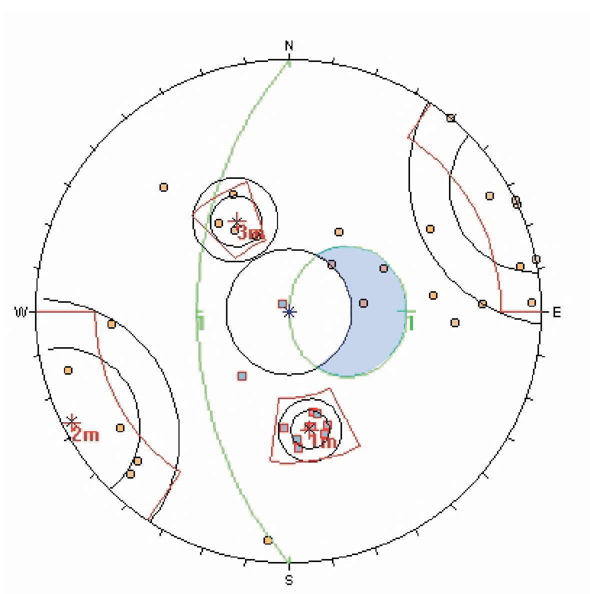
Joint

Bedding

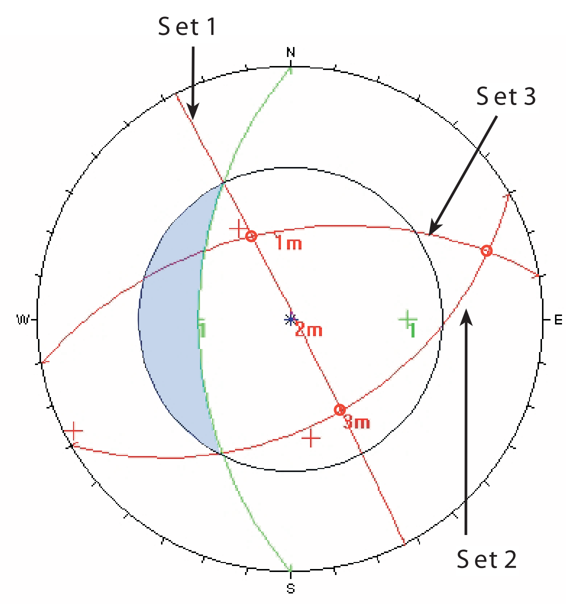
Failure envelope
(based on 28
friction angle)

Failure envelope for topple and planar sliding without poles indicates stable conditions.

Failure envelope for wedge sliding without great circle intersections indicates stable conditions.



B. Planar sliding hazard (low hazard)



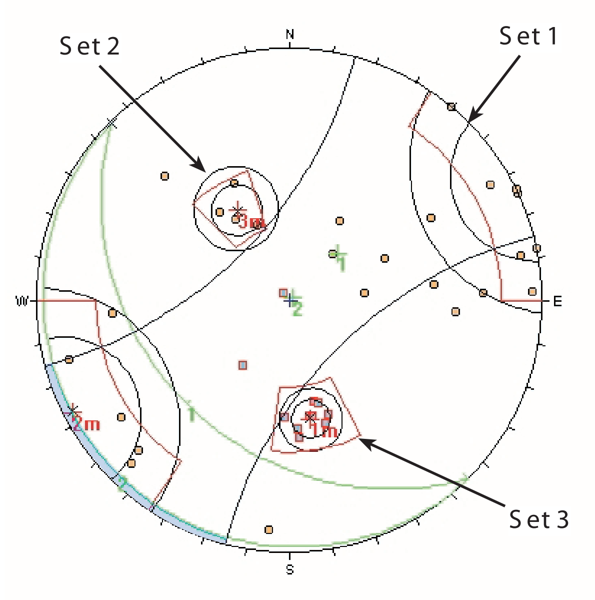
C. Wedge sliding hazard (very low hazard)

Notes

Analysis performed using computer program DIPS (Rocscience, 1999, DIPS: Plotting analysis, and presentation of structural data using spherical projection techniques, version 5.041, Toronto, 86p).

Fracture data from stations 38+00 to 45+00 applied to north-trending cutslope above Reservoir Road from stations 43+00 to 46+00.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-37 KINEMATIC ANALYSES OF NORTH-TRENDING CUTSLOPE OF TRANSPORT ROUTE (STATIONS 43+00 TO 46+00



A. Topple hazard (low hazard)

Explanation

Poles

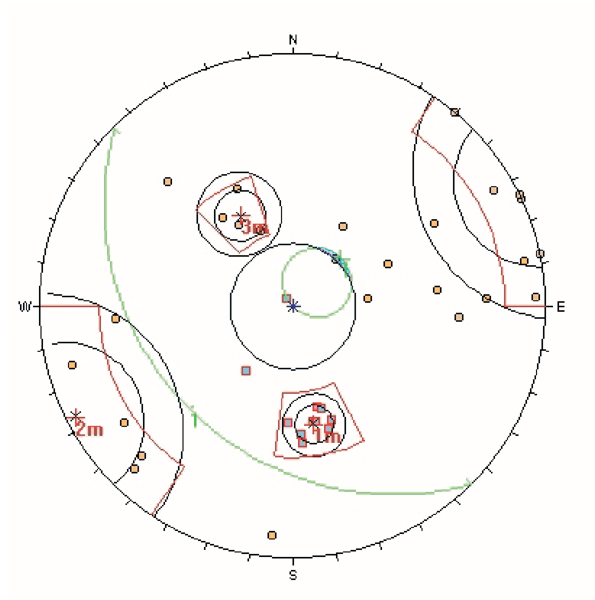
Joint

Bedding

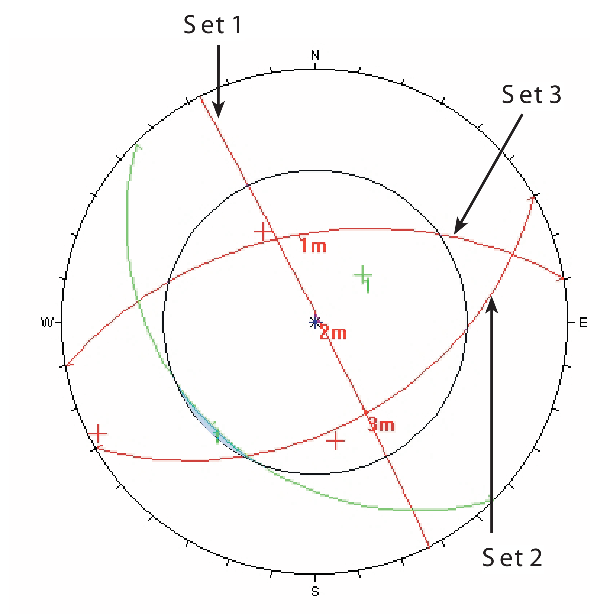
Failure envelope (based on 28 friction angle)

Failure envelope for topple and planar sliding without poles indicates stable conditions.

Failure envelope for wedge sliding without great circle intersections indicates stable conditions.



B. Planar sliding hazard (very low hazard)



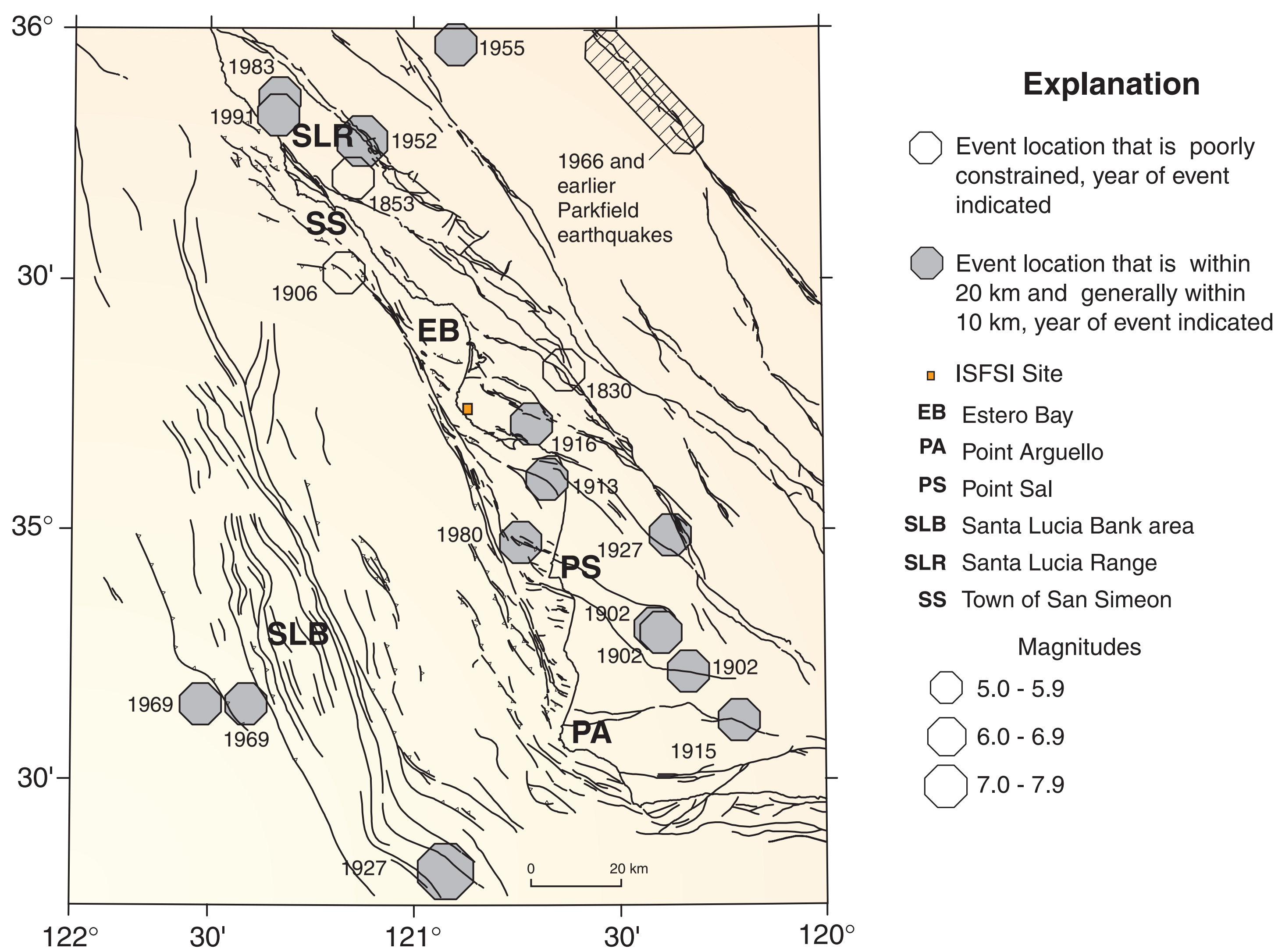
C. Wedge sliding hazard (very low hazard)

Notes

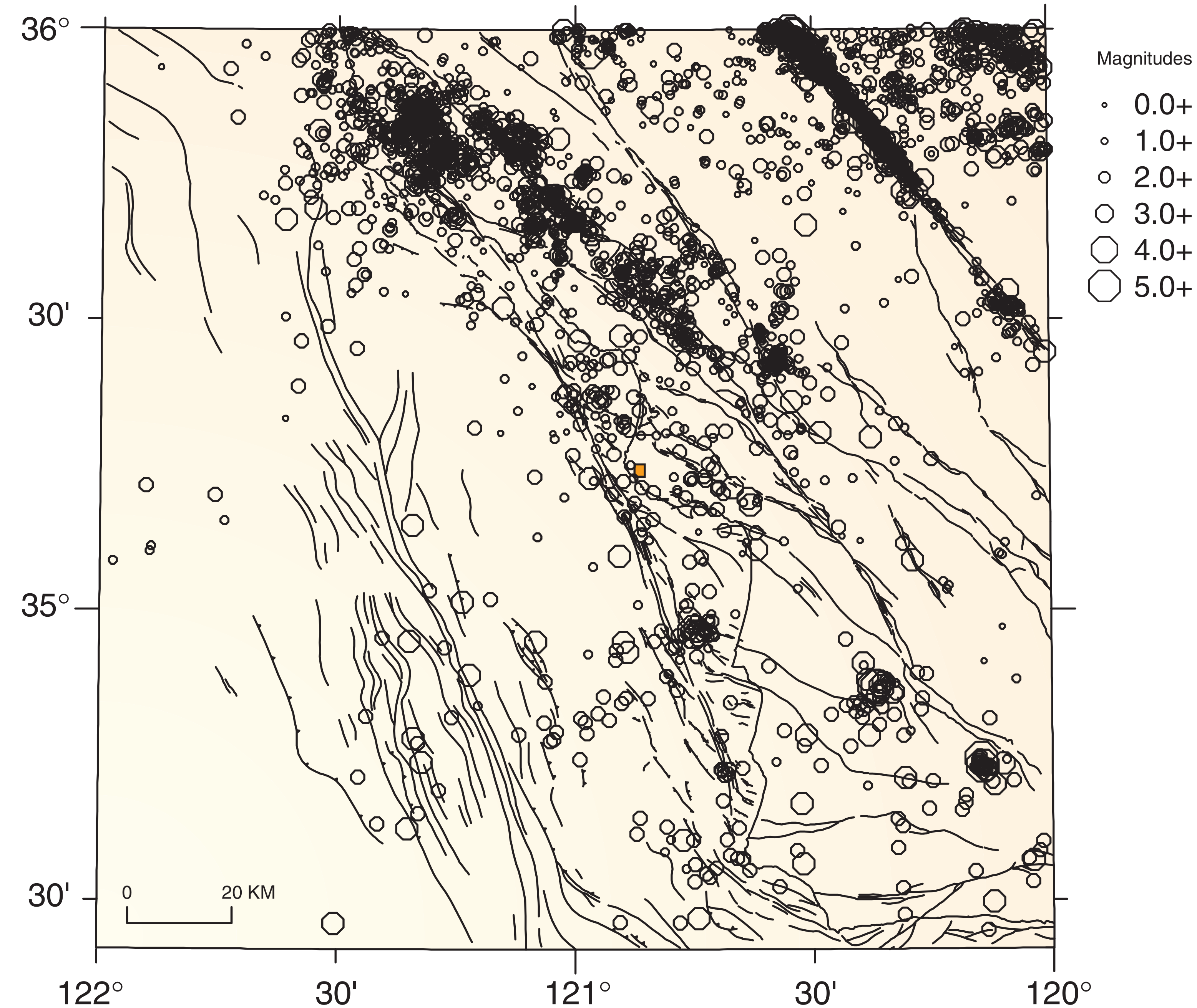
Analysis performed using computer program DIPS (Rocscience, 1999, DIPS: Plotting analysis, and presentation of structural data using spherical projection techniques, version 5.041, Toronto, 86p).

Fracture data from stations 38+00 to 45+00 applied to northwest-trending cutslope above Reservoir Road from stations 35+00 to 43+00.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-38 KINEMATIC ANALYSES OF NORTHWEST-TRENDING CUTSLOPE OF TRANSPORT ROUTE (STATIONS 35+00 TO 43+00

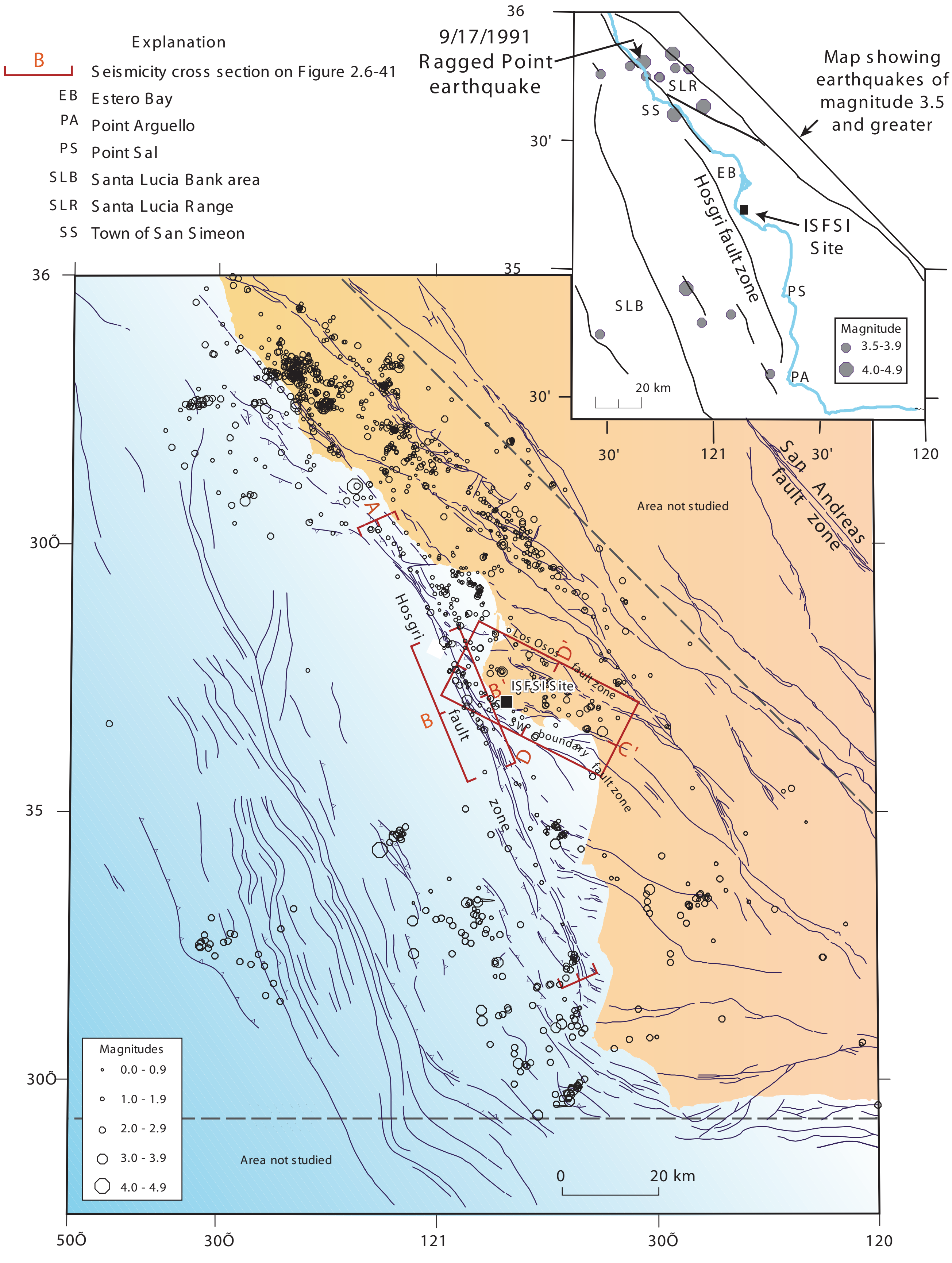


A. Historical earthquakes of magnitude 5 and greater since 1830 (PG&E, Final Report of the Diablo Canyon Long Term Seismic Program, 1988.)



B. Instrumentally recorded seismicity from 1973 through September 1987 (PG&E, Final Report of the Diablo Canyon Long Term Seismic Program, 1988.)

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-39 HISTORICAL EARTHQUAKES OF MAGNITUDE 5 AND GREATER SINCE 1830 AND INSTRUMENTALLY RECORDED SEISMICITY FROM 1973 THROUGH SEPTEMBER 1987



(From M.K. Mc Laren and W.U. Savage, Seismicity of south-central coastal California, October 1987 through January 1997, Bulletin of the Seismological Society of America, in press)

FSAR UPDATE

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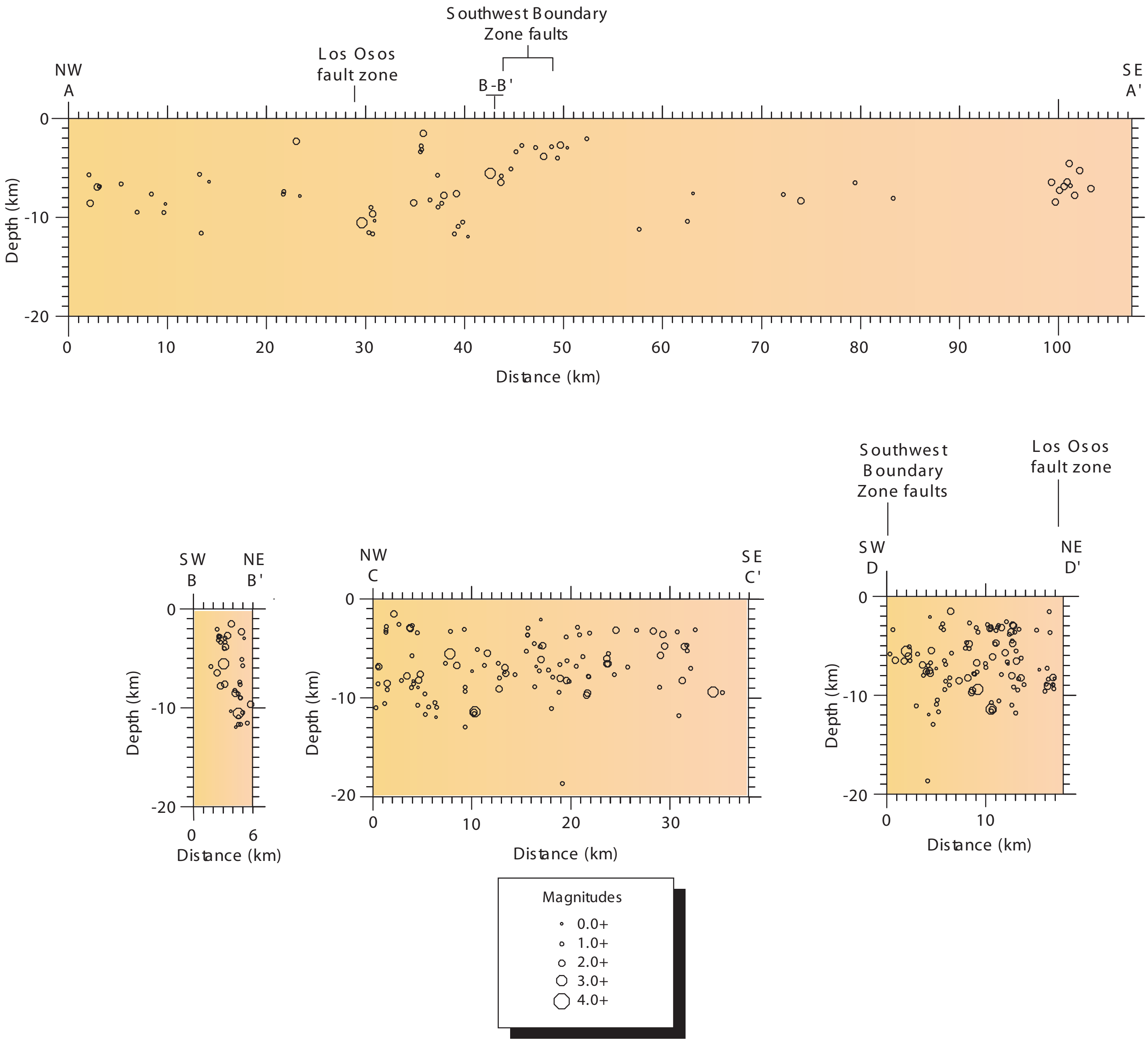
FIGURE 2.6-40
QUATERNARY FAULTS AND SEISMICITY FROM OCTOBER 1987 THROUGH JANUARY 1997



(From M.K. Mc Laren and W.U. Savage, Seismicity of south-central coastal California, October 1987 through January 1997, Bulletin of the Seismological Society of America, in press)

FSAR UPDATE

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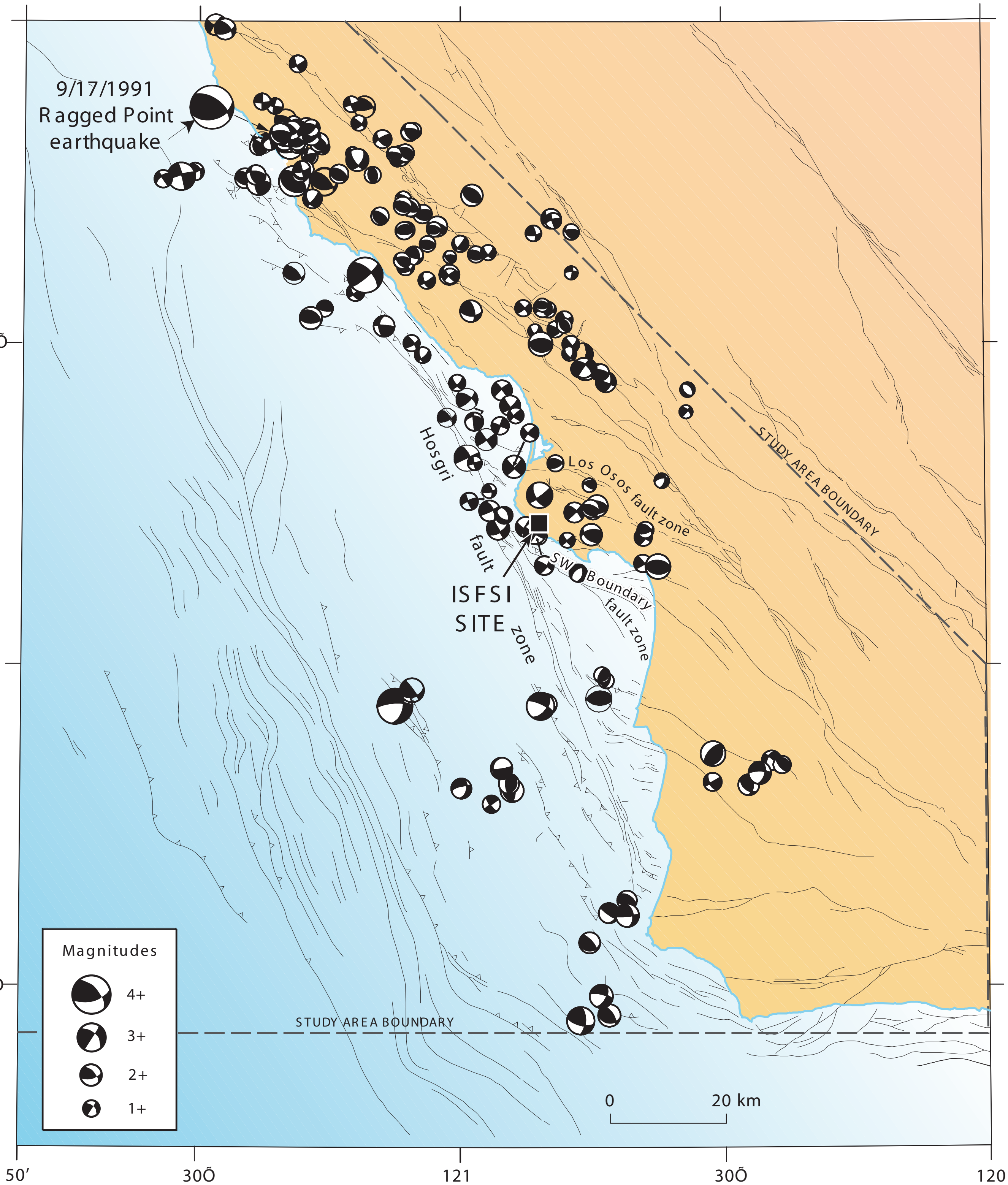
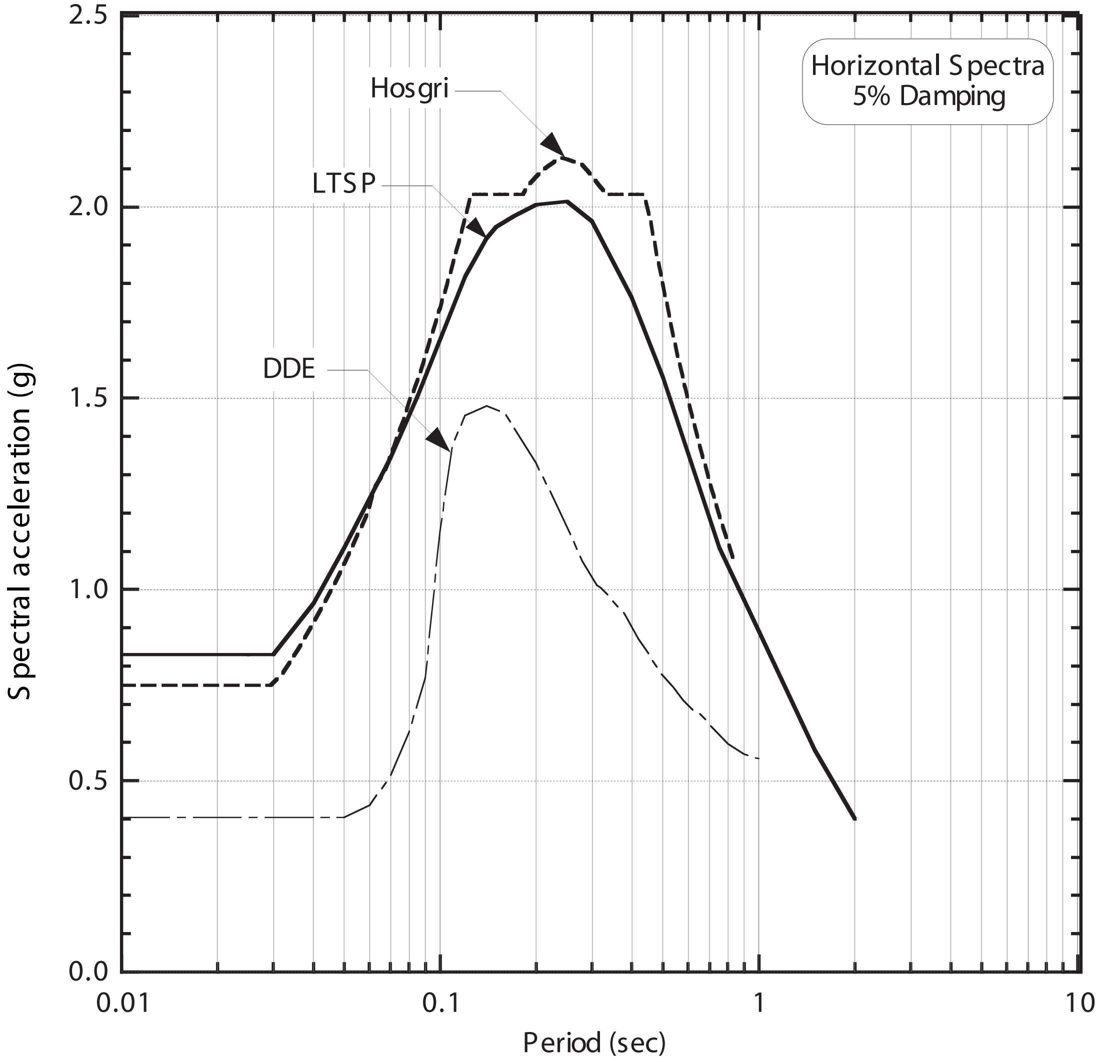
FIGURE 2.6-41
SEISMICITY CROSS SECTION A-A' THROUGH D-D' FOR EARTHQUAKES FROM OCTOBER 1987 THROUGH JANUARY 1997



(From M.K. Mc Laren and W.U. Savage, Seismicity of south-central coastal California, October 1987 through January 1997, Bulletin of the Seismological Society of America, in press)

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-42 LOWER HEMISPHERE, P-WAVE, FIRST-MOTION FOCAL MECHANISM PLOTS OF EARTHQUAKES FROM OCTOBER 1987 THROUGH JANUARY 1997



FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-43 DDE, HOSGRI, AND LTSP HORIZONTAL SPECTRA

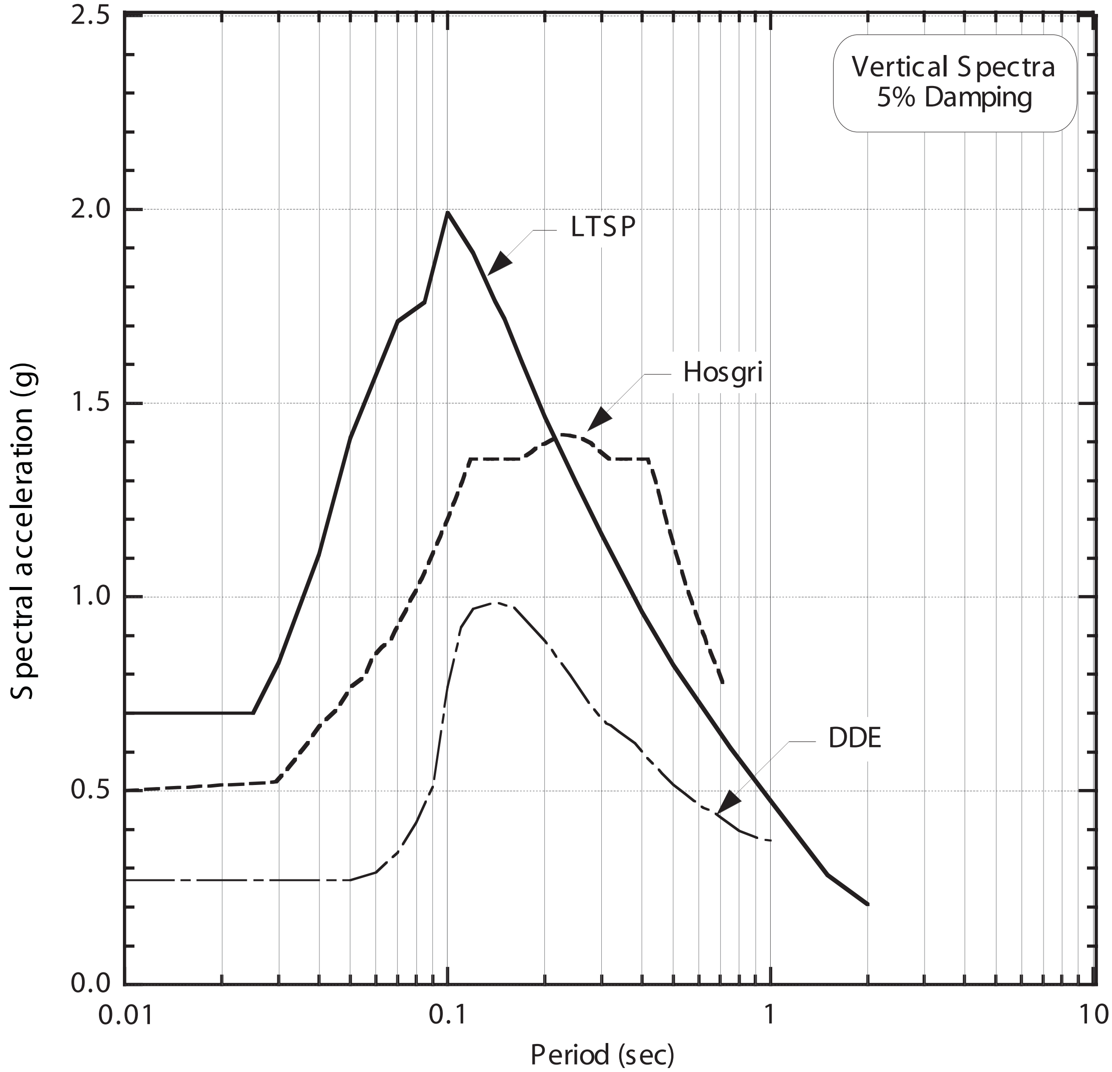


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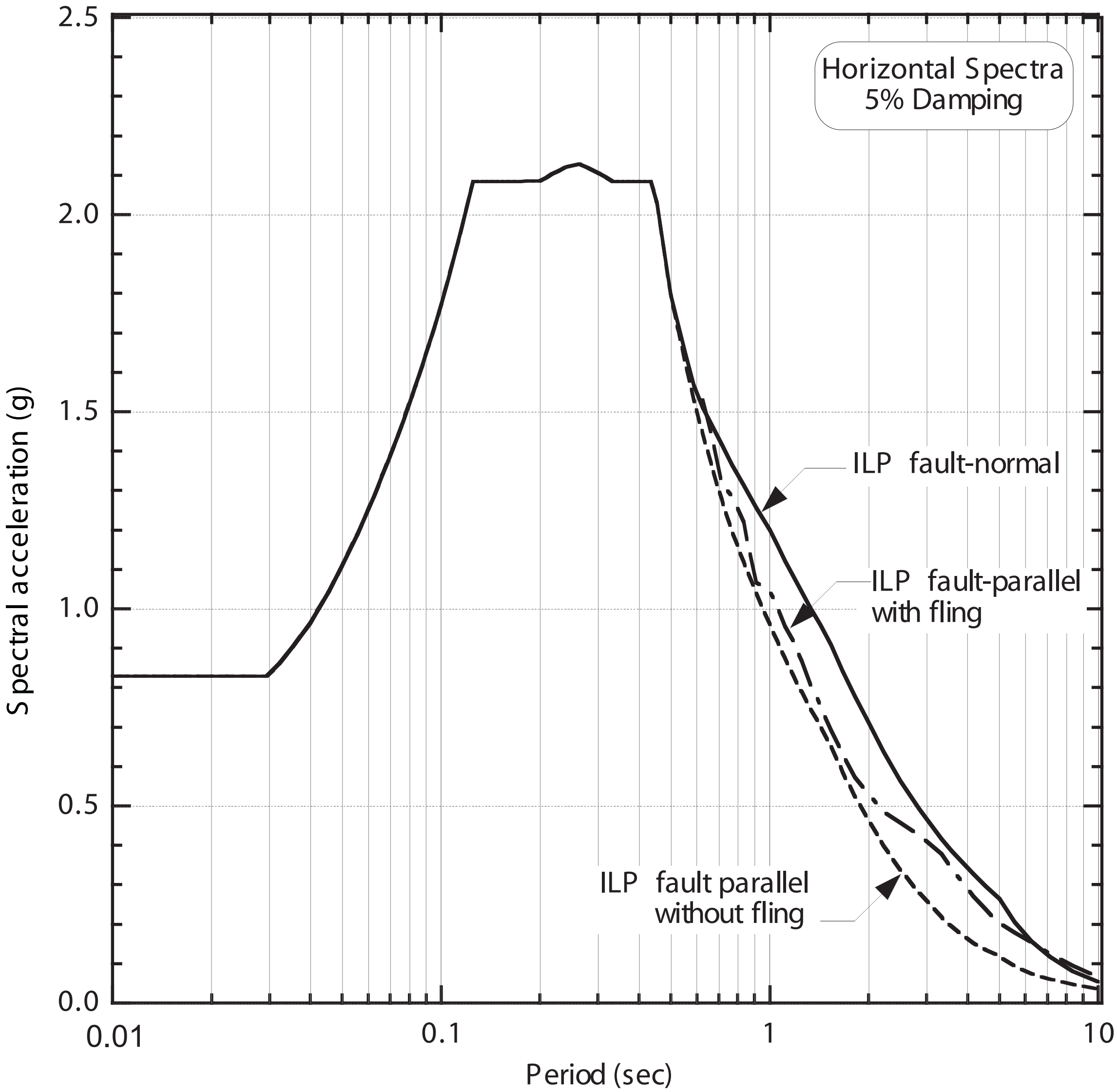
FIGURE 2.6-44 DDE, HOSGRI, AND LTSP VERTICAL SPECTRA



FSAR UPDATE

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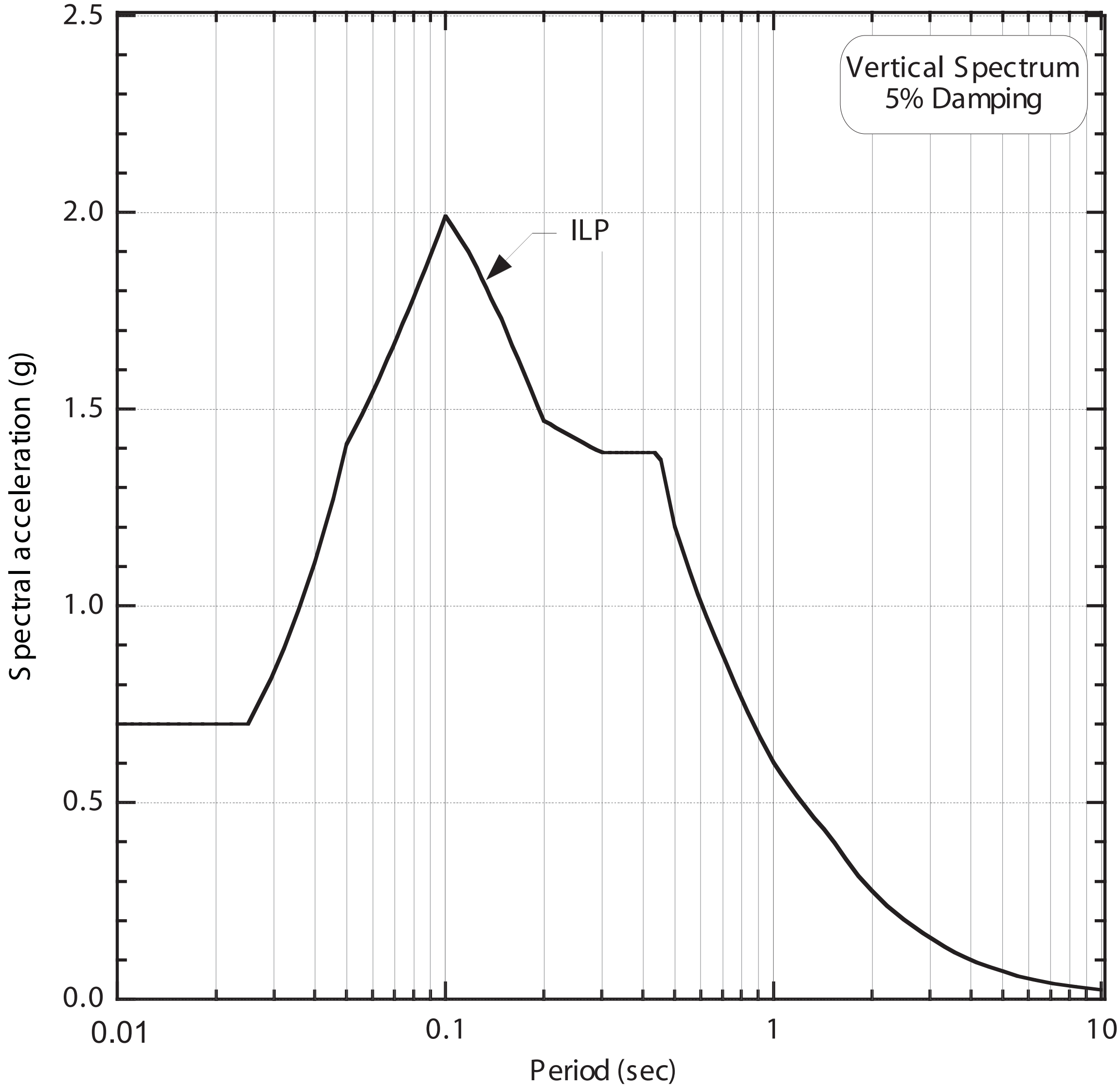
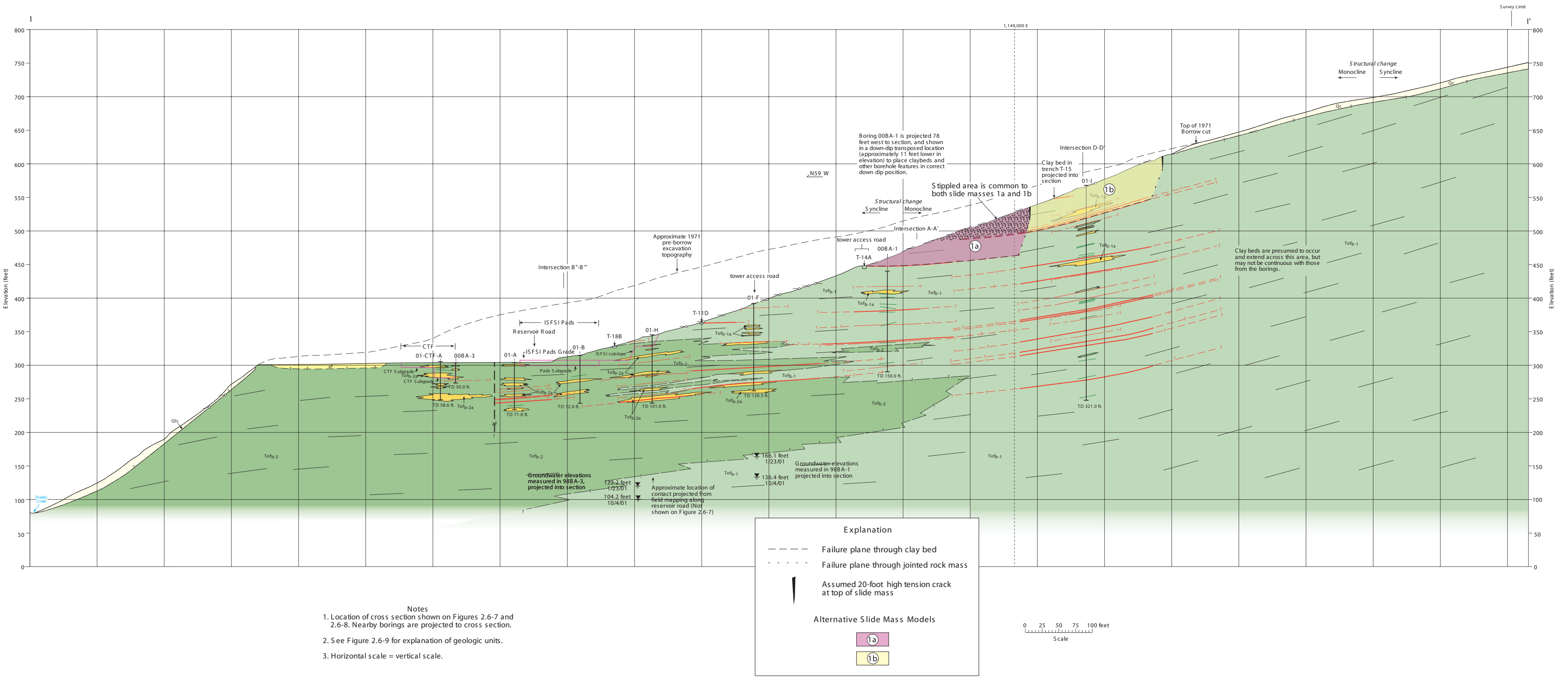
FIGURE 2.6-45
ILP HORIZONTAL SPECTRA



FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-46 ILP VERTICAL SPECTRUM



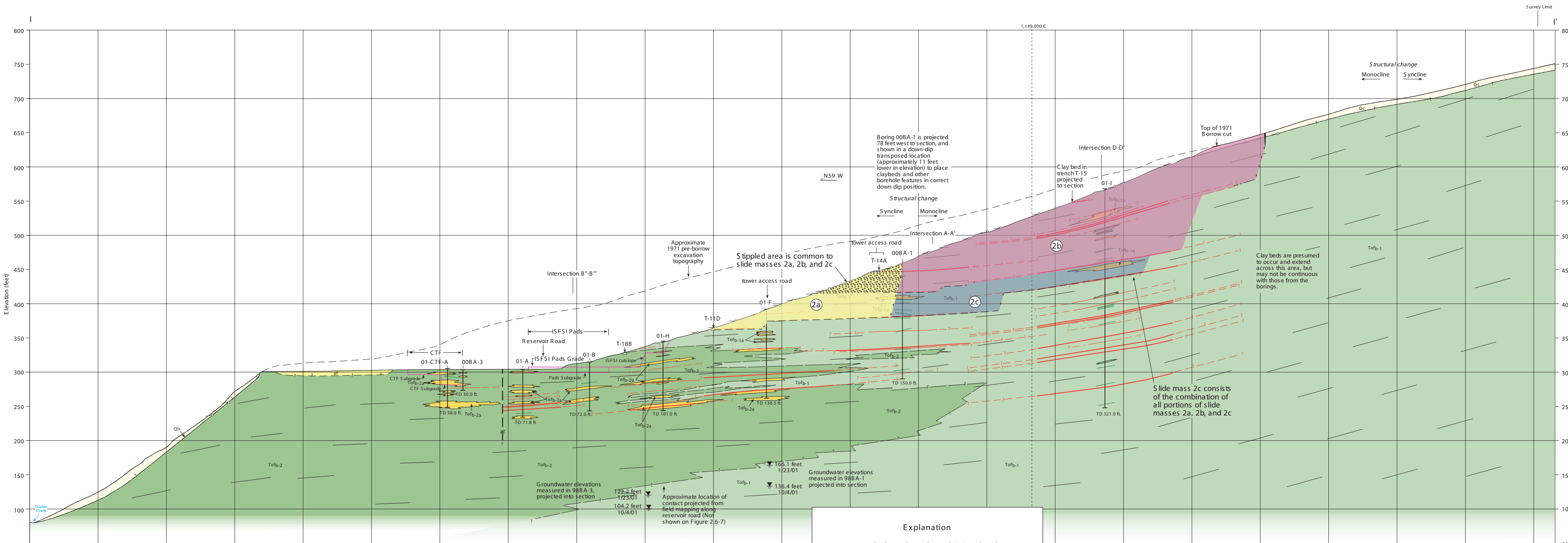
Notes
1. Location of cross section shown on Figures 2.6-7 and 2.6-8. Nearby borings are projected to cross section.
2. See Figure 2.6-9 for explanation of geologic units.
3. Horizontal scale = vertical scale.

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.6-47
SLIDE MASS MODEL 1

Revision 0 June 2004



Explanation

- Failure plane through jointed rock mass
- Failure plane through clay bed
- Assumed 20-foot high tension crack at top of slide mass

Alternative Slide Mass Models

- 2a
- 2b
- 2c

Notes

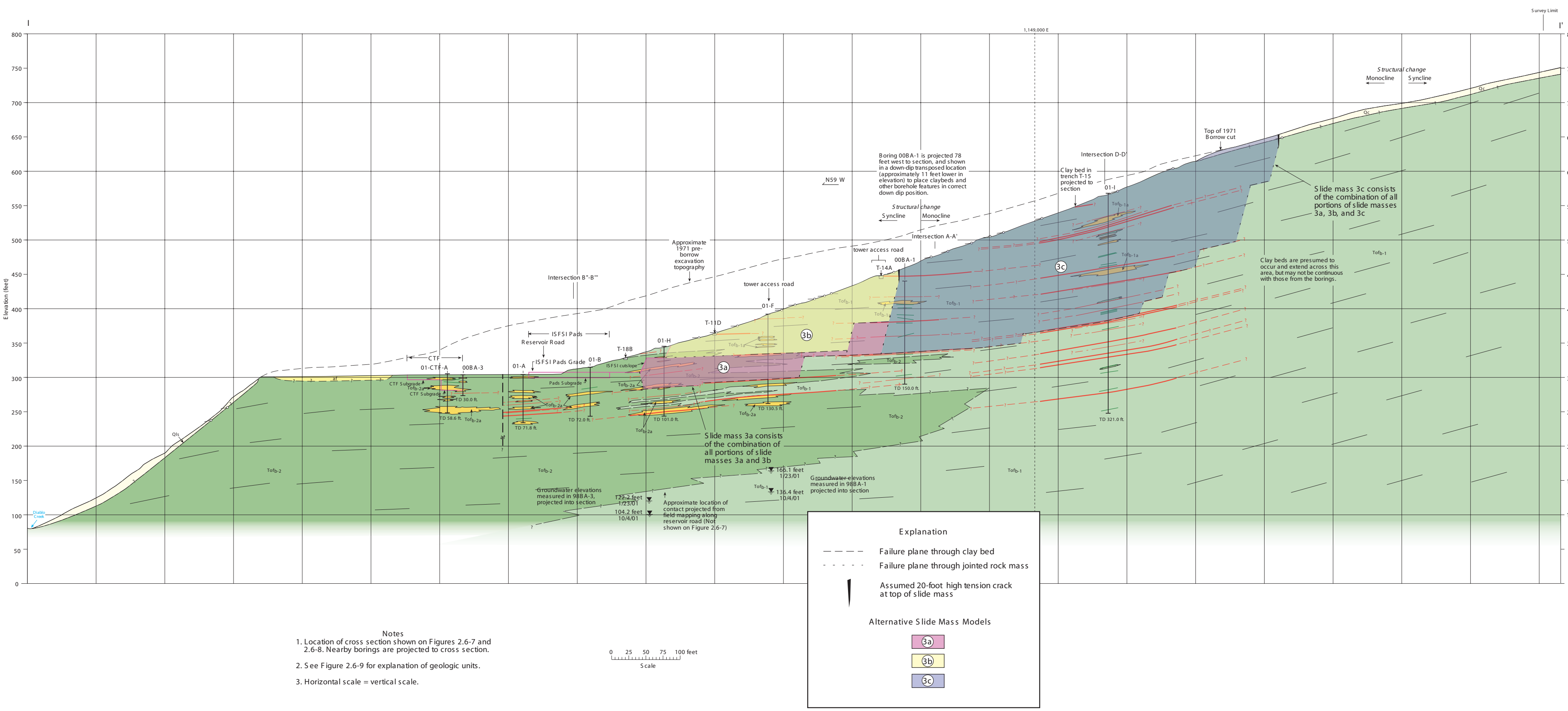
1. Location of cross section shown on Figures 2.6-7 and 2.6-8. Nearby borings are projected to cross section.
2. See Figure 2.6-9 for explanation of geologic units.
3. Horizontal scale = vertical scale.

0 25 50 75 100 feet
Scale

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.6-48
SLIDE MASS MODEL 2



Notes

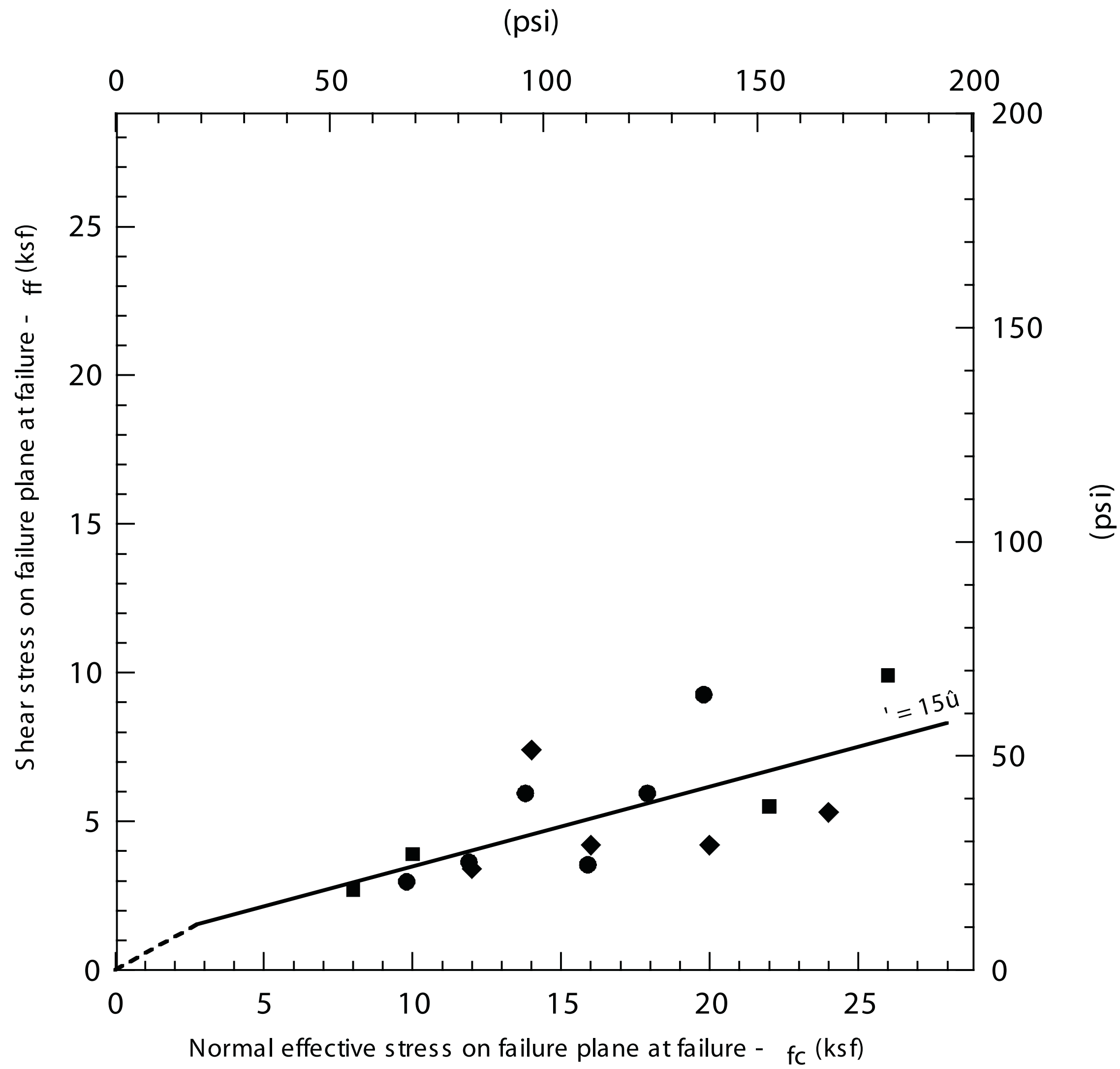
1. Location of cross section shown on Figures 2.6-7 and 2.6-8. Nearby borings are projected to cross section.
2. See Figure 2.6-9 for explanation of geologic units.
3. Horizontal scale = vertical scale.

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.6-49
SLIDE MASS MODEL 3

Revision 0 June 2004



EXPLANATION

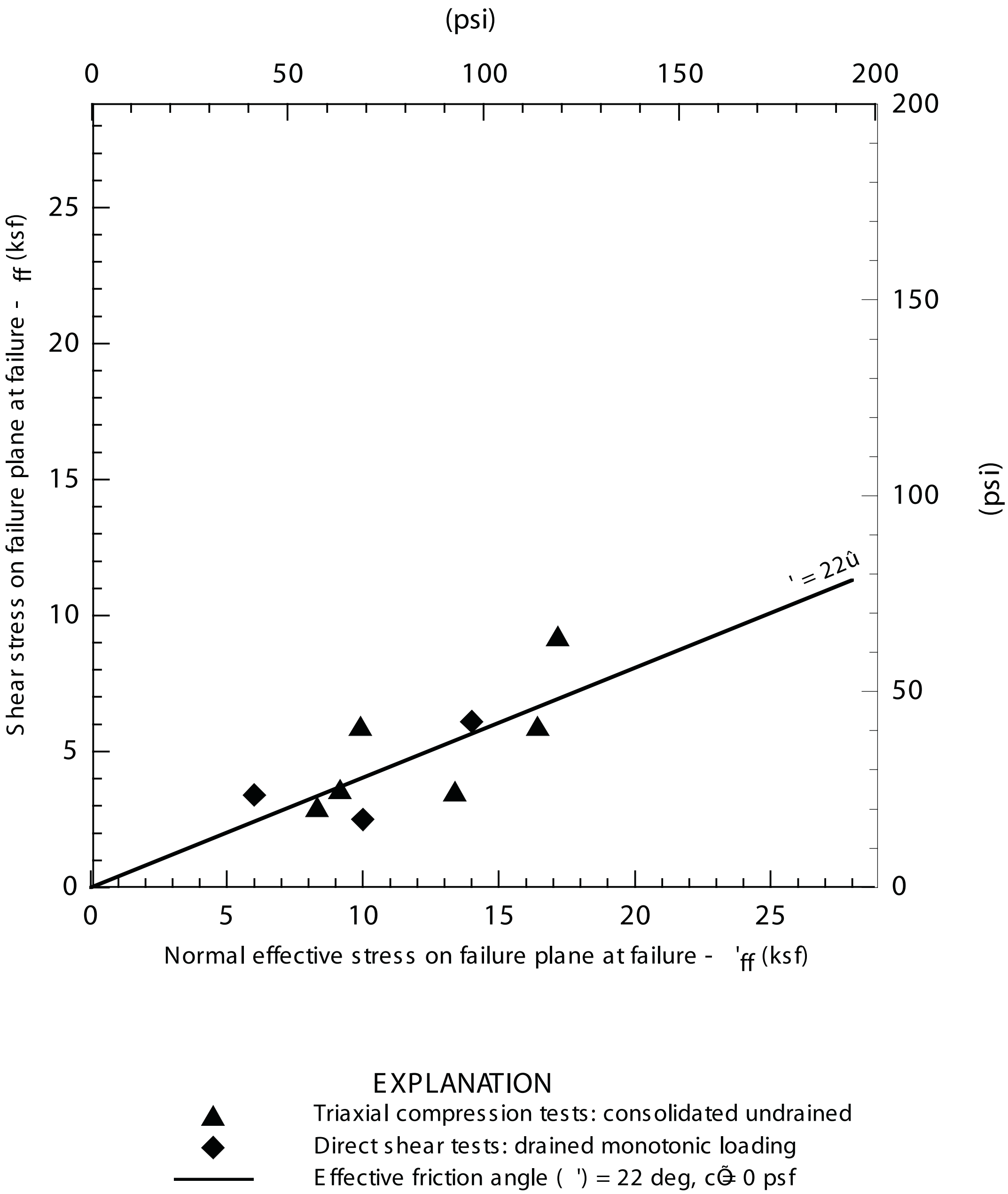
- ● Triaxial compression tests: consolidated undrained
- ◆ Direct shear tests: monotonic loading
- ■ Direct shear tests: cyclic loading
- - - - - Undrained shear strength envelope ff = fc ¥ tan(29û)
- ——— Undrained shear strength envelope ff = 0.8 ksf + fc ¥ tan(15û)

Data from William Lettis & Associates, 2001, Diablo Canyon ISFSI Data Report G, Soil Laboratory Test Data, Cooper Testing Laboratory

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.6-50
DESIGN UNDRAINED STRENGTH OF CLAY BEDS

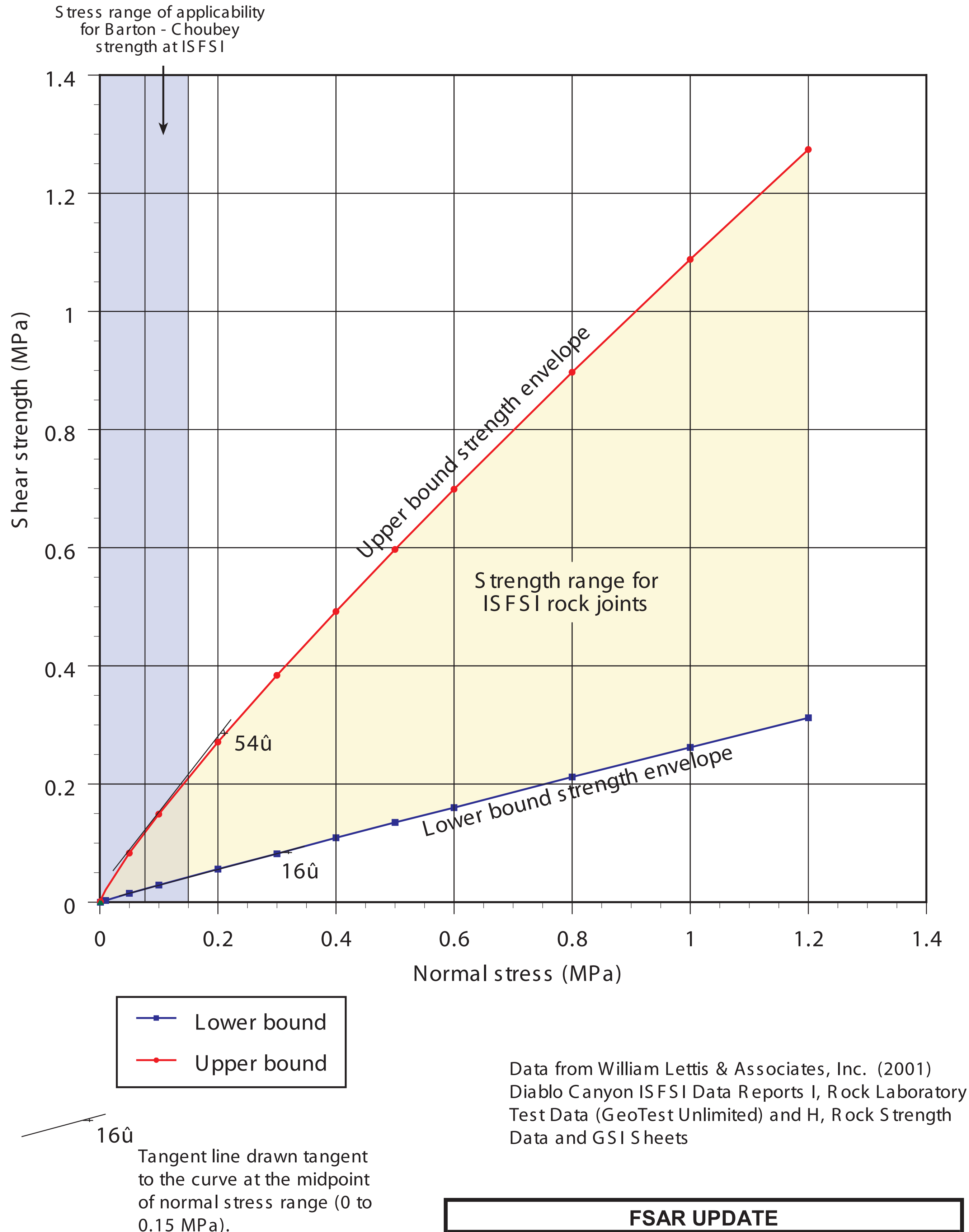


Data from William Lettis & Associates, 2001, Diablo Canyon ISFSI Data Report G, Soil Laboratory Test Data, Cooper Testing Laboratory

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.6-51
DESIGN DRAINED STRENGTH OF CLAY BEDS

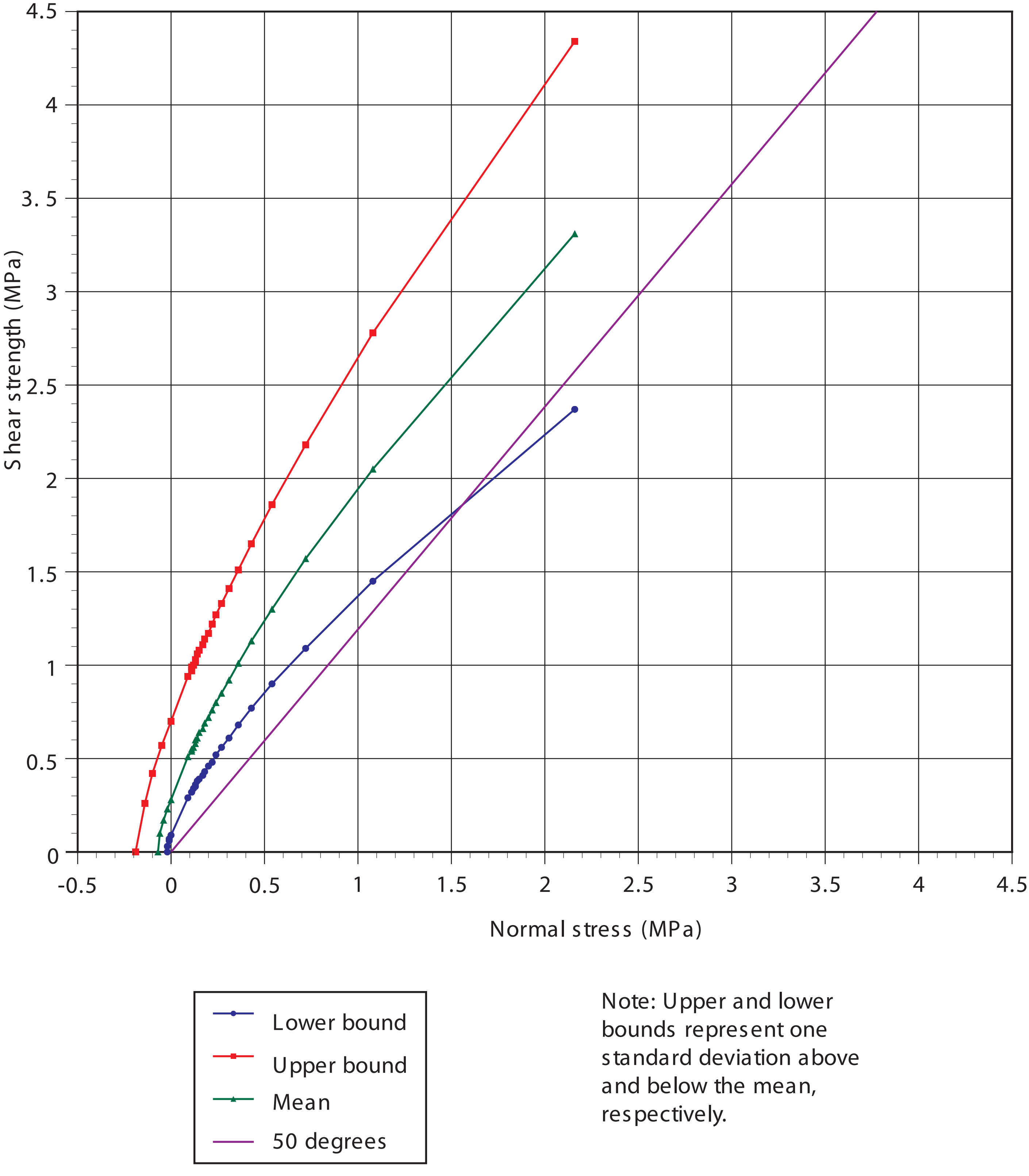


Data from William Lettis & Associates, Inc. (2001)
Diablo Canyon ISFSI Data Reports I, Rock Laboratory
Test Data (GeoTest Unlimited) and H, Rock Strength
Data and GSI Sheets

FSAR UPDATE

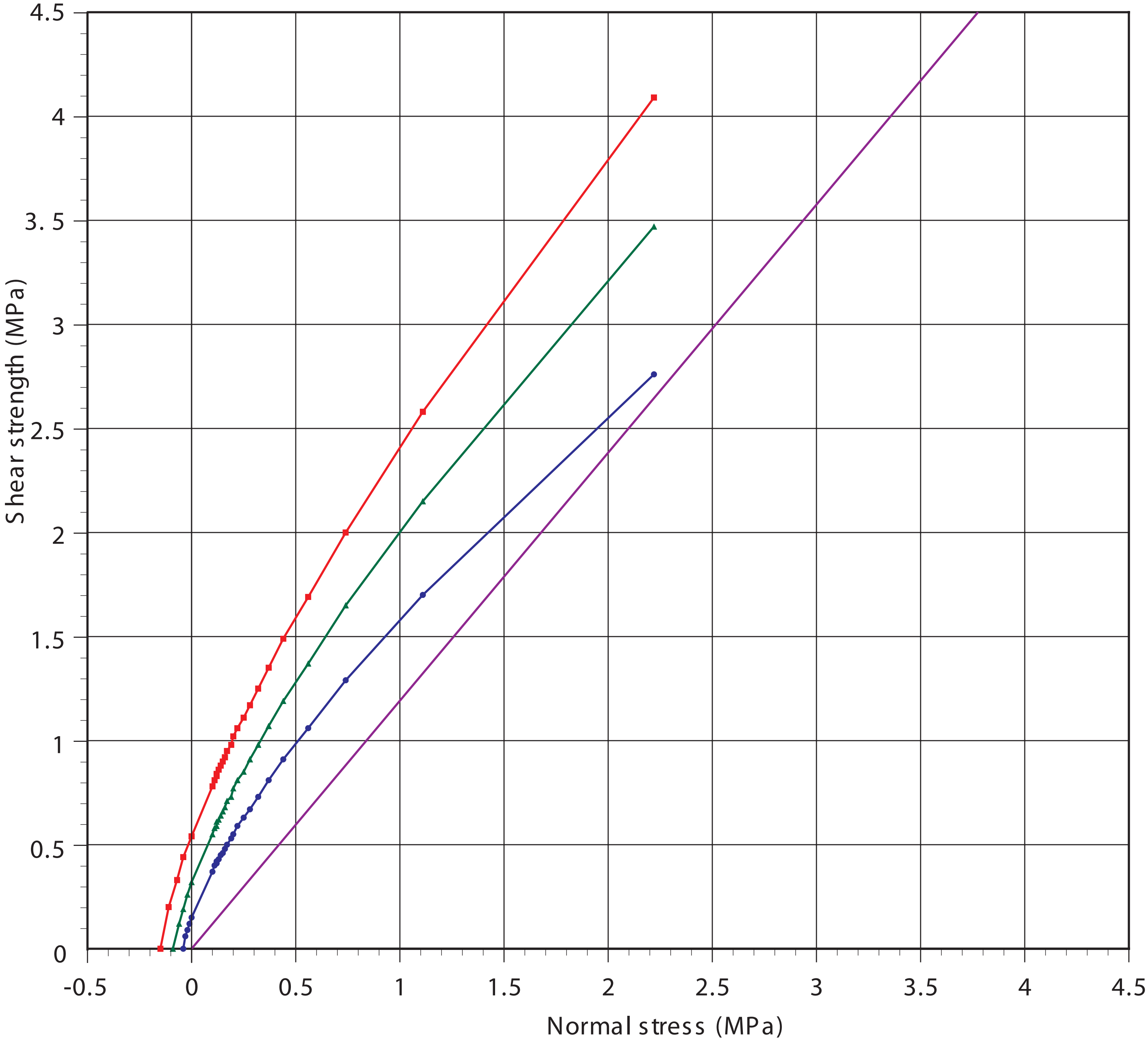
DIABLO CANYON ISFSI

FIGURE 2.6-52
RANGE OF SHEAR STRENGTHS FOR IN SITU DOLOMITE AND SANDSTONE ROCK JOINTS USING THE BARTON-CHOUBEY METHOD



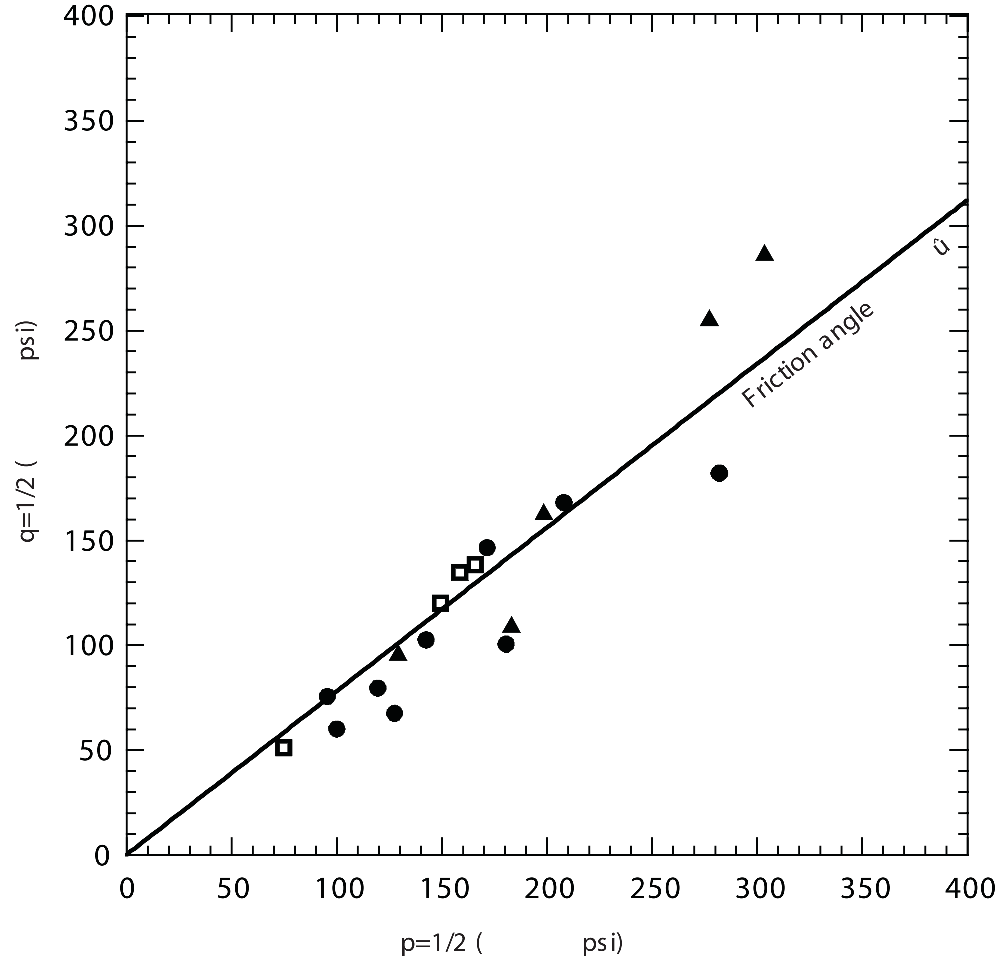
Note: Upper and lower bounds represent one standard deviation above and below the mean, respectively.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-53 COMPARISON OF HOEK-BROWN ENVELOPE FOR DOLOMITE WITH DESIGN STRENGTH OF 50 DEGREES



Note: Upper and lower bounds represent one standard deviation above and below the mean, respectively.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-54 COMPARISON OF HOEK-BROWN ENVELOPE FOR SANDSTONE WITH DESIGN STRENGTH OF 50 DEGREES

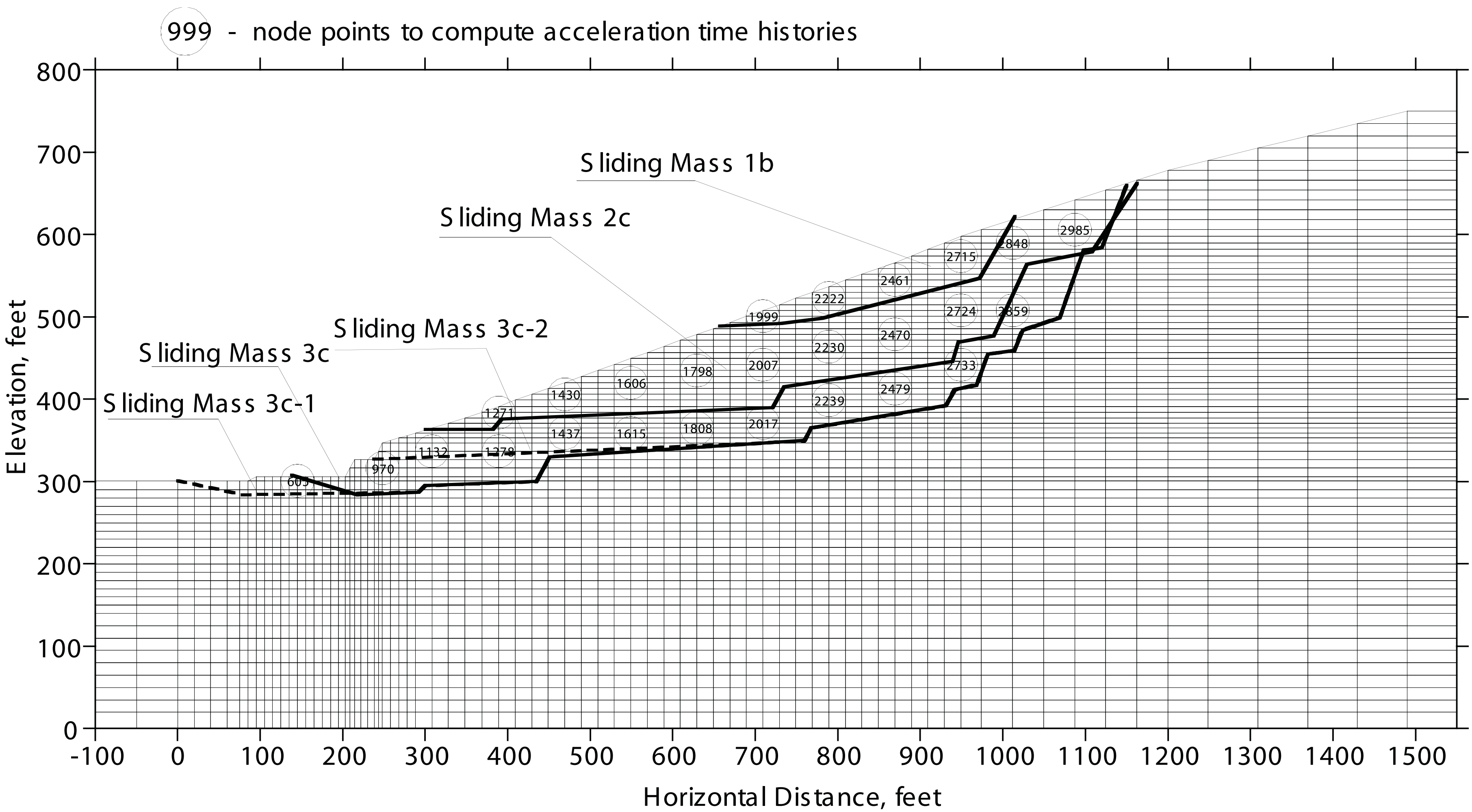


EXPLANATION

- ● Multi-stage triaxial tests with pore-water pressure measurements (pad + slope)
- ▲ Multi-stage triaxial tests without pore-water presure measurements (pad + slope)
- ◻ Multi-stage triaxial tests without pore-water pressure measurements from boring (00BA-2)

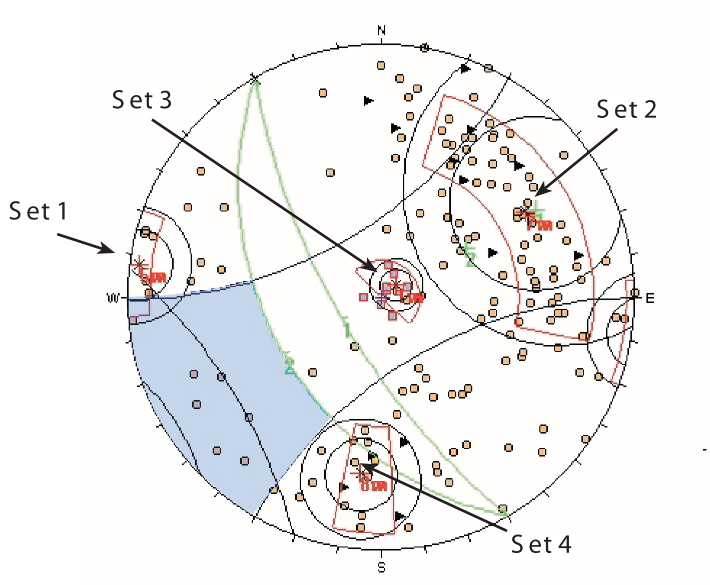
Data from William Lettis & Associates, 2001, Diablo Canyon ISFSI Data Report G, Soil Laboratory Test Data, Cooper Testing Laboratory

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-55 TOTAL STRENGTH ANALYSIS OF FRIABLE SANDSTONE BASED ON TRIAXIAL TESTS



FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-56 POTENTIAL SLIDING MASSES AND NODE POINTS FOR COMPUTED ACCELERATION TIME HISTORIES

Revision 0 June 2004



A. Topple hazard (low hazard)

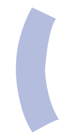
Explanation

Poles

Bedding

Fault

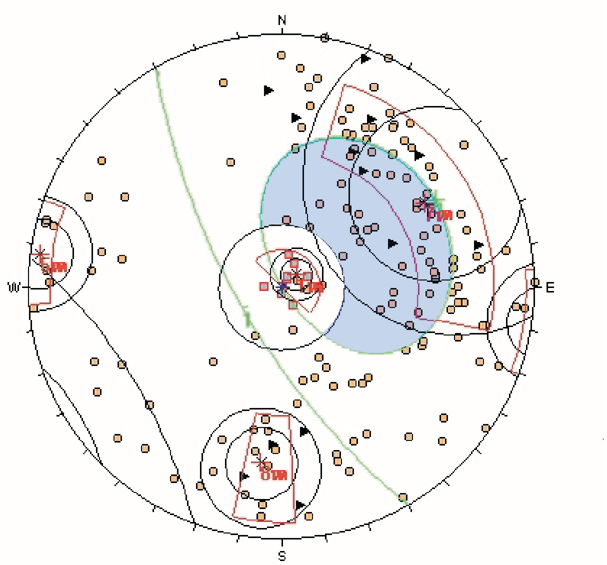
Joint



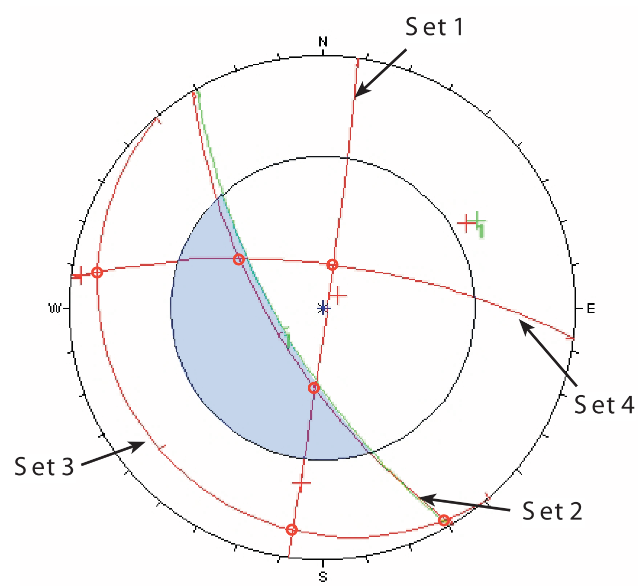
Failure envelope (based on 28 friction angle)

Failure envelope for topple and planar sliding without poles indicates stable conditions.

Failure envelope for wedge sliding without great circle intersections indicates stable conditions.



B. Planar sliding hazard (moderate to high hazard)

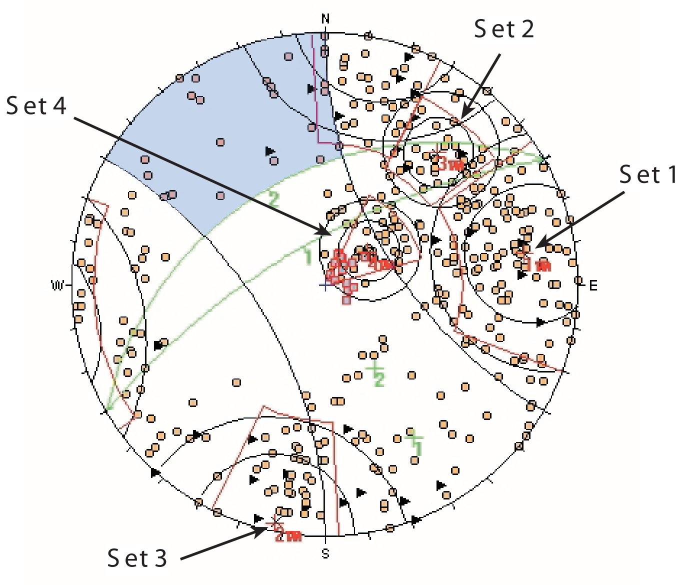


C. Wedge sliding hazard (moderate to high hazard)

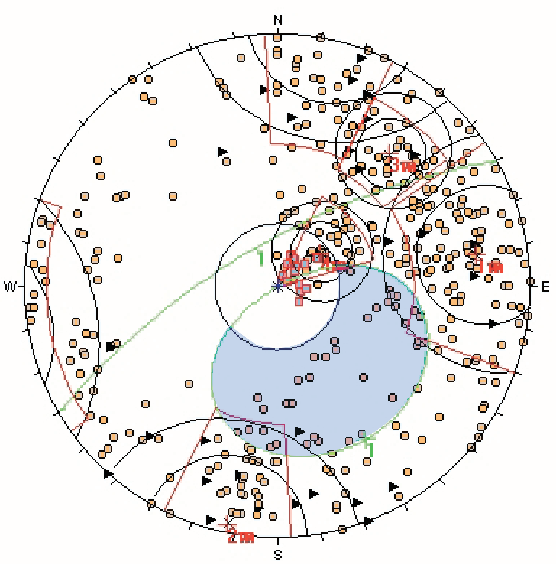
Notes

Analysis performed using computer program DIPS (Rocscience, 1999, DIPS: Plotting analysis, and presentation of structural data using spherical projection techniques, version 5.041, Toronto, 86p).

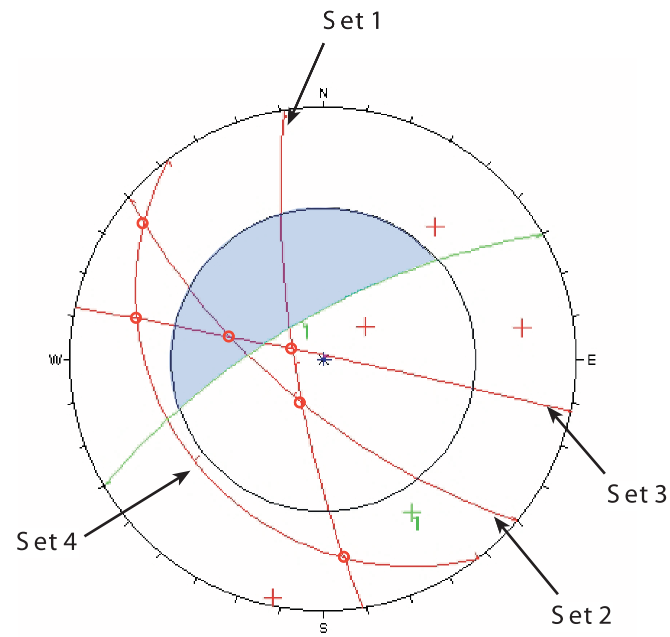
FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-57 KINEMATIC ANALYSES OF EAST CUTSLOPE



A. Topple hazard (low hazard)



B. Planar sliding hazard (low to moderate hazard)



C. Wedge sliding hazard (high hazard)

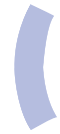
Explanation

Poles

Bedding

Fault

Joint



Failure envelope (based on 28 friction angle)

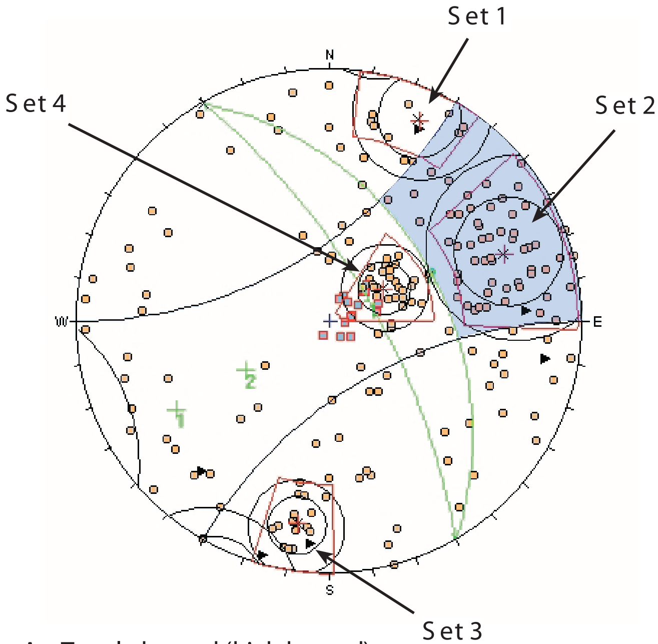
Failure envelope for topple and planar sliding without poles indicates stable conditions.

Failure envelope for wedge sliding without great circle intersections indicates stable conditions.

Notes

Analysis performed using computer program DIPS (Rocscience, 1999, DIPS: Plotting analysis, and presentation of structural data using spherical projection techniques, version 5.041, Toronto, 86p).

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-58 KINEMATIC ANALYSES OF BACK CUTSLOPE



A. Topple hazard (high hazard)

Explanation

Poles

Bedding

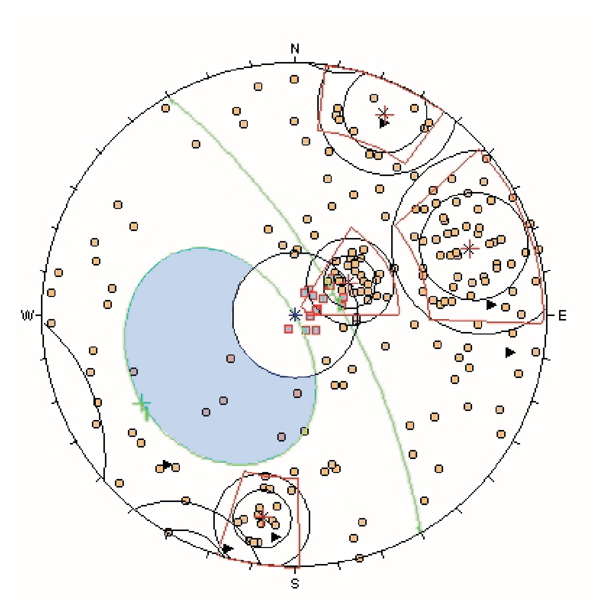
Fault

Joint

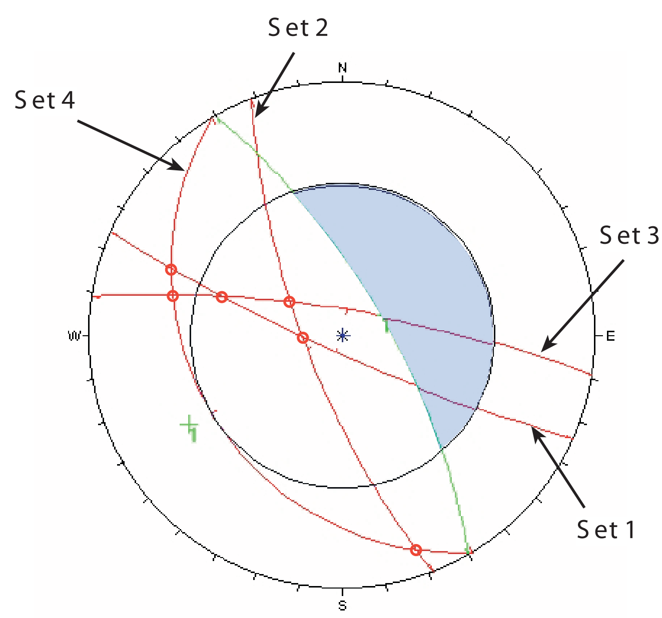
Failure envelope
(based on 28
friction angle)

Failure envelope for topple and planar sliding without poles indicates stable conditions.

Failure envelope for wedge sliding without great circle intersections indicates stable conditions.



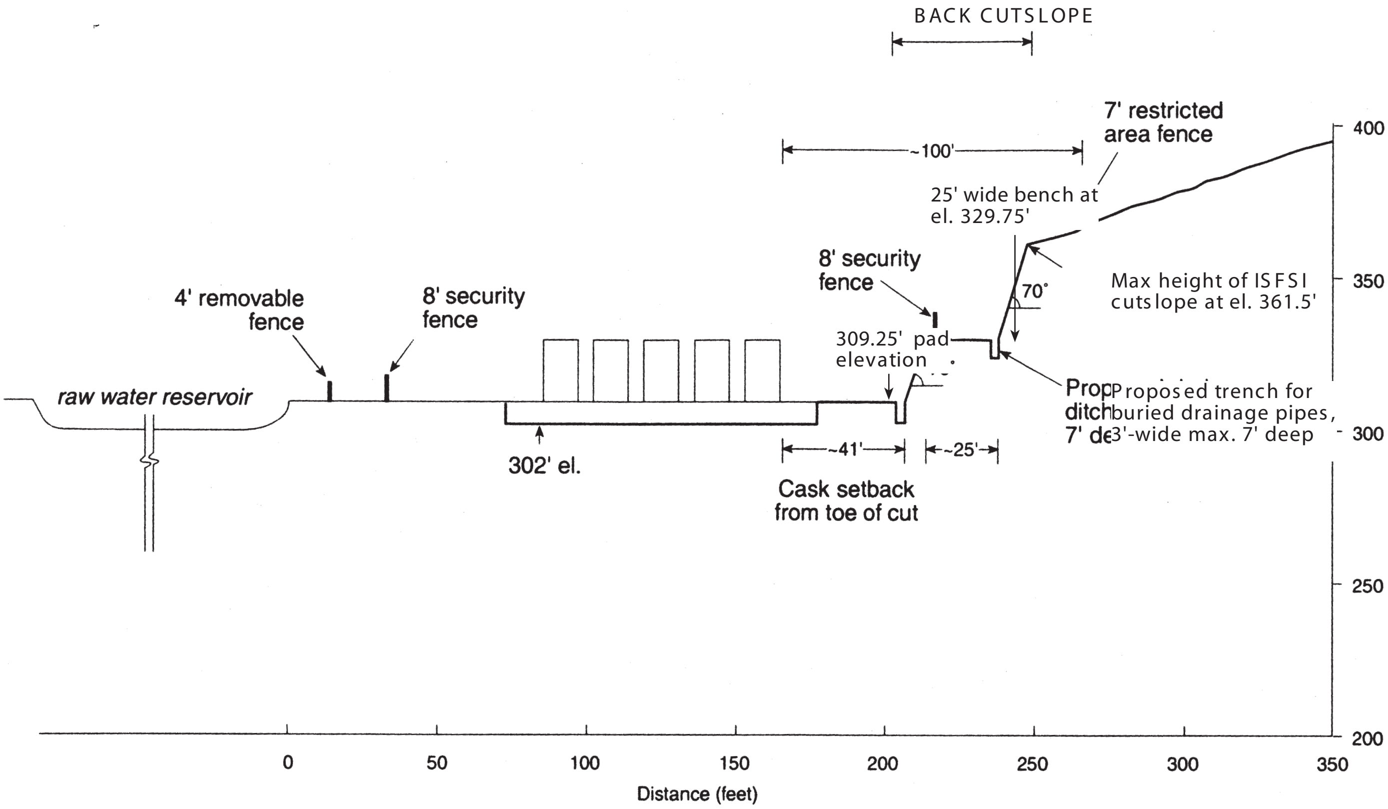
B. Planar sliding hazard (low hazard)



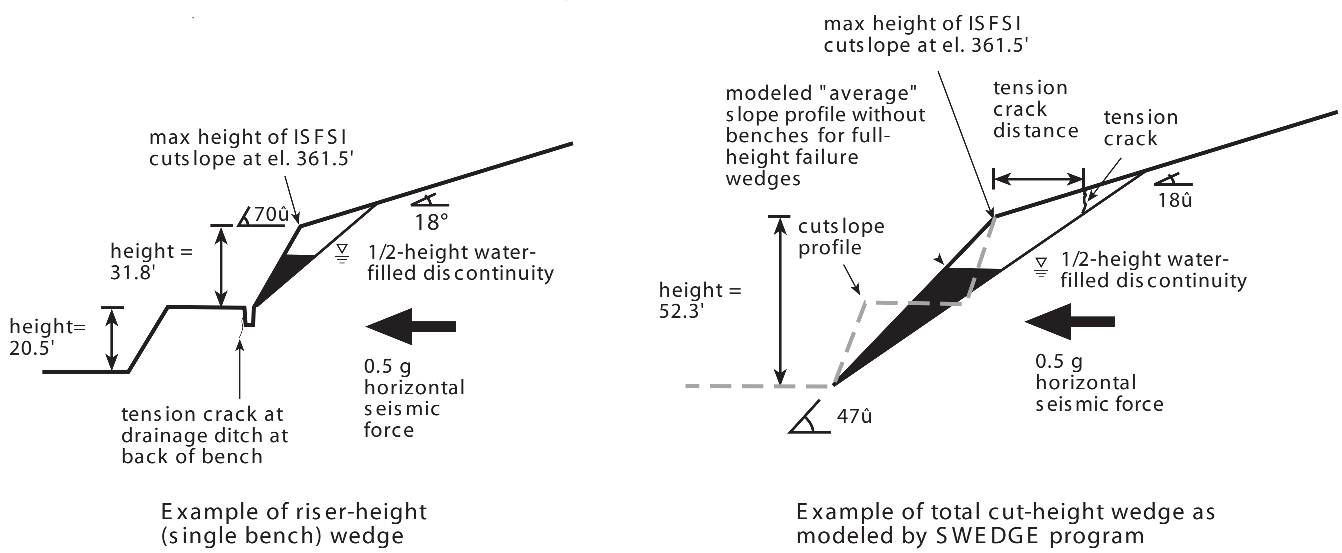
C. Wedge sliding hazard (very low hazard)

Notes
Analysis performed using computer program DIPS (Rocscience, 1999, DIPS: Plotting analysis, and presentation of structural data using spherical projection techniques, version 5.041, Toronto, 86p).

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-59 KINEMATIC ANALYSES OF WEST CUTSLOPE



A. Cross section through ISFSI pad and back cut, looking east

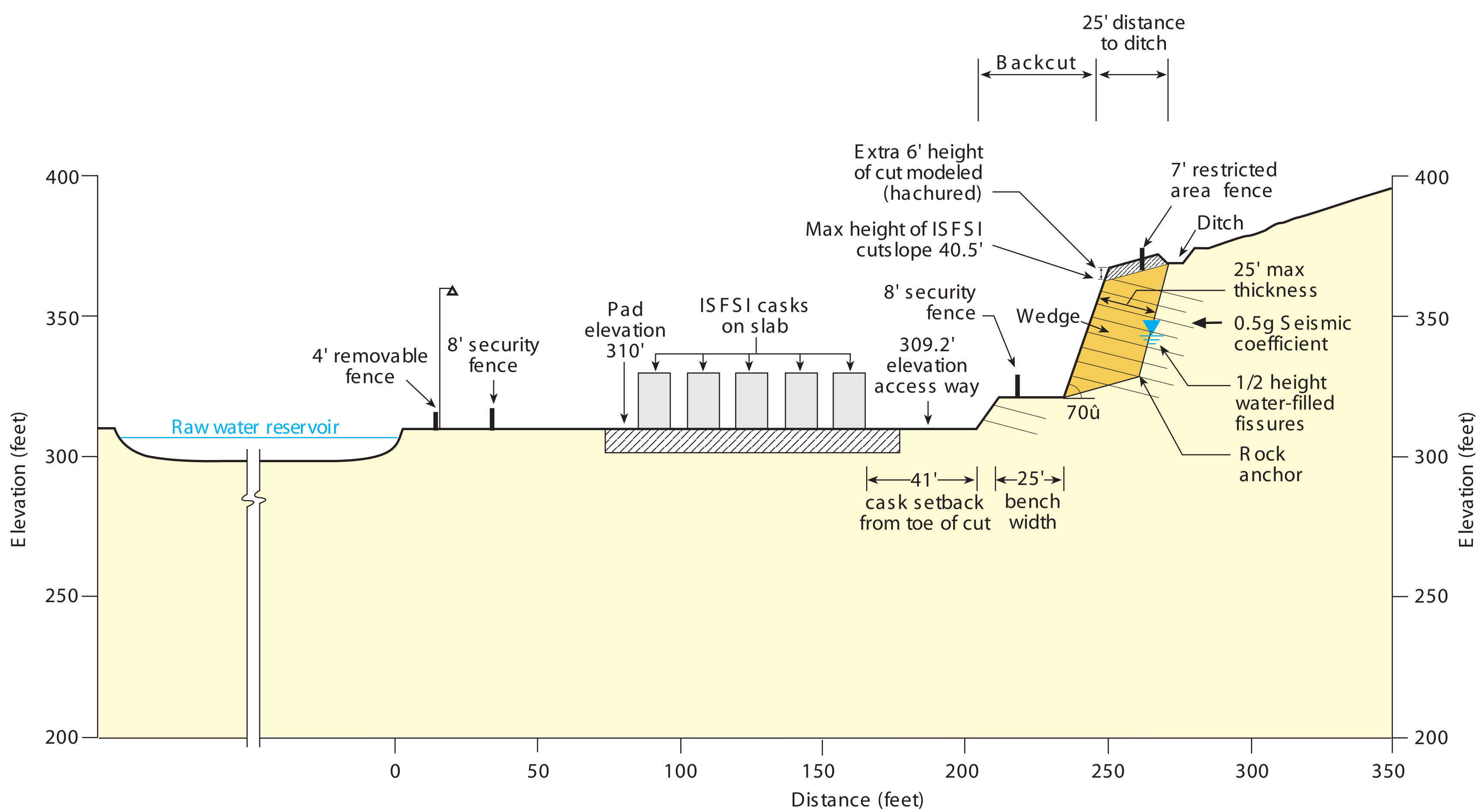


B. SWEDGE analysis cut configurations

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 2.6-60
CUTSLOPE CONFIGURATION USED IN
SWEDGE ANALYSES



Wedge, loading conditions, and example pattern rock anchors.

Note: cutslope geometry is based on PG&E Cannon Associates drawings, numbers CE41240EX1/EX2, 1/26/05.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 2.6-61 REVISED CUTSLOPE CONFIGURATION USED IN SWEDGE ANALYSES

CHAPTER 3

PRINCIPAL DESIGN CRITERIA

CONTENTS

CHAPTER 3

PRINCIPAL DESIGN CRITERIA

TABLES

CHAPTER 3

PRINCIPAL DESIGN CRITERIA

FIGURES

CHAPTER 3

PRINCIPAL DESIGN CRITERIA

This chapter describes the design bases and criteria for the Diablo Canyon ISFSI. Section 3.1 provides the purposes of the installation, while Sections 3.2 through 3.4 provide the design criteria for the ISFSI structures, systems, and components (SSCs) classified as important to safety. These SSCs include the multi-purpose canister (MPC), the storage overpack, the HI-TRAC transfer cask, the storage pads, the cask transporter, and the cask transfer facility (CTF). Section 3.2 provides the design criteria for environmental conditions and natural phenomena, while Section 3.3 provides the other design criteria for these SSCs. Section 3.4 summarizes the principal design criteria. Chapter 4 provides the descriptive design information for these SSCs with emphasis on those design features that are important to safety, are covered by the quality assurance program, and are employed to withstand environmental and accident forces. Appendix A is a discussion of conformance with NRC Interim Staff Guidance 15 (ISG-15) dated January 10, 2001, on dry cask storage materials.

3.1 PURPOSES OF INSTALLATION

The Diablo Canyon ISFSI is designed for interim, dry, and above-ground storage of intact and damaged spent nuclear fuel assemblies, fuel debris, and nonfuel hardware from DCPP Units 1 and 2. The ISFSI will use the Holtec International HI-STORM 100 System storage system, as discussed in Section 1.1. Installation requirements for the storage pad and its embedments, the cask transfer facility, and the lateral restraints for the cask transporter can be found in Sections 3.3.2.3, 3.3.4.2.4, and 3.3.4.2.5, respectively.

The material from the DCPP spent fuel pool (SFP) will be sealed in MPCs, transported to the CTF in the transfer cask, the MPC transferred to the HI-STORM 100SA overpack, and stored in HI-STORM 100SA ventilated storage overpacks arranged on and anchored to a reinforced concrete pad. The stand-alone ISFSI will allow additional spent fuel to be stored in the SFP allowing for continued operation of DCPP. The Diablo Canyon ISFSI is designed to ultimately store up to 4,400 spent fuel assemblies or up to 138 casks, with 2 spare locations.

The MPC-32 will be used to store up to 32 intact fuel assemblies that meet the approved content requirements of the Diablo Canyon ISFSI Technical Specifications (TS). The MPC-24, MPC-24E, and MPC-24EF were originally licensed to store up to 24 fuel assemblies that meet the approved content requirements of the Diablo Canyon ISFSI TS, including limited storage of damaged fuel assemblies and fuel debris. The originally-licensed MPC-24s will require modifications and analyses similar to the MPC-32 prior to their use (refer to Section 1.1).

3.1.1 MATERIAL TO BE STORED

The materials to be stored at the ISFSI consist of intact fuel assemblies, damaged fuel assemblies, fuel debris, and nonfuel hardware. Each fuel assembly contains approximately 1,100 pounds (500 kg) of UO_2. Nonfuel hardware may be stored within fuel assemblies and consists of borosilicate absorber rods, wet annular burnable absorber rods (WABAs), thimble plug devices (TPDs), neutron source assemblies (NSAs), instrument tube tie rods (ITTRs), and rod cluster control assemblies (RCCAs). Discussed herein are the characteristics of these materials and how the HI-STORM 100 System storage system design criteria envelopes these characteristics.

While the fuel rod cladding is a confinement barrier, credit is not taken for it in the design of the MPC or in the Diablo Canyon ISFSI Technical Specifications (TS).

3.1.1.1 Physical Characteristics

The spent fuel assemblies to be stored currently consist of both Westinghouse LOPAR and VANTAGE 5 assemblies. Both types are configured in a 17-by-17 array and the fuel rods consist of UO_2 pellets encapsulated in zirconium alloy tubing that is plugged and seal-welded at the ends. The VANTAGE 5 fuel rods have the same cladding wall thickness as the LOPAR fuel rods, but the fuel rod diameter is reduced to optimize the water-to-uranium ratio. Details of the physical characteristics of the DCPP fuel to be stored are provided in Section 4.2.1.2 and Table 4.1-1 of the DCPP FSAR Update (Reference 1) and are summarized in Table 3.1-1. Also provided in Table 3.1-1 are limiting values from the Holtec Certificate of Compliance (CoC) No. 1014, Amendment 1 (Reference 2). The LOPAR and VANTAGE 5 fuel assemblies (including VANTAGE 5 using Zirlo, sometimes referred to as VANTAGE+) are bounded by the 17x17B and 17x17A array/classes of fuel assemblies, respectively, as described in Holtec CoC No. 1014, Amendment 1. The LOPAR and VANTAGE 5 fuel currently covers all fuel loaded for operation at DCPP through Cycle 16. The Diablo Canyon ISFSI license was modified in LA 2 to allow alternate fuel types that meet the previously established fuel characteristics in anticipation of upcoming changes in fuel loading strategies. The Diablo Canyon ISFSI license was modified in LA 3 to allow loading of HBF with a maximum heat load of 28.74 kW.

The following fuel assembly physical characteristics constitute the most significant limiting parameters for storage of intact fuel assemblies at the Diablo Canyon ISFSI:

(1) Initial Fuel Enrichment

The maximum initial fuel enrichment of any fuel that is stored at the ISFSI is limited to 5 percent as required by the Diablo Canyon ISFSI TS and Section 10.2.

(2) Physical Configuration/Condition

Only fuel and associated nonfuel hardware irradiated at DCPP Units 1 and 2 with the physical configuration described in this section and Section 10.2 is stored in the Diablo Canyon ISFSI.

Fuel records will be maintained that identify the configuration and initial enrichment of each fuel assembly. Each fuel assembly and associated nonfuel hardware are engraved with a unique identification number. A verification of these numbers is made to ensure that only approved fuel and associated nonfuel hardware is loaded in MPCs in accordance with the Diablo Canyon ISFSI TS and Section 10.2.

3.1.1.2 Thermal and Radiological Characteristics

Details of the thermal and radiological characteristics of the DCPP fuel to be stored are provided in Table 3.1-2. The following fuel assembly thermal and radiological characteristics constitute the most significant limiting parameters for storage of fuel assemblies at the Diablo Canyon ISFSI.

(1) Heat Generation

The maximum heat generation rate for an assembly that is stored at the Diablo Canyon ISFSI is less than or equal to that specified in Section 10.2.

The heat generation rate of an individual fuel assembly is dependent on four factors: the initial fuel enrichment, the uranium mass, the fuel burnup, and the amount of cooling time. Fuel records are used to ensure that the heat generation per assembly is less than or equal to that specified in Section 10.2. Although not required, PG&E will conservatively apply a 5 percent burnup uncertainty allowance when calculating the decay heat for each loaded MPC.

(2) Fuel Burnup

The maximum average fuel burnup per assembly of any fuel that is stored at the ISFSI is limited to that specified in Section 10.2 and the Diablo Canyon ISFSI TS. The maximum allowed burnup is a function of the fuel cooling time.

A review of Materials License SNM-2511 and its associated Safety Evaluation Report; PG&E Letter DIL-04-002; and ISG-11, Revision 3, shows there is no regulatory requirement to include burnup uncertainty when evaluating compliance with TS burnup limits. Therefore, burnup uncertainty will not be applied to calculated fuel assembly burnup values when evaluating the eligibility of fuel assemblies for storage at the Diablo Canyon ISFSI.

(3) Cooling Time

The cooling time of any fuel that is stored at the ISFSI is greater than or equal to 5 years as specified in Section 10.2. The minimum required cooling time is a function of the fuel burnup and decay heat.

(4) Surface Dose Rates

The transfer cask and overpack surface dose rates from the fuel assemblies stored in an individual HI-STORM 100SA overpack at the Diablo Canyon ISFSI are dependent on the initial fuel enrichment, uranium mass, burnup, cooling time, and the presence of nonfuel hardware.

Fuel records are used to verify that, prior to loading, all fuel parameters are in compliance with Section 10.2.

3.1.1.3 Nonfuel Hardware

Nonfuel hardware, consisting of borosilicate absorber rods, wet annular burnable absorber rods, thimble plug devices, neutron source assemblies, instrument tube tie rods, and rod cluster control assemblies may be stored integral with the spent fuel assemblies. The nonfuel hardware type, burnup, and cooling time will be limited to that specified in Section 10.2.

3.1.2 GENERAL OPERATING FUNCTIONS

The overall operation of the HI-STORM 100 system is summarized in Chapter 1 of the HI-STORM 100 System FSAR. The following major operational sequences include:

- Moving the transfer cask containing the empty MPC into the SFP
- Loading of spent fuel assemblies into the MPC in the SFP
- Removal of the loaded MPC and transfer cask from the SFP
- MPC closure welding and draining, drying, and helium backfill operations
- Transfer of the MPC from the transfer cask to the overpack at the CTF
- Movement of the loaded overpack to the ISFSI storage pad

The above operational sequences are discussed generically in the HI-STORM 100 System FSAR. PG&E will develop site-specific implementing procedures that meet the intent of the HI-STORM 100 System FSAR and consider site-specific needs and capabilities. An overview of HI-STORM 100 System loading operations at Diablo

Canyon is provided below. A more detailed discussion of operations is provided in Sections 4.4 and 5.1.

After the HI-STORM 100 System components are received onsite, inspected, and cleaned as necessary, they are prepared for movement into the DCPP fuel handling building/auxiliary building (FHB/AB). The transfer cask is moved into the Unit 2 cask washdown area where the MPC is installed. The transfer cask containing an empty MPC is then lifted and lowered into the SFP. DCPP spent fuel assemblies are loaded into the MPC in the SFP. After the completion of fuel loading and fuel assembly verification, the MPC lid is lowered into the MPC. The loaded transfer cask is lifted vertically out of the SFP and moved laterally to a point above the Unit 2 cask washdown area. The loaded transfer cask is lowered into the cask washdown area, decontaminated to the extent practicable, and prepared for welding operations.

The MPC lid is welded to the MPC shell. The transfer cask water jacket is filled with water to provide neutron shielding (this may occur before or after lid welding at the discretion of the DCPP radiation protection organization). The MPC is then drained of water, dried by forced helium dehydration, and backfilled with helium. If the MPC contains high-burnup fuel, the supplemental cooling system is placed in service. The vent and drain port cover plates are welded on and leak testing performed, and the MPC closure ring is welded on. The transfer cask lid is installed, and the loaded transfer cask is lifted and placed onto the low profile transporter (LPT).

At the CTF, the cask transporter positions the transfer cask above an empty overpack that has been previously placed in a below-grade vault at the CTF. The MPC is lowered from the transfer cask into the overpack and the transfer cask is removed from atop the overpack. The overpack top lid is installed and the cask transporter is used to lift the overpack out of the CTF and transport it to its designated storage location on the ISFSI storage pad, where it is anchored in place. Section 5.1 discusses the detailed operational steps involved in this process. Equipment required to be available to mitigate off-normal conditions such as a loss of transporter power or hydraulics are discussed in Chapter 8.

While in its storage configuration, no active components are needed to ensure safe storage of the spent fuel. Cooling is provided by natural convective flow of ambient air into the inlet air vents at the bottom of the overpack and out of the outlet vents at the top of the overpack. No utilities (that is, water, compressed air, electric power) are required to cool the spent fuel during storage. Adequate cooling air is assured through periodic surveillance of the overpack air duct inlet and outlet perforated plates (screens) at the ISFSI pad to verify that the air duct perforated plates (screens) are not blocked and are intact as required by the Diablo Canyon ISFSI TS.

3.1.3 REFERENCES

1. Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update.

2. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 1, July 15, 2002.

3. Deleted in Revision 2.

4. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 1A, January 2003.

5. Interim Staff Guidance Document 11 (ISG-11), Revision 3, Cladding Considerations for the Transportation and Storage of Spent Fuel, USNRC, November 17, 2003.

6. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 3, May 29, 2007.

3.2 DESIGN CRITERIA FOR ENVIRONMENTAL CONDITIONS AND NATURAL PHENOMENA

This section describes the design criteria for the Diablo Canyon ISFSI that are classified as important to safety and designed to withstand the effects of site-specific environmental conditions and natural phenomena. Regulatory requirements and guidance were drawn, as applicable, from 10 CFR 72 (Reference 1), Regulatory Guide 3.62 (Reference 2), the Standard Review Plan for ISFSIs (Reference 3), and the Standard Review Plan for Dry Cask Storage Systems (Reference 4). Diablo Canyon site-specific information for environmental conditions and natural phenomena was taken primarily from other parts of this FSAR and from the DCPP FSAR Update (Reference 5). Holtec storage system design information was taken primarily from the Holtec Certificate of Compliance (CoC) No. 1014, Amendment 1 (Reference 6), and the HI-STORM 100 System FSAR (Reference 7).

As discussed in Section 4.5, the ISFSI structures, systems, and components (SSCs) are classified as important-to-safety (ITS) or not important-to-safety (NITS) based on their design function. Among the SSCs classified as important to safety are the multi-purpose canisters (MPCs), the HI-STORM 100SA overpack, the storage pad, the HI-TRAC transfer cask, the onsite cask transporter, and the cask transfer facility (CTF). The ITS classification indicates that at least one subcomponent of the main component is classified as ITS. Other subcomponents may be classified as NITS, based on the function of the subcomponent. Design criteria for environmental conditions and natural phenomena for these entire key ISFSI SSCs are described in this section. Other design criteria for these key ISFSI SSCs are contained in Section 3.3.

Environmental conditions and natural phenomena specific to the Diablo Canyon ISFSI and DCPP sites are described and characterized in Chapters 2 and 3 and in the DCPP FSAR Update. The DCPP FSAR Update is maintained up to date by PG&E and is, for the most part, directly applicable to the Diablo Canyon ISFSI. Some natural phenomena are different for the ISFSI site than for the power plant site. For example, flooding is not a credible event at the ISFSI site because of drainage and elevation differences between the power plant and the ISFSI site. Such differences are appropriately considered in this and other parts of the ISFSI FSAR.

The storage system selected for the Diablo Canyon ISFSI, the HI-STORM 100 System, is designed to ensure that fuel criticality is prevented, fuel cladding and confinement integrity are maintained, the fuel remains retrievable, and reasonable radiation shielding is maintained under all Diablo Canyon site-specific design-basis loadings due to environmental conditions and natural phenomena.

The safe storage of the spent fuel assemblies depends upon the capability of the HI-STORM 100 System to perform its design functions. The HI-STORM 100 System is a self-contained, independent, passive system that does not rely on any other system for operation. At Diablo Canyon, the shortened and anchored version of the HI-STORM 100 System overpack, known as the HI-STORM 100SA, will be used. A description of the HI-STORM 100SA overpack is provided in Section 4.2.3. Stability under design-basis seismic loadings at the Diablo Canyon ISFSI is ensured by anchoring the HI-STORM 100SA overpack to the ISFSI pads, as described in Section 4.2.1. The overpack anchorage is the only required interface between the HI-STORM 100SA overpacks and other ISFSI components. Except for the anchorage details, all other overpack design features and functions are identical to the freestanding version of the system described in the HI-STORM 100 System FSAR. Therefore, the text of this section will refer to HI-STORM 100 System for simplicity.

The criteria used for the design of the HI-STORM 100 System were developed for generic certification of the HI-STORM 100 System under 10 CFR 72, Subpart L. The design criteria were chosen to bound the site-specific design criteria for most nuclear power plants in the United States, so that virtually any 10 CFR 50 licensee could use the HI-STORM 100 System at an onsite ISFSI under the general license provisions of 10 CFR 72. The principal design criteria for the HI-STORM 100 System meet all requirements of 10 CFR 72 and are described in Chapter 2 of the HI-STORM 100 System FSAR.

Environmental conditions and phenomena are summarized in this section for the important-to- safety SSCs, and include:

- Tornado and wind loadings, including tornado-borne missiles
- Water level (flood) design
- Seismic design
- Snow and ice loadings
- Combined load criteria
- Lightning
- Temperature and solar radiation.

The HI-STORM 100 System design features are evaluated in detail for fuel handling activities in the DCPP FHB/AB in 10 CFR 50 LAR 02-03 submitted to the NRC in April 2002 (Reference 22). The LAR describes MPC fuel loading in the spent fuel pools; draining, drying, sealing, helium filling, and helium leak testing the MPC while inside the HI-TRAC transfer cask; and loading the transfer cask onto the cask transporter for onsite transfer to the CTF. The NRC issued DCPP License Amendments 162 and 163 (Reference 23) to allow implementation of LAR 02-03.

3.2.1 TORNADO AND WIND LOADINGS

3.2.1.1 Applicable Design Parameters

As stated in Section 2.3.2, the highest recorded peak wind gust at the DCPP site was 84 mph. For storage system design purposes, a wind velocity of 80 mph is used (Section 3.3.1 of the DCPP FSAR Update) with a gust factor of 1.1, which envelopes the recorded, peak-gust value of 84 mph.

Tornado winds and outdoor tornado-borne missiles for the DCPP site are included in Section 3.3.2.1 of the DCPP FSAR Update. Specific wind speeds, pressure drops, and missile descriptions applicable to the operating configurations associated with the ISFSI site are presented in Tables 3.2-1 and 3.2-2. As shown in Table 3.2-1, the Diablo Canyon ISFSI tornado wind speeds are based on the DCPP FSAR Update and are bounded by those evaluated for licensing of the HI-STORM 100 System.

The HI-STORM 100 System, which includes all operating configurations applicable to the Diablo Canyon ISFSI, is generically designed to withstand pressures, wind loads, and missiles generated by a tornado as described in Section 2.2.3.5 of the HI-STORM 100 System FSAR. The design-basis tornado and wind loads for the HI-STORM 100 System are consistent with Regulatory Guide 1.76 (Reference 9), ANSI/ANS 57.9 (Reference 10), and ASCE 7-88 (Reference 11).

The tornado wind and missile evaluations for the DCPP ISFSI are based on the DCPP site licensing-basis wind speed of 200 mph shown in Table 3.2-1, and are considered representative of the ISFSI site. The tornado missiles evaluated for the Diablo Canyon ISFSI are listed in Table 3.2-2 and are a compilation of those from the DCPP FSAR Update; NUREG-0800, Section 3.5.1.4 (Reference 12) Spectrum II missiles; and three 500-kV tower missiles specific to the Diablo Canyon ISFSI site. Several of these missiles differ from those identified in the HI-STORM 100 System FSAR. The effects of these missiles are evaluated for Level D stress limits and cask penetration. The evaluation is consistent with the design criteria, as specified in NUREG-0800, Section 3.5.1.4, to withstand tornados in accordance with 10 CFR 72.120(a) and 72.122(b).

3.2.1.2 Determination of Forces on Structures

Tornado wind loads include consideration of the following, as applicable: (a) tornado wind load, (b) tornado differential pressure load, and (c) tornado missile impact load. The method of combining the applicable effective tornado wind, differential pressure, and missile impact loads to determine the total tornado load is done in accordance with NUREG-0800, Section 3.3.2.

3.2.1.3 Tornado Missiles

The HI-STORM 100 System, including the overpack and the transfer cask, is generically designed to withstand three types of tornado-generated missiles in accordance with NUREG-0800, Section 3.5.1.4, as noted in Table 3.2-2. The design basis for these missiles is discussed in Section 2.2.3.5 of the HI-STORM 100 System FSAR. The mass and velocity of these missiles, along with the design-basis tornado missiles for the Diablo Canyon ISFSI site are presented in Table 3.2-2. Table 3.2-2 also lists the DCPP licensing-basis tornado missiles. Due to the proximity of a 500-kV transmission tower to the ISFSI site, other missiles were evaluated as shown in Table 3.2-2. Missile evaluations are described in detail in Section 8.2.2 for cask transport from the FHB/AB, activities at the CTF, and at the ISFSI storage pad.

3.2.2 WATER LEVEL (FLOOD) DESIGN

The Diablo Canyon ISFSI pad is located at elevation +310 ft mean sea level (MSL). The Diablo Canyon ISFSI site surface hydrology is described in Section 2.4. It is concluded in Section 2.4 that there is no potential for flooding in the vicinity of the ISFSI. Therefore, flooding is not a consideration for ISFSI operations or on the capability of the dry storage cask system to safely store the spent fuel. Likewise, due to the elevation of the ISFSI site, tsunami is not a threat to the HI-STORM 100 Systems that are stored on the pad. Since the CTF is located adjacent to the ISFSI, these conclusions are also applicable for the potential flooding and tsunami impact on the CTF. A design-basis flooding event occurring during movement of the cask to or from the CTF along the transport route is not considered credible. Flooding of the overpack while it is located in the underground vault at the CTF is precluded by the use of a sump designed to remove any significant accumulation of water in the vault.

Therefore, while the HI-STORM 100 System is designed to withstand pressure and water forces associated with floods, such design features are unnecessary for the Diablo Canyon ISFSI and do not need to be evaluated. In conclusion, the ISFSI design is consistent with the design criteria of NUREG-0800 and ASCE 7-88 and can withstand floods as required by 10 CFR 72.120(a) and 72.122(b).

3.2.3 SEISMIC DESIGN

In accordance with 10 CFR 72.102(f)(1), the seismic design of the important-to-safety ISFSI SSCs, which include the HI-TRAC transfer cask, the HI-STORM 100SA overpack, the MPC, the CTF, the onsite cask transporter, LPT and ISFSI storage pads, is based on design-earthquake ground motions that have been established for the plant site. Site seismic characteristics and vibratory ground motion are discussed in Sections 2.6.1 and 2.6.2.

The ISFSI SSCs are designed to withstand seismic loads during: (a) onsite transport of the loaded transfer cask, (b) transfer operations at the CTF, (c) transport of loaded overpack to the storage pad, and (d) storage of the overpack on the ISFSI pad. The design bases for the ISFSI SSCs, including analyses and design procedures, are discussed in Sections 4.2, 4.3, 4.4.5, and 8.2.1. Seismic design for the loading and handling of the transfer cask while in the FHB/AB are addressed as part of the 10 CFR 50 LAR submitted to the NRC (Reference 22).

The HI-STORM 100SA is the short, anchored version of the HI-STORM 100S System. In contrast to a freestanding cask, the HI-STORM 100SA relies upon the anchorage hardware and its embedment into the ISFSI pad for resistance to overturning and sliding. The primary structural difference between the freestanding and anchored overpacks is the enlargement of the overpack base-plate diameter to accommodate a flange bolt circle, an upper ring, and a number of vertical gussets (Figure 4.2-7). Pretensioned anchor bolts are used to secure the overpack to an embedment in the pad. The ISFSI pads and associated embedments are an integral part of the seismic design of the cask system.

3.2.4 SNOW AND ICE LOADINGS

As noted in Section 2.3.2, essentially no snow or ice occurs at the ISFSI site. Therefore, even though the HI-STORM 100 System is designed to accommodate snow and ice loadings typical of the contiguous United States and Alaska, such design features are unnecessary for the Diablo Canyon ISFSI and do not need to be evaluated. In summary, the ISFSI meets the requirements of 10 CFR 72.120(a) and 72.122(b) for snow and ice loadings.

3.2.5 COMBINED LOAD CRITERIA

The HI-STORM 100 System is designed for normal, off-normal, and accident conditions, the definitions and design criteria for which are described in HI-STORM 100 System FSAR Sections 2.2.1, 2.2.2, and 2.2.3, respectively. The service limits, design loads, and load combinations are described in Sections 2.2.5, 2.2.6 and 2.2.7 of the HI-STORM 100 System FSAR.

HI-STORM 100 System FSAR Section 3.1.2 provides additional detail regarding the generic analyses performed using the design criteria, loads and load combinations. This section also includes discussion of the methodologies used in the analyses. Load combinations for the CTF steel structure and equipment are discussed in Section 2.3.3.1 of the HI-STORM 100 System FSAR. The load combinations for the concrete portions of the CTF are in Section 3.3.4.2.4.1. Load combinations for the ISFSI pad concrete and HI-STORM 100SA anchor studs are in Section 3.3.2.3.1 and 3.3.2.3.2, respectively. As noted in Section 3.3.4.2.4, the cask transporter meets the applicable load criteria for the CTF, which are delineated in Section 2.3.3.1 of the HI-STORM 100 System FSAR. Therefore, the load combinations specified by the design criteria are appropriately considered for the design of ITS SSCs, as required by 10 CFR 72.122(b).

3.2.6 LIGHTNING

As noted in Section 2.3.1, thunderstorms at west-coast sites are rare phenomena. However, potential lightning strikes have been evaluated for the HI-STORM 100 System. This evaluation is described in Section 11.2.12.2 of the HI-STORM 100 System FSAR and in Section 8.2.8. The HI-STORM 100 System is a large, metal/concrete cask designed to be stored on an unsheltered ISFSI pad. As such, it may be subject to lightning strikes. If the HI-STORM 100 SYSTEM overpack is struck by lightning, the charge will travel through the steel shell of the overpack into the pad and ultimately into the ground. The overpack outer shell is made of a conductive material (carbon steel). This same shell will have two copper ground cables attached to it providing a direct path to the ground grid. The anchors associated with the HI-STORM 100SA overpack would further enhance grounding. The MPC is protected by the overpack and not subject to direct lightning strikes, which will be absorbed by the overpack. The possibility of lightning striking the cask during transport to and from the CTF is addressed in Sections 4.3 and 8.2.8. Therefore, the lightning design criteria meet the requirements of 10 CFR 72.122(b).

3.2.7 TEMPERATURE AND SOLAR RADIATION

Ambient temperature and incident solar radiation (insolation) values applicable to the ISFSI site are summarized in Section 2.3.2. The highest and lowest hourly temperature, as recorded at one of the recording stations at the Diablo Canyon site, is 97°F in October 1987 and 33°F in December 1990, respectively. The annual average temperature is approximately 55°F. The maximum insolation values for the ISFSI site are estimated to be 766 g-cal/cm^2 per day for a 24-hour period and 754 g-cal/cm^2 for a 12-hour period.

Table 2.2.2 of the HI-STORM 100 System FSAR provides the design environmental and soil temperatures for the HI-STORM 100 System. This includes temperatures and insolation (or lack thereof), as applicable for normal, off-normal, and extreme (accident) conditions. The design temperature for normal conditions is an annual average temperature of 80°F. The extreme (three-day average) temperature limits for the HI-STORM 100 System are -40°F and 125°F, although cask loading, transport, and unloading operations must be conducted with a working area ambient temperature greater than or equal to 0°F (Reference 6, Appendix B, Section 3.4.8).

Sections 4.4.1.1.8 and 4.5.1.1.3 of the HI-STORM 100 System FSAR describe how the HI-STORM 100 System design meets the 10 CFR 71.71(c) insolation requirements (that is, 800 g-cal/cm^2 for flat surfaces and 400 g-cal/cm^2 for curved surfaces) for normal storage conditions and normal handling and transport conditions, respectively. By meeting the insolation requirements of 10 CFR 71.71(c), the HI-STORM 100 System design bounds the maximum insolation values expected for the ISFSI site.

In summary, the HI-STORM 100 System design bounds both the temperature and insolation values expected at the Diablo Canyon ISFSI site. Evaluation of the thermal design for the cask system was carried out during licensing of the HI-STORM 100 System and is documented in the NRC's HI-STORM 100 System Safety Evaluation Report supporting the HI-STORM 100 System CoC.

In support of allowing HBF with a maximum heat load of 28.74 kW a site-specific thermal analysis was performed which evaluated two normal ambient temperatures depending on the storage or transfer configurations (Reference 24). A normal ambient temperature of 65°F was assumed for a loaded MPC contained in a HI-STORM overpack on the ISFSI pad and for a loaded MPC contained located in a HI-STORM overpack within the CTF. All transport configurations with a loaded MPC contained within the HI-TRAC assumed a normal ambient temperature of 100°F.

3.2.8 CRITERIA FOR SLOPE STABILIZATION MEASURES

The ISFSI site is designed to provide a pad site and slopes that are: (a) stable in the long-term under seismic conditions, and (b) conform to the requirements in Appendix A of 10 CFR 100, 10 CFR 72.102, and guidance in NUREG-1567. The design is based on site conditions, field investigations, laboratory testing, material properties, slope analyses, and recommendations discussed in Section 2.6. Surface and overall stability of cut slopes were evaluated using kinematic, limit equilibrium, pseudostatic, and dynamic analyses.

Slope anchorage will conform to Post Tensioning Institute guidelines (Reference 13) and the manufacturer design, installation, and proof testing criteria. Anchor design shall provide a factor of safety over rock block seismic forces of 1.3, as determined in Section 2.6.5.2.2.5. Locations and numbers of anchors will be adjusted as necessary to accommodate any change in site conditions encountered during excavation and installation.

Measures will be taken as required to prevent raveling and limit weathering of the surface and to drain water from inside the hillside to limit buildup of hydrostatic pressure. Design, installation, and testing are to be per ACI 211.2-1998, 214-1997, 304.24-1996, and 506.2-1995; and ASTM A185-1997, C39-2001, and C1116-2000 (References 14 through 20), at a minimum.

Measures will be taken to mitigate any debris or rock falls from the slopes. A defense-in-depth design approach was adopted and an ISFSI slope hazard mitigation system designed that incorporates several protection elements. The rockfall fencing impact design criteria were developed using very conservative results based on the Diablo Canyon slope field observations. A design criterion of 295 ft-tons is used for the maximum impact loading, which envelopes analyses results. The kinetic energy of 295 ft-tons was selected using a hypothetic 5-ft diameter by 10-ft long cylindrical block that has a mass approximately 10 times the mass of the more realistic 3-ft diameter by 3-ft long cylindrical block or close to 20 times the mass of a 3-ft elongated rectangular block that PG&E considers the most probable block size that can reasonably be expected at the site. The rockfall barrier will be manufactured to ISO 9001 quality standards (Reference 21). Additional description of the rockfall barrier fence is provided in Section 4.2.1.1.9.2.

The bench in the cutslope will be sloped to the level of the storage pads.

A drainage system will divert and collect water from slopes, benches, and ISFSI pads in a controlled fashion and convey it to site drainage. Erosion control measures will protect vegetated slopes around the perimeter of the excavated slopes.

3.2.9 REFERENCES

1. 10 CFR 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

2. Regulatory Guide 3.62, Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks, USNRC, February 1989.

3. Standard Review Plan for Spent Fuel Dry Storage Facilities, USNRC, NUREG-1567, March 2000.

4. Standard Review Plan for Dry Cask Storage Systems, USNRC, NUREG-1536, January 1997.

5. Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update.

6. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 1, July 15, 2002.

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14. ACI-211.2-98, Standard Practice for Selecting Proportions for Lightweight Concrete, American Concrete Institute.

15. ACI-214-97, Recommended Practice for Evaluation of Strength Test Results of Concrete, American Concrete Institute.

16. ACI-304.24-96, Placing Concrete by Pumping Methods, American Concrete Institute.

17. ACI-506.2-95, Specification for Shotcrete, American Concrete Institute.

18. ASTM A185-97, Standard Specification for Steel Welded Wire Fabric, Plain, for Concrete Reinforcement, American Society for Testing and Materials.

19. ASTM C39-2001, Standard Test Method for Compressive Strength of Cylindrical Concrete Specimens, American Society for Testing and Materials.

20. ASTM C1116-2000, Standard Specification for Fiber-Reinforced Concrete and Shotcrete, American Society for Testing and Materials.

21. ISO 9001 Quality Standards, Quality Management Systems Requirements, Third Edition, 2000.

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

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

24. Holtec International Document No. HI-2125191, “Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System with up to 28.74 kW Decay Heat”, Revision 6.

3.3 DESIGN CRITERIA FOR SAFETY PROTECTION SYSTEMS

The Diablo Canyon ISFSI is designed for safe storage of spent nuclear fuel and associated nonfuel hardware. The ISFSI storage facility in general, and the HI-STORM 100 System storage casks in particular, are designed to protect the MPC contents and prevent release of radioactive material under normal, off-normal, and accident conditions in accordance with applicable regulatory requirements contained in 10 CFR 72 (Reference 1). Section 3.2 provides the design criteria for environmental conditions and natural phenomena for ISFSI SSCs. This section provides the other design criteria for the ISFSI SSCs.

3.3.1 HI-STORM 100 SYSTEM

3.3.1.1 General

The primary safety functions of each of the major components comprising the Diablo Canyon ISFSI are summarized below, with appropriate references to the HI-STORM 100 System FSAR (Reference 2) or other sections of this FSAR for additional information. Table 3.4-6 provides a list of ASME Code alternatives for the HI-STORM 100 System.

3.3.1.1.1 Multi-Purpose Canister

The MPC is comprised of a cylindrical, strength-welded shell, fuel basket, lid, vent and drain port cover plates, and a welded closure ring. The MPC provides criticality control, decay heat removal, shielding, and acts as the primary confinement boundary for the storage system. The MPC may contain, at prescribed fuel basket locations, a damaged fuel container (DFC) that provides confinement, structural support, and retrievability for damaged fuel assemblies or fuel debris. A detailed description, drawings, and a summary of the design criteria for the MPCs are provided in Sections 1.2.1.1, 1.5, and 2.0.1, respectively, of the HI-STORM 100 System FSAR.

3.3.1.1.2 HI-STORM 100SA Overpack

The HI-STORM 100SA overpack is a rugged, heavy-walled, cylindrical, steel structure. The structure is comprised of inner and outer concentric, carbon-steel shells, a baseplate, and a bolted (with modified anchor thread to preclude bolt binding) top lid (fabricated as a steel-encased concrete disk) with integral outlet vents. The bottom baseplate diameter is increased with gusseted weldments to provide a bolt circle with 16 holes for anchor studs to fasten the overpack to its ISFSI pad anchorage embedment. Either field-installed shims or a permanent circumferential shim plate weldment is used to ensure the proper pre-load is obtained in each anchor stud. The annulus between the inner and outer shells is filled with concrete. A shortened, seismically-anchored version of the overpack, denoted as the HI-STORM 100SA, is used at the Diablo Canyon ISFSI.

The overpack provides support and protection for the MPC during normal, off-normal, and accident conditions including natural phenomena such as tornadoes and earthquakes; provides radiation shielding; and facilitates rejection of decay heat from the MPC to the environs to ensure fuel cladding temperatures remain below acceptable limits. Detailed descriptions, drawings, and a summary of the design criteria for the overpack are provided in Sections 1.2.1.2.1, 1.5, and 2.0.2, respectively, of the HI-STORM 100 System FSAR.

3.3.1.1.3 HI-TRAC 125 Transfer Cask

The HI-TRAC 125 transfer cask is a rugged, heavy-walled, cylindrical steel vessel weighing a maximum of 125 tons during use. The cask guides, retains, protects, and supports the MPC during load handling and transfer operations, including submersion in the SFP where the MPC is loaded. The transfer cask also features a single bottom lid that is removed at the CTF to facilitate the transfer of the MPC to or from the overpack. While submerged, the transfer cask prevents most of the exterior surfaces of the MPC from becoming contaminated by preventing contact with the SFP water.

Upon removal from the SFP, the transfer cask provides shielding to maintain personnel exposure ALARA, and facilitates heat transfer from the MPC to the environs. A more detailed description and a summary of the design criteria for the transfer cask are provided in Sections 1.2.1.2.2, and 2.0.3, respectively, of the HI-STORM 100 System FSAR and in Sections 5.1 and 10.2 of this FSAR. A Diablo Canyon specific derivative of the HI-TRAC 125D version of the transfer cask is used to support Diablo Canyon ISFSI operations. See Section 4.2.3.2.4 for more detailed discussion of HI-TRAC 125D.

3.3.1.2 Protection by Multiple Confinement Barriers and Systems

3.3.1.2.1 Confinement Barriers and Systems

The HI-STORM 100 System provides several confinement barriers for the radioactive contents. Intact fuel assemblies have cladding that provides the first boundary within the MPC preventing release of the fission products. (The MPC confinement and radiological evaluations do not take credit for the cladding.) A DFC prevents the dispersal of gross particulates within the MPC for any fuel assemblies classified as damaged fuel or fuel debris. The MPC is a strength-welded enclosure that provides the confinement boundary for all normal, off-normal and accident conditions, including natural phenomena. The MPC confinement boundary is defined by the MPC baseplate, shell, lid, port cover plates, and the welds joining these components, as shown in Figure 3.3-1. The closure ring provides a redundant boundary. Refer to Figure 4.2-13 and Figure 4.2-14 for details of the MPC confinement boundary design.

3.3.1.2.2 Cask Cooling

The HI-STORM 100 System provides decay heat removal both during processing and final storage of the MPC. As described previously, the transfer cask conducts heat from the MPC until the MPC is transferred to the overpack where convective cooling is established. For loading operations with MPCs containing high burnup fuel when utilizing temporary shielding on the transfer cask, or when unloading MPCs containing high burnup fuel (HBF) that were loaded under Amendment 2 of this license, heat transfer from the transfer cask is augmented by the supplemental cooling system to maintain the fuel cladding temperature below the long term temperature limit for HBF. For other situations, heat transfer from the transfer cask may be augmented by the supplemental cooling system to reduce MPC temperature for operational handling reasons. The thermal design of the HI-STORM 100 System is discussed in detail in Chapter 4 of the HI-STORM 100 System FSAR and in Section 4.2.3.3.3 of this FSAR.

3.3.1.3 Protection by Equipment and Instrumentation Selection

3.3.1.3.1 Equipment

The cask transporter and CTF provide protection functions to the MPC and are discussed in Sections 3.3.3 and 3.3.4, respectively.

3.3.1.3.2 Instrumentation

No instrumentation is required for storage of spent nuclear fuel and associated nonfuel hardware at the Diablo Canyon ISFSI. Due to the welded closure of the MPC, the passively-cooled storage cask design, and the Diablo Canyon ISFSI Technical Specifications (TS) requirement for periodic checks of the casks, the loaded overpacks do not require continuous surveillance and monitoring or operator actions to ensure the safety functions are performed during normal, off-normal or postulated accident conditions.

3.3.1.4 Nuclear Criticality Safety

The HI-STORM 100 System is designed to ensure the stored fuel remains subcritical with k_{eff} less than 0.95 under all normal, off-normal, and accident conditions. A detailed discussion of the criticality analyses for the HI-STORM 100 System is provided in Chapter 6 of the HI-STORM 100 System FSAR. Section 4.2.3.3.5 of this FSAR includes a summary discussion of the HI-STORM 100 System criticality design.

3.3.1.4.1 Control Methods for Prevention of Criticality

The design features and control methods used to prevent criticality for all MPC configurations are the following:

(1) Incorporation of permanent neutron absorbing material (Boral or Metamic) attached to the MPC fuel basket walls with a minimum required loading of the ^{10}B isotope.

(2) Favorable geometry provided by the MPC fuel basket.

(3) Loading of certain fuel assemblies is performed in water with a soluble boron content as specified in the Diablo Canyon ISFSI TS.

There are a number of conservative assumptions used in the HI-STORM 100 System criticality analyses, including not taking credit for fuel burnup, fuel-related burnable neutron absorbers, and only crediting 75 percent of ^{10}B isotope loading in the Boral neutron absorbers or 90 percent of the ^{10}B isotope in the Metamic neutron absorbers. A complete list of the conservative assumptions in the HI-STORM 100 System criticality analyses is provided in Section 6.1 of the HI-STORM 100 System FSAR.

3.3.1.4.2 Error Contingency Criteria

Provisions for error contingency are built into the criticality analyses discussed in Chapter 6 of the HI-STORM 100 System FSAR. Because biases and uncertainties are explicitly evaluated in the analyses, it is not necessary to introduce additional contingency for error.

3.3.1.4.3 Verification Analyses

The criticality analyses for the HI-STORM 100 System were performed using computer codes validated for use in this application under the Holtec International Quality Assurance Program. A discussion of the analysis and the applicable computer codes is provided in Section 6.1 of the HI-STORM 100 System FSAR. Criticality benchmark experiments are discussed in Section 6.5 of the HI-STORM 100 System FSAR.

3.3.1.5 Radiological Protection

Radiation exposure due to the release of material from the storage system is precluded by the confinement boundary design, as discussed in Section 3.3.1.2. The confinement boundary is designed to maintain its integrity during all normal, off-normal, and accident conditions including natural phenomena. Radiation exposure due to direct and sky shine radiation is minimized to the extent practicable through the use of the “time, distance, and shielding” philosophy. This philosophy is implemented at the

Diablo Canyon ISFSI through access control, minimization of required maintenance, and the design of the HI-STORM 100 System.

3.3.1.5.1 Access Control

The Diablo Canyon ISFSI storage pads are surrounded by two fences. The inner is a security fence in compliance with the requirements of 10 CFR 73.55, and may also be a restricted area fence in compliance with 10 CFR 20. The outer is a "nuisance fence," which is capable of being utilized as a restricted area fence, in compliance with 10 CFR 20. Only authorized personnel with a need to be in these areas are permitted entrance. These areas do not require the continuous presence of operators or maintenance personnel. During normal storage operations, the HI-STORM 100 System requires only infrequent, short-duration personnel activity to perform necessary checks on the material condition of the casks and to ensure the overpack air ducts are free of blockage. Higher occupancy times with a greater number of personnel occur during placement of loaded overpacks at the storage pads and during construction of any additional storage pads. These activities are governed by the DCPP radiation protection program to ensure occupational radiation exposures are maintained ALARA. Chapter 7 and Section 9.6 provide additional details regarding the implementation of access control at the Diablo Canyon ISFSI.

3.3.1.5.2 Shielding

The HI-STORM 100 System is designed to minimize radiation doses to DCPP personnel and the public through the use of a combination of concrete, lead, and steel shielding. The HI-STORM 100 System is designed to meet the annual dose limit of 25 mrem specified in 10 CFR 72.104 for annual dose at the DCPP owner-controlled-area boundary. The steel shell of the overpack includes concentric inner and outer shells. The annulus between the shells is filled with unreinforced concrete. The requirements for the unreinforced concrete used for shielding are stated in Appendix 1.D to the HI-STORM 100 System FSAR. The overpack lid is designed as a steel-encased concrete disk to minimize the dose contribution due to sky shine.

The transfer cask is also fabricated from concentric steel shells. The annulus between the shells is filled with lead to provide significant gamma shielding while maintaining the diameter of the transfer cask small enough for loading into the SFP. The transfer cask also includes a water jacket surrounding the main body of the cask to provide necessary shielding for neutrons after the water is drained from the inside of the MPC. The MPC lid and the transfer cask top lid are designed to provide necessary shielding during onsite transport of the transfer cask.

The objective of shielding is to ensure that radiation dose rates at the following locations are below acceptable levels for those locations:

- Immediate vicinity of the storage cask

- Restricted area boundary

- Controlled area (site) boundary

Dose rates in the immediate vicinity of the loaded overpack are an important factor in consideration of occupational exposure. A design objective for the maximum average radial surface dose rate has been established as 60 mrem/hr. Areas adjacent to the inlet and exit vents that pass through the radial shield are limited to 60 mrem/hr. The average dose rate at the top of the overpack is limited to less than 60 mrem/hr.

A detailed discussion of the HI-STORM 100 System generic shielding evaluation, including modeling, source-term assumptions, and resultant dose rates is provided in Chapter 5 of the HI-STORM 100 System FSAR. The site-specific shielding analysis is discussed in Section 7.3. Cumulative estimated occupational exposures and offsite doses for fuel loading, cask handling activities, and storage at the Diablo Canyon ISFSI have been evaluated for DCPP fuel and are discussed also in Sections 7.4 and 7.5.

3.3.1.5.3 Radiological Alarm Systems

The HI-STORM 100 System, when used outside the FHB/AB, does not produce any solid, liquid, or gaseous effluents. Release of loose contamination is not a factor because the HI-STORM overpack is not submerged in the SFP or otherwise subject to contamination. The transfer cask and MPC are submerged in the SFP, but contamination of the MPC is limited to the top of the MPC lid by the annulus seal, which prevents SFP water from coming into contact with the sides and bottom of the MPC. Upon removal from the SFP, the transfer cask and top of the MPC are decontaminated. Therefore, the inadvertent release of loose contamination from the transfer cask produces a negligible dose effect.

The dose rates for a given storage cask at the Diablo Canyon ISFSI are stable and decreasing over time due to the decay of the sources stored inside. There is no credible event that could cause an increase in dose rate from the casks.

Based on the foregoing, there is no need for either airborne or area radiological alarms at the Diablo Canyon ISFSI storage pads or CTF. Radiological alarms, if required for operations inside the FHB/AB, will be implemented under the DCPP radiological protection program.

3.3.1.6 Fire and Explosion Protection

There are no combustible or explosive materials associated with the HI-STORM 100 System, except for the fuel contained in the cask transporter fuel tank. Such materials are not permanently stored within the Diablo Canyon ISFSI protected area. The cask transporter may be parked within the ISFSI, which has been evaluated. However, for conservatism, several hypothetical fire and explosion events were evaluated for the

Diablo Canyon ISFSI. Design criteria for fires and explosions are discussed in Section 2.2 and summarized in Section 3.4.

The generic fire evaluations for both the loaded overpack and the loaded transfer cask are described in Section 11.2.4 of the HI-STORM 100 System FSAR. The fire evaluations assume a maximum of 50 gallons of combustible fuel. Therefore, any transport vehicle used to move the loaded overpack or transfer cask is limited by the Diablo Canyon ISFSI TS to 50 gallons. A site-specific fire evaluation for the Diablo Canyon ISFSI site is provided in Section 8.2.5.

Small overpressures may result from accidents involving explosive materials that are stored or transported near the storage site. Explosion is an accident loading condition evaluated in Section 3.4.7.2 of the HI-STORM 100 System FSAR. A Diablo Canyon ISFSI explosion evaluation for transport to and from the CTF, at the CTF, and at the ISFSI storage pads is discussed in Section 8.2.6.

3.3.1.7 Materials Handling and Storage

3.3.1.7.1 Spent Fuel Handling and Storage

Spent fuel is moved within the DCPP SFP and loaded into the HI-STORM 100 System in accordance with Diablo Canyon ISFSI TS, DCPP TS, and plant procedures. Only fuel assemblies meeting the burnup, cooling time, decay heat, and other limits of the Diablo Canyon ISFSI TS and Section 10.2 are loaded. Burnup uncertainty is not considered when evaluating the eligibility of the fuel assemblies for storage, as an allowance for this uncertainty is not required by regulations. However, PG&E conservatively applies a 5 percent burnup uncertainty allowance when calculating the decay heat load for each loaded MPC. Administrative controls are used to ensure that no unauthorized fuel assemblies are loaded into the HI-STORM 100 System. The Diablo Canyon ISFSI TS and Section 10.2 limits on fuel assemblies authorized for loading, in combination with the design features of the cask system described earlier in this section, ensure that:

- The keff for the stored fuel remains less than 0.95.
- Adequate cooling is provided to ensure peak fuel cladding temperature limits will not be exceeded.
- Radiation dose rates and accumulated doses to plant personnel and the public are less than applicable limits.

The fuel selection process includes a review of reactor operating records for each fuel assembly and nonfuel hardware chosen for loading into the HI-STORM 100 System. Each fuel assembly is classified as intact fuel, damaged fuel, or fuel debris, in accordance with the applicable definitions in the Diablo Canyon ISFSI TS and

Section 10.2. Fuel assemblies classified as damaged fuel or fuel debris are required to be placed in DFCs for storage in the HI-STORM 100 System.

Section 3.3.1.5 discusses contamination as it relates to the operation of the HI-STORM 100 System. The Diablo Canyon ISFSI TS and Section 10.2 provide the necessary limits on MPC moisture removal, helium backfill, and helium leakage prior to declaring the MPC ready for storage. Chapter 8 of the HI-STORM 100 System FSAR provides generic operating procedures for all facets of fuel loading, MPC preparation, and cask handling. The general operating sequence specific to the Diablo Canyon ISFSI is discussed in Sections 5.1 and 10.2 of this FSAR. Implementation procedures are developed based on both generic and site-specific guidelines, as applicable.

The HI-STORM 100 System is designed to allow retrievability of the fuel, as necessary. If the situation warrants fuel retrieval, the MPC is removed from the overpack and returned to the FHB/AB in the transfer cask. The MPC cavity gas is cooled, in accordance with the requirements of the Diablo Canyon ISFSI TS and Section 10.2 and the HI-STORM 100 System FSAR. The MPC is reflooded, the lid removed, and the fuel assemblies are returned to the SFP. Fuel removal activities take place entirely inside the FHB/AB, ensuring that any radiological conditions are controlled and maintained ALARA.

3.3.1.7.2 Radioactive Waste Treatment

There are no radioactive wastes created by the HI-STORM 100 System while in storage at the storage pads, transport to or from the CTF, or at the CTF. During fuel loading and cask preparation activities inside the plant facility, any radioactive wastes created (for example, from decontamination activities) is treated and handled like any other radioactive waste under the DCPP radwaste management program.

3.3.2 ISFSI CONCRETE STORAGE PAD

The Diablo Canyon ISFSI includes a number of individual storage pads, which will be constructed periodically to meet fuel storage needs of DCPP. For simplicity, this discussion refers to a single storage pad. The design criteria are identical for all pads comprising the ISFSI.

3.3.2.1 General

The ISFSI concrete storage pad must be designed to support the weight of the loaded overpacks under all design basis static and dynamic conditions of storage. The pad must also be designed to support the studs that anchor the overpack to the pad and to maintain the integrity of the fastening mechanism embedded in the pad during a postulated design-basis event. The ISFSI pad has been evaluated for the physical uplift, pad sliding, and overturning moments caused by extreme environmental events (for example, tornado missiles, earthquakes, etc.). Therefore, the pad is engineered as a thick, heavily reinforced concrete structure. Concrete shrinkage and thermal stresses are evaluated in Reference 10. Steel reinforcement of the pad is described in Reference 11.

Because tipover of a cask installed in an anchored configuration is not a credible event, the pad does not need to be engineered to accommodate this non-mechanistic event. Since the lifting devices are designed, fabricated, inspected, maintained, operated, and tested in accordance with NUREG-0612 (Reference 4), a drop of the loaded overpack will not occur; therefore, a specific lifting height limit for the cask at the ISFSI is not required to be established. Based on these two criteria, there is no maximum limit on the hardness of the concrete pad and subgrade.

3.3.2.2 Natural Phenomena

The Diablo Canyon ISFSI concrete storage pad is engineered to perform its design function under all loadings induced by design basis natural phenomena. The design criteria for the natural phenomena applicable to the Diablo Canyon ISFSI site, including seismic loadings, tornado wind, and missile loadings, are discussed in Section 3.2.

3.3.2.3 Design Criteria

The design of the ISFSI pad and its embedments must comply with Regulatory Guides 1.142 (Reference 17) and 1.199 (Reference 18), respectively, as well as NUREG-1536 (Reference 5). Regulatory Guide 1.142 endorses ACI 349-97 (Reference 8) with the exception of Appendix B. Regulatory Guide 1.199 endorses Appendix B (February 2001) to ACI 349-01 (Reference 19), which deals with embedments, with exceptions in the area of load combinations. Specifically, the design strength capacity of the embedded base plate, concrete bearing, and diagonal tension shear capacity are in accordance with the design provisions of Regulatory Guide 1.142 and the embedded anchorage is to meet the ductile anchorage provisions set forth in Regulatory Guide 1.199. The materials of construction (for example, anchor stud material and additives in the pad concrete) have been chosen to be compatible with the environment at the Diablo Canyon ISFSI site. ISFSI pad design life is 40 years. The surface anchorage studs (i.e. SA-193 B7 studs and the exposed embedment plates) are properly coated for corrosion protection. Both the ISFSI pad and its embedments are installed in accordance with ACI 349-01.

The use of an embedded steel structure underneath the cask and in the concrete storage pad is employed at the Diablo Canyon ISFSI. The embedded structure permits the cask anchor studs to be preloaded, while the embedded steel structural connection to the concrete does not involve a preload. The embedded structure, while not part of the cask system, is designed in accordance with the AISC Manual of Steel Construction (Reference 6) and Regulatory Guide 1.199 positions.

3.3.2.3.1 Load Combinations for the Concrete Storage Pad

Factored load combinations for ISFSI pad design are provided in the ACI-349-97 and supplemented by the factored load combinations from NUREG-1536, Table 3.1 and Regulatory Guide 1.142, as applicable.

Overturning and Sliding

Since the casks at the Diablo Canyon ISFSI are anchored to the concrete pads, the load combinations from Table 3-1 of NUREG-1536 associated with gross sliding and overturning at the cask/pad interface are not applicable to the cask. The gross sliding of the loaded pad structure was evaluated using a dynamic non-linear seismic analysis to determine the extent of sliding. Pad overturning is not considered as a credible failure mechanism due to the size and geometry of the pad (that is, 68 ft wide by 105 ft long by 7.5 ft thick). The sliding analysis acceptance criterion is: The analysis is to show insignificant impact on the pad's ability to meet its functional requirements and the cask design qualifications as a result of potential pad sliding.

3.3.2.3.2 Load Combinations for the Cask Anchor Studs

The design of the cask anchor studs is governed by the ASME Code, Section III, Subsection NF and Appendix F (Reference 7). The applicable load combinations and allowable stress limits for the anchor studs attaching the cask to the intervening steel support structure are:

Normal Conditions:

Load Combination: D

Code Reference for Stress Limits: NF-3322.1 and NF-3324.6

Off-Normal Conditions:

Load Combination: D+F

Code Reference for Stress Limits: NF-3322.1 and NF-3324.6 with all stress limits increased by a factor of 1.33

Accident Conditions:

Load Combinations: D+E and D+Wt

Code Reference for Stress Limits: Appendix F, Sections F-1334 and F-1335

The axial stress in the cask anchors induced by pretensioning is kept below 75 percent of the material yield stress, such that during a seismic event the maximum stud axial stress remains below the limit prescribed for bolts in the ASME Code, Section III, Subsection NF, for Level D conditions.

3.3.2.3.3 Maximum Permissible Tornado Wind and Missile Load

During a tornado event, the HI-STORM 100 System may be subjected to a constant wind force and differential pressures. It may also be subjected to impacts by tornado-borne missiles. In contrast to a free-standing cask, the anchored cask system is capable of withstanding greater lateral pressures and impulsive loads from large missiles. The anchored HI-STORM 100SA cask design at the Diablo Canyon ISFSI has been analyzed assuming the lateral force from the site-specific design-basis, large-tornado-missile impact occurs at the worst-case height on the cask and the force created by the tornado wind action and differential pressure acts simultaneously at cask mid-height. The resulting overturning moment is bounded by the maximum seismic overturning moment applied to the cask anchorage embedment and the pad.

3.3.3 CASK TRANSPORTER

3.3.3.1 General

The cask transporter is a U-shaped tracked vehicle used for lifting, handling, and onsite transport of loaded overpacks and the transfer cask. The cask transporter will also be used to transfer the MPC from the transfer cask to the overpack. The functional specification for the transporter is provided in Reference 12. The cask transporter does not have a suspension system (for example, springs). The transporter consists of the vehicle main frame, the lifting towers, an overhead beam system that connects the parallel lifting towers, a cask restraint system, the drive and control systems, and a series of cask lifting attachments. The casks are individually carried within the internal footprint of the transporter tracks (Sections 4.3 and 4.4 provide more detailed descriptions of cask transportation components and operating characteristics). The cask is supported by the lifting attachments that are connected to the overhead beam. The overhead beam is supported at the ends by a pair of lifting towers. The lifting towers transfer the cask weight directly to the vehicle frame and ultimately to the tracks and the transport route surface. The cask transporter has the added capability of being able to raise and lower an MPC between the transfer cask and the overpack when used in conjunction with the CTF and the capability of lifting the overpack in and out of the CTF. The transporter’s CTF functions are contained in Section 1.2 of Reference 12.

3.3.3.2 Design Criteria

The key design criteria for the cask transporter are summarized in Table 3.4-4. The bases for these criteria are discussed in the sections below.

3.3.3.2.1 Design Life

The cask transporter design life of 20 years has been established based on a reasonable length of time for a vehicle of its type with normal maintenance. The cask transporter may be replaced or recertified for continued use at the end of its design life.

3.3.3.2.2 Environmental Design Criteria

The cask transporter is an “all-weather” vehicle. It is designed to operate in both rain and snow over a temperature and humidity range that bounds the historical conditions at the Diablo Canyon site. Materials that would otherwise degrade in a coastal marine environment are appropriately maintained.

A lightning strike on the cask transporter would not structurally affect the ability of the transporter to hold the load. Due to the massive amount of steel in the structure, the current would be transmitted to the ground without significantly damaging the transporter. However, should the lightning strike result in a loss of electrical or engine function, the vehicle will be automatically stopped (when moving) and the brakes applied or load movement will be stopped and mechanical locks applied to hold the load (when lifting or lowering). The driver may be affected by a lightning strike. Therefore, the transporter design includes fail-safe features to automatically stop the vehicle (when moving) or stop load movement and apply mechanical locks to hold the load (when lifting or lowering) if the operator is incapacitated for any reason.
Flooding is not a concern on the transport route as discussed in Section 3.2.2. Sources of fires and explosions have been identified and evaluated. Fixed sources of fire and explosion are sufficiently far from the transport route to not be of concern (Section 2.2). Mobile sources of fire and explosion, such as fuel tanker trucks, are kept at a safe distance away from the transporter during cask movement through the use of administrative controls. The cask transporter is diesel-powered and is limited to a maximum fuel volume consistent with that used in the HI-STORM 100 System fire accident analysis. The hydraulic fluid used in the cask transporter is nonflammable.

3.3.3.2.3 Regulatory Design Criteria and Industry Standards

The transporter is designed, fabricated, inspected, maintained, operated, and tested in accordance with applicable guidelines of NUREG-0612, which allows the elimination of the need to establish a cask lift height limit.

3.3.3.2.4 Performance Design Criteria

As described in Section 4.4, the cask transporter must lift and transport either the loaded transfer cask or the loaded overpack, including the weight of all necessary ancillary lift devices such as rigs and slings. The loaded overpack, being the heavier of the two casks to be lifted, provides the limiting weight for the design of the transporter.

3.3.3.2.5 Stability Design Criteria

The cask transporter is custom designed for the Diablo Canyon site, including the transport route with its maximum grade of approximately 8.5 percent. It remains stable and does not experience structural failure, tip over, or leave the transport route should a design-basis seismic event occur while the loaded transfer cask is being moved to the CTF, while transferring an MPC at the CTF, while moving a loaded overpack from the CTF to the storage pad, or while moving a loaded overpack on the storage pad. In addition, the cask transporter is designed to withstand design-basis tornado winds and tornado-generated missiles without an uncontrolled lowering of the load or leaving the transport route. All design criteria for natural phenomena used to design the cask transporter are specific to the Diablo Canyon site (Sections 3.2 and 3.4 provide further information).

3.3.3.2.6 Drop Protection Design Criteria

In accordance with NUREG-0612, prevention of a cask or MPC drop is provided by enhancing the reliability of the load supporting systems by design, using a combination of component redundancy and higher factors of safety than would normally be used for a commercial lift device. Load supporting components include the special lifting devices used to transfer the force of the payload to the cask transporter lift points (including attachment pins, as appropriate), the cask transporter lift points, the overhead beam, the lifting towers, and the vehicle frame. The design criteria for each of the components of the cask transporter are the following:

<u>Slings and Special Lifting Devices</u>

The HI-TRAC/HI-STORM lift links, MPC downloader slings and overpack lifting brackets are designed to applicable guidelines of NUREG-0612.

<u>Cask Transporter Lift Points, Overhead Beam, Vehicle Body and Seismic Restraints</u>

The cask transporter lift points, overhead beam, and load supporting members of the vehicle body (whose failure would result in an uncontrolled lowering of the load) are designed to applicable guidelines of NUREG-0612.

<u>Lifting Towers</u>

The lifting towers are designed with redundant drop protection features. The primary cask lifting device is the hydraulic system, which prevents uncontrolled cask lowering through the control of fluid pressure in the system. A mechanical backup load retaining device, independent of the hydraulic lifting cylinders, is provided in case of failure of the hydraulic system. This consists of load blocks, pawl and detent, locking pins, or other suitably designed positive mechanical locking device.

3.3.3.2.7 Drive System Design Criteria

The cask transporter is capable of forward and reverse movement as well as turning and stopping. It includes an on-board engine capable of supplying enough power to perform its design functions. The cask transporter includes fail-safe service brakes (that automatically engage in any loss of power (i.e., a loss of hydraulic or electrical) and an independent parking brake on each tractor motor. The brake system is capable of stopping a fully loaded cask transporter on the maximum design grade. The cask transporter is also equipped with an automatic drive brake system that applies the brakes if there is a loss of hydraulic pressure (e.g., spring set tractor motor brake) or decelerates if the drive controls are released (e.g., hydraulic system pressure relief). The fully-loaded cask transporter is not capable of coasting on a 10 percent downward grade with the brakes disengaged due to the passive resistance in the drive system that is inherent in the design of each multi-stage, planetary gear, tractor drive transmission.

3.3.3.2.8 Control System Design Criteria

The cask transporter is equipped with a control panel that is suitably positioned on the transporter frame to allow the operator easy access to the controls located on the control panel and, at the same time, allow an unobstructed view of the cask handling operations. The control panel provides for all-weather operation. The control panel includes controls for all cask transporter operations including speed control, steering, braking, load raising and lowering, cask restraining, engine control and “dead-man” and external emergency stop switches. A radio-remote control module can alternatively be used to control the drive and lift functions.
The drive and lift functions are capable of being operated by a single operator from an on-board console or from the ground using the radio-remote. The operator is within view of all gauges and instruments necessary for the operator to monitor the condition and performance of both the power source and hydraulic systems when using the vehicle-mounted control panel. A cask lift-height indicator is provided to ensure the loaded casks are lifted only to those heights necessary to accomplish the operational objective in progress.

3.3.3.2.9 Cask Restraint Design Criteria

The cask transporter is equipped with a cask restraint to secure the cask during movement. The restraint is designed to prevent lateral and transverse swinging of the cask during cask transport. The restraint is designed to preclude damage to the cask exterior with padding or other shock dampening material used, as necessary.

3.3.3.2.10 Lateral Restraint Design Criteria

The cask transporter structure is designed to accommodate external loading from a lateral restraint system at the CTF to preclude seismic interaction with the cask system during MPC transfer operations in the CTF. The structural components of the transporter resisting the restraint loads are designed to the applicable limits of ASME Section III, Subsection NF including Appendix F (Reference 7).

3.3.4 CASK TRANSFER FACILITY

3.3.4.1 General

The CTF is used in conjunction with the cask transporter to accommodate MPC transfers between the transfer cask and the overpack. The CTF is designed to position an overpack sufficiently below grade where the transfer cask can be mated to the overpack using the cask transporter. The surface of the CTF contains an approach pad that supports the loaded transporter and provides a laydown area for the transfer cask, mating device, seismic restraint, and other load handling equipment.

3.3.4.2 Design Criteria

The design criteria for the specific subcomponents are discussed below. The CTF is designed to withstand a design-basis seismic event. The design life of the CTF is 40 years. Design criteria for the CTF are summarized in Table 3.4-5 and presented in Reference 13.

3.3.4.2.1 Main Shell Design Criteria

A cylindrical steel shell forms the opening in the ground into which the overpack is lowered to the CTF baseplate. The main shell forms a cylindrical opening of approximately 150 inches in diameter and approximately 178 inches deep. The shell is also equipped with a sump for collecting and disposing of incidental water from the CTF. The surrounding area is reinforced concrete. The resulting structure is a flat-surfaced pad with a steel-lined hole. The main shell is designed in accordance with applicable portions of ASME Section III, Subsection NF.

3.3.4.2.2 HI-STORM Mating Device

A mating device 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. The mating device also facilitates the removal of the bottom lid from the transfer cask prior to MPC transfer operations. During the use of the mating device additional temporary shielding is provided around the mating device as needed to minimize occupational dose. Use of the temporary shielding will be administratively controlled.

3.3.4.2.3 Adjustable Wedges

Adjustable wedges shim the gap between the overpack and the top of the CTF main shell to provide lateral structural support.

3.3.4.2.4 Reinforced Concrete Support Structure

The reinforced concrete surrounding the shell is capable of supporting a loaded transporter and handling any seismic loads applied through the shell. The reinforced concrete base pad supports the CTF shell and a steel baseplate. The approach pad is designed for the weight of the transporter with a loaded overpack. Independent tie-down blocks at the surface of the CTF are provided to hold the transporter in place during the MPC transfer operation. The reinforced concrete structure is designed in accordance with ACI-349-97 and Appendix B to ACI 349-01 as endorsed by Regulatory Guides 1.142 and 1.199, respectively. The reinforced concrete structure is installed in accordance with ACI 349-01.

3.3.4.2.4.1 Design Load Combinations

Factored load combinations for the CTF concrete structure design are provided in the ACI 349-97 and supplemented by the factored load combinations from NUREG-1536 (Reference 5), Table 3.1, and Regulatory Guide 1.142, as applicable.

3.3.4.2.5 Cask Transporter Lateral Restraints

The cask transporter lateral restraint system is designed to apply external restraint loading to the cask transporter structure. As discussed in Reference 16, the restraints will be steel struts or similar equipment suitably sized to restrain the transporter by transferring the restraint loading to the ground adjacent to the CTF foundation. The restraints are designed to meet the stress limits of ASME Section III, Subsection NF including Appendix F (Reference 7). The surface-level, in-ground portion of the restraints are designed in accordance with ACI 349-97 and Appendix B to ACI 349-01, which are endorsed by Regulatory Guides 1.142 and 1.199, respectively. The cask transporter lateral restraints are installed in accordance with ACI 349-01.

3.3.5 REFERENCES

1. 10 CFR 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

2. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 1A, January 2003.

3. Deleted in Revision 2.

4. Control of Heavy Loads at Nuclear Power Plants, USNRC, NUREG-0612, July 1980.

5. Standard Review Plan for Dry Cask Storage Systems, USNRC, NUREG-1536, January 1997.

6. Manual of Steel Construction, American Institute of Steel Construction, 9th Edition.

7. Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, American Society of Mechanical Engineers, 1995 Edition including 1996 and 1997 Addenda.

8. ACI-349-97, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, 1997.

9. Deleted in Revision 1.

10. Calculation PGE-009-CALC-006, “ISFSI Cask Storage Pad Concrete Shrinkage and Thermal Stresses.”

11. Calculation PGE-009-CALC-007, “ISFSI Cask Storage Pad Steel Reinforcement.”

12. Holtec International Report No. HI-2002501, “Functional Specification for the Diablo Canyon Transporter,” Revision 8.

13. Holtec International Report No. HI-2002570, “Design Criteria Document for the Diablo Canyon Cask Transfer Facility,” Revision 5.

14. Deleted in Revision 2.

15. Deleted in Revision 2.

16. PG&E Letter DIL-03-015 to the NRC, Additional Information on Cask Transfer Facility Cask Transporter Lateral Restraint System, December 4, 2003.

17. Regulatory Guide 1.142, Safety Related Concrete Structure for Nuclear Power Plants (Other than Reactor Vessels and Containment), USNRC, November 2001.

18. Regulatory Guide 1.199, Anchoring Components and Structural Supports in Concrete, USNRC, November 2003.

19. ACI 349-01, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, 2001.

3.4 SUMMARY OF DESIGN CRITERIA

The major ISFSI structures, systems, and components (SSCs) classified as important to safety are the HI-STORM 100 System, the storage pad, the transporter, and the cask transfer facility (CTF). The principal design criteria for these SSCs are summarized in Tables 3.4-1 through 3.4-6.

- Table 3.4-1 provides the site-specific design criteria for environmental conditions and natural phenomena.
- Table 3.4-2 provides design criteria applicable to the HI-STORM 100 System. Detailed design criteria for the MPC, the overpack, and the transfer cask are listed in the HI-STORM 100 System FSAR (References 1 and 3), Tables 2.0.1, 2.0.2, and 2.0.3, respectively. Detailed anchorage design requirements are discussed in Section 4.2.
- Table 3.4-3 provides the design criteria for the storage pad.
- Table 3.4-4 provides the design criteria for the cask transporter.
- Table 3.4-5 provides the design criteria for the CTF.
- Table 3.4-6 provides a list of ASME Code alternatives for the HI-STORM 100 System.

3.4.1 REFERENCES

1. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 1A, January 2003.

2. Deleted in Revision 2.

3. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 7, August 17, 2009

TABLE 3.1-1

SUMMARY OF FUEL PHYSICAL CHARACTERISTICS

Fuel Parameter	Diablo Canyon[a]	MPC Limiting Values[b]
Fuel Assemblies		
Rod array	17 x 17	17 x 17
$U0_2$ rods per assembly	264	264
Rod pitch, in.	0.496	≤0.496
Overall dimensions, in.	8.426 x 8.426	≤8.54 x 8.54
Uranium weight per assembly, kg	461.2/423.0[c]	≤467
Assembly weight with nonfuel hardware, lb	≤1,621	≤1,680
Number of guide thimbles per assembly	24	≤25
Fuel Rods		
Active fuel length, in.	144	≤150
Cladding outside diameter, in.	0.374/0.360	≥0.372/≥0.360
Cladding inside diameter, in.	0.329/0.315	≤0.331/≤0.315
Cladding material	Zircaloy-4 or ZIRLO	Zr or Zr alloy
Fuel Pellets		
Material	UO_2 sintered	UO_2
Diameter, in.	0.3225/0.3088	≤0.3232/≤0.3088

(a) These are the DCPP fuel characteristics. See Table 4.1-1 of the DCPP FSAR Update.

(b) In many instances, allowable fuel parameters are a function of several factors such as canister type and fuel condition. In all cases, the fuel stored is within the limits controlled by the Diablo Canyon ISFSI Technical Specifications and specified in FSAR Section 10.2, which are consistent with the applicable limiting values from the Holtec CoC No. 1014, Amendment 1.

(c) LOPAR/VANTAGE 5

TABLE 3.1-2

SUMMARY OF FUEL THERMAL AND RADIOLOGICAL CHARACTERISTICS

Parameter	Diablo Canyon[a]	MPC Limiting Values
Maximum decay heat per assembly	1,500 Watts	See footnote (b)
Maximum assembly average burnup	~58,000 MWD/MTU	See footnote (b)
Maximum initial enrichment	5 percent	See footnote (b)
Minimum cooling time	5 years	See footnote (b)

(a) These are the DCPP fuel characteristics. The DCPP license limits the peak fuel rod burnup to 62,000 MWD/MTU, which corresponds to a fuel assembly average burnup of approximately 58,000 MWD/MTU.

(b) In many instances, allowable fuel parameters are a function of several factors such as MPC type, fuel condition, and the use of a uniform or regionalized loading strategy. Some are also dependent upon one another (that is, burnup and cooling time or decay heat and cooling time). The limiting assembly decay heat, burnup, initial enrichment, and cooling times are specified in FSAR Section 10.2, which is consistent with the applicable limiting values in the Holtec CoC No. 1014, Amendment 3. In all cases, the fuel stored is within the limits controlled by the Diablo Canyon ISFSI Technical Specifications and specified in FSAR Section 10.2.

TABLE 3.2-1

HI-STORM 100 SYSTEM AND DIABLO CANYON ISFSI SITE
TORNADO DESIGN PARAMETERS

Parameters	Value	
	HI-STORM 100 System[a]	Diablo Canyon ISFSI Site[b]
Rotational wind speed (mph)	290	157
Translational wind speed (mph)	70	43
Maximum wind speed (mph)	360	200
Pressure drop (psi)	3.0	0.86
Rate of pressure drop (psi/sec)	Instantaneous	0.36

(a) Table 2.2.4 of HI-STORM 100 System FSAR, except for rate of pressure drop, which is provided in FSAR Section 3.4.8

(b) Section 3.3.2.1.1 of DCPP FSAR Update

TABLE 3.2-2

HI-STORM 100 SYSTEM AND DIABLO CANYON SITE
TORNADO MISSILE DESIGN PARAMETERS

HI-STORM 100 System [(a)]		
Missile Description	**Mass (kg)**	**Velocity (mph)**
Automobile	1,800	126
Rigid Solid Steel Cylinder (8 in. diameter)	125	126
Solid Sphere (1 in. diameter)	0.22	126

Diablo Canyon Power Plant Site [(b)]		
Missile Description	**Mass (kg)**	**Velocity [(c)] (mph)**
Automobile	1,814	33.3
10 ft long x 3 in. diameter Schedule 40 pipe	34.5	66.7
4 in. x 12 in. x 10 ft board	49.0	200

Additional Missiles Evaluated for Diablo Canyon ISFSI Site			
Missile Description	**Mass (kg)**	**Velocity (mph)**	
		Diablo Licensing Basis	**Holtec Evaluation[(d)]**
6 in. diameter Sch 40 pipe[(e)]	130	7	93.9
Utility Pole[(e)]	510	35	107.4
12 in. diameter Sch 40 pipe[(e)]	340	5	62.6
(2 in. x 2 in. x 1/8 in. x 5 ft) Long Steel Angle[(f)]	3.9	157[(g)]	157
500-kV Insulator String[(f)]	344.7	157[(g)]	157
500-kV Insulator Segments and Miscellaneous Conductor Hardware[(f)]	6.8	157[(g)]	157
1 in. diameter Steel Rod[(e)]	4	5	89.5

(a) Table 2.2.5 of HI-STORM 100 System FSAR
(b) Section 3.3.2.1.2 of the DCPP FSAR Update
(c) Tornado wind velocity is 200 mph per Section 3.3.2.1.1 of the DCPP FSAR Update. Missile velocities are presented in Section 3.3.2.1.2 of the DCPP FSAR Update as fractions of tornado wind speed.
(d) Velocities used by Holtec in a bounding analysis of missile effects based on Region II 300 mph wind velocity.
(e) Additional missile based on NUREG-0800, Section 3.5.1.4, Spectrum II missile table.
(f) Unique missile for Diablo Canyon ISFSI.
(g) Conservatively assumed as equal to tornado rotational wind speed.

TABLE 3.4-1

DESIGN CRITERIA FOR ENVIRONMENTAL CONDITIONS AND NATURAL PHENOMENA APPLICABLE TO THE MAJOR ISFSI STRUCTURES, SYSTEMS, AND COMPONENTS

Design Criterion	Design Value	Reference Documents
Wind	80 mph with a gust factor of 1.1 Condition is bounded by tornado wind	Diablo Canyon ISFSI FSAR Section 3.2.1
Tornado	200 mph maximum speed 157 mph rotational speed 43 mph translational speed 0.86 psi pressure drop 0.36 psi/sec pressure drop rate	Diablo Canyon ISFSI FSAR Section 3.2.1, Table 3.2-1
Tornado Missiles	See Diablo Canyon ISFSI FSAR Table 3.2-2	Diablo Canyon ISFSI FSAR Section 3.2.1
Flood	Design-basis flooding event is not considered credible	Diablo Canyon ISFSI FSAR Section Section 3.2.2
Seismic	See Diablo Canyon ISFSI FSAR Section 3.2.3	Diablo Canyon ISFSI FSAR Section 3.2.3
Snow & Ice	Design-basis snow and ice loadings are not considered credible	Diablo Canyon ISFSI FSAR Section 3.2.4
Explosion	A fuel tank for the transporter, load handling equipment, or other vehicle 7-gallon propane bottles being transported via Reservoir Road Standard acetylene bottles transported to the vehicle maintenance shop via Reservoir Road A 250-gallon propane tank, a 2,000-gallon #2 diesel fuel oil tank, and a 3,000-gallon gasoline tank located in close proximity to each other and beside the main plant road and approximately 1,200 ft from the transport route to the ISFSI storage pad	Diablo Canyon ISFSI FSAR Sections 2.2.2.3, 3.3.1.6, and 8.2.6

Design Criterion	Design Value	Reference Documents
Explosion (continued)	The Unit 2 main bank transformers, which contain approximately 13,000 gallons each of mineral oil and are located 160 ft from the transport path Standard compressed gas bottles (air, nitrogen, argon, CO_2) located inside the RCA and near the El. 115' south gate Hydrogen gas facility adjacent to the transport route Acetylene bottles stored on the east side of the cold machine shop	
Fire	A fuel tank for the transporter, load handling equipment, or other vehicle Local stationary fuel tanks Local combustible materials Nearby grass/brush fire	Diablo Canyon ISFSI FSAR Sections 2.2.2.2, 3.3.1.6, and 8.2.5
Ambient Temperatures	Annual Average = 55°F Minimum recorded = 33°F Maximum Recorded = 97°F Extreme Temperature Range = 24°F to 104°F	Diablo Canyon ISFSI FSAR Sections 2.3.2, 3.2.7, 8.2.6, and 8.2.10
Insolation	766 g-cal /cm^2 maximum for a 24-hr period	Diablo Canyon ISFSI FSAR Section 3.2.7
Lightning or 500-kV Line Strike	No loss of canister confinement integrity Minor degradation of cask system shielding due to arc damage is bounded by tornado missile analysis criterion above	Diablo Canyon ISFSI FSAR Sections 3.2.6 and 8.2.8
500-kV Transmission Tower Collapse	No loss of canister confinement integrity	Diablo Canyon ISFSI FSAR Sections 3,2.5 and 8.2.16
Slope Stability	No loss of canister confinement integrity	Diablo Canyon ISFSI FSAR Section 3.2.8

TABLE 3.4-2

PRINCIPAL DESIGN CRITERIA APPLICABLE TO THE HI-STORM 100 SYSTEM

Design Criterion	Design Value	Reference Documents
GENERAL		
HI-STORM 100 System Design Life	40 years	Holtec FSAR[(a)], Section 2.0.1 and Diablo Canyon ISFSI FSAR Section 3.3.1.3.1
ISFSI Storage Capacity	140 casks (138 required + 2 spare locations)	Diablo Canyon ISFSI FSAR Section 3.1
Number of Fuel Assemblies	4,400 (approx.)	Diablo Canyon ISFSI FSAR Section 3.1
Nonfuel Hardware	Borosilicate absorber rods Wet annular burnable absorber rods Thimble plug devices Rod cluster control assemblies Neutron Source Assemblies Instrument Tube Tie Rods	Diablo Canyon ISFSI FSAR Section 3.1.1.3 and Table 3.1-1 and Table 10.2-10
SPENT FUEL SPECIFICATIONS		
Type of Fuel	Non-consolidated PWR - Westinghouse 17 x 17 LOPAR and VANTAGE 5, and later variants	Diablo Canyon ISFSI FSAR Section 3.1.1, 10.2.1.1, and Tables 10.2-1 through 10.2-5
Fuel Characteristics	See Diablo Canyon ISFSI FSAR Tables 3.1-1 and 3.1-2 for physical, thermal, and radiological characteristics	See Diablo Canyon ISFSI FSAR Section 3.1.1, 10.2.1.1 and Tables 10.2-1 through 10.2-5
Fuel Classification	Intact, Damaged, Debris	Diablo Canyon ISFSI FSAR Section 3.1.1, 10.2.1.1, Tables 10.2-1 through 10.2-10, and Diablo Canyon ISFSI TS
STRUCTURAL DESIGN		
Design Codes	ASME III-95, with 1996 and 1997 Addenda, Subsection NB ASCE 7-88; ANSI N14.6 (93); ACI-318 (95); and ACI-349 (85)	Holtec FSAR[(a)], Tables 2.2.6, 2.2.7, 2.2.14, and 2.2.15
Environmental Conditions and Natural Phenomena	See Diablo Canyon ISFSI FSAR Table 3.4-1	Diablo Canyon ISFSI FSAR Sections 3.2 & 3.3

TABLE 3.4-2

Design Criterion	Design Value	Reference Documents
Weights	Maximum loaded transfer cask handling weight = 250,000 lb Maximum loaded overpack weight = 360,000 lb Transporter weight = 190,000 lb	Reference 4 Section 9.1.4.2.1.3 (fuel handling building crane capacity); Holtec FSAR[(a)], Section 3.2 HI-2002501, Rev.8, Table 4.2
STRUCTURAL DESIGN (continued)		
MPC Internal Pressure	Normal/off-normal = 100 psig Accident = 200 psig	Holtec FSAR[(a)], Table 2.0.1
Cask Loads and Load Combinations	See HI-STORM 100 System FSAR	Holtec FSAR[(a)], Sections 2.2.1 through 2.2.3 and Tables 2.2.13 and 2.2.14
THERMAL DESIGN		
Maximum Cask Heat Duty	Varies by MPC model, fuel loading strategy (uniform loading vs. regionalized loading), fuel assembly burnup, and cooling time Maximum PWR basket heat duty = 28.74 kW	Holtec FSAR[(a)], Section 4.4.2 and Table 4.4.28
Peak Fuel Cladding Temperature Limits	Long term (normal) = 752°F (400°C) Short term (accident) = 1058°F	Holtec FSAR[(b)], ISG-11, Rev. 3, Tables 4.3.7 and 4.A.2 for normal conditions and Table 4.3.1 for short term conditions
Other SSC Temperature Limits	Varies by material	Holtec FSAR[(a)(b)], Tables 2.0.1 through 2.0.3
MPC Backfill Gas	99.995% pure helium	Holtec FSAR[(a)], Section 1.2.2.1 and Diablo Canyon ISFSI FSAR Section 10.2.2.4
Maximum Air Inlet to Outlet Temperature Rise	126°F	Holtec CoC No. 1014, Amendment 1, Appendix A, LCO 3.1.2
RADIATION PROTECTION AND SHIELDING DESIGN		
Storage Cask Dose Rate Objectives	60 mrem/hr on the sides, top, and adjacent to air ducts	Holtec FSAR[(a)], Section 2.3.5.2; and Diablo Canyon ISFSI FSAR 3.3.1.5.2
Occupational Exposure Dose Limits	5 rem/yr or equivalent	10 CFR 20.1201
Restricted Area Boundary Dose Rate Limit	2 mrem/hr	10 CFR 20.1301
Normal Operation Dose Limits to Public	25 mrem/yr whole body 75 mrem/yr thyroid 25 mrem/yr and other critical organ	10 CFR 72.104

TABLE 3.4-2

Design Criterion	Design Value	Reference Documents
RADIATION PROTECTION AND SHIELDING DESIGN (continued)		
Accident Dose Limits to Public	5 rem TEDE 50 rem DDE plus CDE 15 rem lens dose equivalent 50 rem shallow dose equivalent to skin or extremity	10 CFR 72.106
Overpack Unreinforced Concrete	Various	Holtec FSAR[b], Appendix 1.D, NUREG-1536
CRITICALITY DESIGN		
Maximum initial fuel enrichment	≤ 5%	Holtec FSAR[a], Section 6.2.2.4; and Diablo Canyon ISFSI FSAR Sections 3.3.1.4.1 and 3.1.1.1, Tables 10.2-1 through 10.2-5, and the Diablo Canyon ISFSI TS
Control Method (Design Features)	MPC-32 fuel storage cell pitch ≥ 9.158 In and B-10 loading: ≥ 0.0372 g/cm^2 (Boral) or ≥ 0.0310 g/cm^2 (Metamic) MPC 24: flux trap size ≥1.09 inch and B-10 loading: ≥0.0267 g/cm^2 (Boral) or ≥ 0.0223 g/cm^2 (Metamic) MPC-24E AND 24EF: flux trap size ≥0.776 inch for cells 3,6, 19 and 22; ≥1.076 inch for all other fuel cells; and B-10 loading ≥ 0.0372 g/cm^2 (Boral) or ≥ 0.0310 g/cm^2 (Metamic)	Diablo Canyon ISFSI TS

TABLE 3.4-2 Sheet 4 of 4

Design Criterion	Design Value	Reference Documents
Control Method (Operational)	For all MPC with maximum initial enrichment of $\leq$ 4.1 wt % $\geq$ 2000 ppm soluble boron in the MPC water during loading and unloading For MPC-24/24E/24EF with maximum initial enrichment of >4.1 and $\leq$ 5.0 wt% $\geq$ 2000 ppm soluble boron in the MPC water during loading and unloading For MPC-32 with maximum initial enrichment of 5.0 wt% $\geq$ 2600 ppm soluble boron in the MPC water during loading and unloading (soluble boron scaled based on MPC maximum initial enrichment between 4.1 and 5.0 wt%).	Diablo Canyon ISFSI TS
CRITICALITY DESIGN (continued)		
Maximum k_{eff}	<0.95	Holtec FSAR[a], Table 2.0.1; and Diablo Canyon ISFSI FSAR Section 3.3.1.4
CONFINEMENT DESIGN		
Confinement Method	MPC with redundant welds	Holtec FSAR[a], Section 2.3.2.1 and Chapter 7
Confinement Barrier Design	Multi-purpose canister: ASME III, NB	Holtec FSAR[a], Tables 2.2.6 and 2.2.15, and Diablo Canyon ISFSI TS
Maximum Confinement Boundary Leak Rate	5.0×10^{-6} atm-cm^3/sec for MPCs not containing high burnup fuel Leaktight per ANSI N14.5-1997 (1.0×10^{-7} atm-cc^3/sec), for MPCs containing high burnup fuel	Diablo Canyon ISFSI FSAR Section 10.2.2.5

(a) Holtec HI-STORM 100 FSAR, Revision 1A
(b) Holtec HI-STORM 100 FSAR, Revision 7

TABLE 3.4-3

DESIGN CRITERIA FOR STORAGE PAD

Design Criterion	Design Value	Reference Documents
Storage Pad Design Codes	NUREG-1536; RG 1.142 (ACI-349-97), RG 1.199 (ACI 349-01 Appendix B)	Diablo Canyon ISFSI FSAR Sections 3.3.2.3 and 4.2.1.1.2
Design Life	40 years	Diablo Canyon ISFSI FSAR 3.3.2.3
Maximum Single Loaded Cask Weight	360,000 lb	Holtec FSAR[(a)], Table 2.0.1
Transporter with Loaded HI STORM	550,000 lb	Holtec FSAR[(a)], Table 2.0.1 and assumed value
Maximum Number of Casks on a Single Pad	20	Diablo Canyon ISFSI FSAR Sections 1.3 and 4.1
Maximum Number of Pads at the ISFSI	7	Diablo Canyon ISFSI FSAR Sections 1.3 and 4.1
Operating Temperature Range	0-100^{o}F	DCPP FSAR Update, Section 2
Concrete Pad Strength	5,000 psi at 90 days	RG 1.142 (ACI 349-97); RG 1.199 (ACI 349-01 Appendix B)
Pad Loads and Load Combinations	Various	NUREG-1536, Table 3-1
Cask Anchor Stud Loads and Load Combinations	Various	ASME, Section III, Subsection NF and Appendix F; and Diablo Canyon ISFSI FSAR Section 3.3.2.3.2
Environmental Conditions and Natural Phenomena	See Diablo Canyon ISFSI FSAR Table 3.4-1	Diablo Canyon ISFSI FSAR Sections 3.2 & 3.3

(a) Holtec HI-STORM 100 System FSAR, Revision 1A

TABLE 3.4-4

DESIGN CRITERIA FOR TRANSPORTER

Design Criterion	Design Value	Reference Documents
Transporter Design Codes	Purchase commercial grade and qualify by testing prior to use in accordance with NUREG-0612	Diablo Canyon ISFSI FSAR Sections 3.3.3 and 4.3.2.1, and Diablo Canyon ISFSI TS
Design Life	20 years	Diablo Canyon ISFSI FSAR Section 3.3.3.2.1
Maximum Payload	360,000 lb	Holtec FSAR[(a)], Table 2.0.1
Transporter Weight	190,000 lb	HI-2002501, Rev. 8, Table 4.2
Loaded Travel Speed	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	50 gallons	Diablo Canyon ISFSI TS and FSAR Section 2.2.2.3
Maximum Hydraulic Fluid Volume	Unlimited (must be non-flammable)	Diablo Canyon ISFSI TS and FSAR Section 3.3.3.2.2
Operating Temperature Range	0-100^{o}F	DCPP FSAR Update, Section 2
Redundancy and Safety Factors for Load Path Structures and Special Lifting Devices	Per the applicable guidelines of NUREG-0612	Holtec FSAR[(a)], Section 2.3.3.1
Hoist Load Factor	15%	CMAA 70 (94)
Position Control Maintained with Loss of Motive Power	Stops in position	Applicable Guidelines of NUREG-0612
Environmental Conditions and Natural Phenomena	See Diablo Canyon ISFSI FSAR Table 3.4-1	Diablo Canyon ISFSI FSAR Sections 3.2 & 3.3

[(a)] Holtec HI-STORM 100 System FSAR, Revision 1A

TABLE 3.4-5

DESIGN CRITERIA FOR CASK TRANSFER FACILITY

Design Criterion	Design Value	Reference Documents
CTF Design Codes	ASME III NF (95 Edition with 96 and 97 Addenda); NUREG-1536; RG 1.142 (ACI-349-97); RG 1.199 (ACI 349-01 Appendix B)	Diablo Canyon ISFSI FSAR Sections 3.3.4 and 4.4.5.2
Design Life	40 years	Holtec FSAR, Section 2.3
Loads and Load combinations	Various	Diablo Canyon ISFSI FSAR Section 3.3.4.2.7; ASME, Section III, Subsection NF and Appendix F
Operating Temperature Range	0-100^{o}F	DCPP FSAR Update, Section 2
Environmental Conditions and Natural Phenomena	See Diablo Canyon ISFSI FSAR Table 3.4-1	Diablo Canyon ISFSI FSAR Sections 3.2 & 3.3

TABLE 3.4-6

LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC MPC Basket Assembly HI-STORM Overpack (steel structure) HI-TRAC Transfer Cask (steel structure)	Subsection NCA	Design Specification Design Report Overpressure Protection Report Data Report Certification Stamping Nameplates.	Not required (see Note #1). Not required (see Note #1). Not required (see Note #1). Not required (see Note #3). Not required (see Notes #1, #2 and #3). Not required (see Notes #2, #3 and #4). Not required (see Note #5). <u>Note #1</u> Because the MPC, MPC Basket Assembly, HI STORM Overpack, and HI TRAC Transfer Cask are not ASME Code “N” stamped items, the Design Specifications, Design Reports, Certificates of Authorization, and Over Pressure Protection Report are not required. The HI-STORM FSAR includes the design criteria, service conditions, and load combinations for the design and operation of the HI-STORM 100 System. In addition it includes the stress analyses results that demonstrates

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
			that applicable Code stress limits are met. <u>Note #2</u> Because the MPC, MPC Basket Assembly, HI-STORM Overpack, and HI-TRAC Transfer Cask are not certified to the ASME Code (Section III), the term “Certificate Holder” is not applicable. To eliminate ambiguity, the responsibilities assigned in ASME Section III to the Certificate Holder, shall be interpreted to apply to PG&E (and by extension, to Holtec and its fabricators) if the requirement must be fulfilled. <u>Note #3</u> The fabricator (including the entity responsible for the final MPC closure weld) is not required to have an ASME-accredited QA program. The Fabricator will apply an approved QA program that meets the applicable regulatory requirements to all important-to-safety items and activities. As such, ASME Certification and Stamping is not required. The QA documentation package for each item will be in accordance with the applicable QA program. <u>Note #4</u> The ASME Section III term “Inspector” is herein defined as the Quality Assurance personnel assigned by PG&E to perform oversight (e.g., audit, inspection) of the design and manufacturing processes.

TABLE 3.4-6

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
			Note #5 In lieu of the requirements for Nameplates, items will be marked in accordance with 10 CFR 71 and 10 CFR 72 and the Holtec QA Program.
MPC basket supports and lift lugs	NB-1130	NB-1132.2(d) requires that the first connecting weld of a nonpressure-retaining structural attachment to a component shall be considered part of the component unless the weld is more than 2t from the pressure-retaining portion of the component, where t is the nominal thickness of the pressure-retaining material. NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is	The MPC basket supports (nonpressure-retaining structural attachment) and lift lugs (nonstructural attachments used exclusively for lifting an empty MPC) are welded to the inside of the pressure-retaining MPC shell, but are not designed in accordance with Subsection NB. The basket supports and associated attachment welds are designed to satisfy the stress limits of Subsection NG and the lift lugs and associated attachment welds are designed to satisfy the stress limits of Subsection NF, as a minimum. These attachments and their welds are shown by analysis to meet the respective stress limits for their service conditions. Likewise, non-structural items, such as shield plugs, spacers, etc. if used, can be attached to pressure-retaining parts in the same manner.

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
		within 2t from the pressure-retaining portion of the component.	
MPC	NB-2000	Requires materials to be supplied by an ASME Material Organization.	Materials will be procured in accordance with an approved quality assurance program.
MPC, MPC basket assembly, HI-STORM overpack, and HI-TRAC transfer cask	NB-3100 NG-3100 NF-3100	Provides requirements for determining design-loading conditions, such as pressure, temperature, and mechanical loads.	These requirements are not applicable. The HI-STORM FSAR, serving as the Design Specification, establishes the service conditions and load combinations for the storage system.
MPC	NB-3350	NB-3352.3 requires, for Category C joints, that the minimum dimensions of the welds and throat thickness shall be as shown in Figure NB-4243-1.	The MPC shell-to-baseplate weld joint design (designated Category C) does not include a reinforcing fillet weld or a bevel in the MPC baseplate, which makes it different than any of the representative configurations depicted in Figure NB-4243-1. The transverse thickness of this weld is equal to the thickness of the adjoining shell (1/2 inch). The weld is designed as a full penetration weld that receives VT and RT or UT, as well as final surface PT examinations. Because the MPC shell design thickness is considerably larger than the minimum thickness required by the Code, a reinforcing fillet weld that would intrude into the MPC cavity space is not included. Not including this fillet weld provides for a higher quality radiographic examination of the full

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
			penetration weld. From the standpoint of stress analysis, the fillet weld serves to reduce the local bending stress (secondary stress) produced by the gross structural discontinuity defined by the flat plate/shell junction. In the MPC design, the shell and baseplate thicknesses are well beyond that required to meet their respective membrane stress intensity limits.
MPC, HI-STORM overpack steel structure, HI-TRAC transfer cask steel structure	NB-4220 NF-4220	Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.	The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-transfer cask) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal. Additionally, a weld efficiency factor of 0.45 has

TABLE 3.4-6

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
		shell per NB-3352.3).	been applied to the analyses of these welds.
MPC LID TO SHELL WELD	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT alone is used, at a minimum, it will include the root and final weld layers and each approximately 3/8 inch of weld depth (3 weld layers minimum).
MPC CLOSURE RING, VENT AND DRAIN COVER PLATE WELDS	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The MPC vent and drain cover plate welds are leak tested. The closure ring provides independent redundant closure for vent and drain cover plates.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The MPC enclosure vessel is seal welded in the field following fuel assembly loading. The MPC enclosure vessel shall then be hydrostatically tested as defined in HI-STORM FSAR Chapter 9. Accessibility for leakage inspections preclude a Code compliant hydrostatic test. All MPC enclosure vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination, except the MPC lid-to-shell weld shall be verified by volumetric or multi-layer PT examination. If PT alone is used, at a minimum, it must include the root and final layers and each approximately 3/8 inch of weld depth (3 weld layers minimum). For either UT or PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
			Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. The inspection process, including findings (indications), shall be made a permanent part of the user's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The vent/drain cover plate weld is confirmed by leakage testing and liquid penetrant examination and the closure ring weld is confirmed by liquid penetrant examination. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350 for PT or NB-5332 for UT.
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. The function of MPC enclosure vessel is to contain the radioactive contents under normal, off-normal, and accident conditions. The MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Basket Assembly	NG-2000	Requires materials to be supplied by an ASME Material Organization.	Materials will be procured in accordance with an approved quality assurance program.

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC Basket Assembly	NG-4420	NG-4427(a) requires a fillet weld in any single continuous weld may be less than the specified fillet weld dimension by not more than 1/16 inch, provided that the total undersize portion of the weld does not exceed 10 percent of the length of the weld. Individual undersize weld portions shall not exceed 2 inches in length.	Modify the Code requirement (intended for core support structures) with the following text prepared to accord with the geometry and stress analysis imperatives for the fuel basket: For the longitudinal MPC basket fillet welds, the following criteria apply: 1) The specified fillet weld throat dimension must be maintained over at least 92 percent of the total weld length. All regions of undersized weld must be less than 3 inches long and separated from each other by at least 9 inches. 2) Areas of undercuts and porosity beyond that allowed by the applicable ASME Code shall not exceed 1/2 inch in weld length. The total length of undercut and porosity over any 1-foot length shall not exceed 2 inches. 3) The total weld length in which items (1) and (2) apply shall not exceed a total of 10 percent of the overall weld length. The limited access of the MPC basket panel longitudinal fillet welds makes it difficult to perform effective repairs of these welds and creates the potential for causing additional damage to the basket assembly (e.g., to the neutron absorber and its sheathing) if repairs are attempted. The acceptance criteria provided in the foregoing have been established to comport with the objectives of the basket design and preserve the margins demonstrated in the supporting stress analysis. From the structural standpoint, the weld acceptance criteria are established to ensure that any departure from the ideal, continuous fillet weld seam would not

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
			alter the primary bending stresses on which the design of the fuel baskets is predicated. Stated differently, the permitted weld discontinuities are limited in size to ensure that they remain classifiable as local stress elevators ("peak stress", F, in the ASME Code for which specific stress intensity limits do not apply).
Overpack Steel Structure	NF-2000	Requires materials to be supplied by an ASME Material Organization.	Materials will be procured in accordance with an approved quality assurance program.
Transfer Cask Steel Structure	NF-2000	Requires materials to be supplied by an ASME Material Organization.	Materials will be procured in accordance with an approved quality assurance program.
Overpack baseplate	NF-4441	Requires special examinations or requirements for welds where a primary member of thickness 1 inch or greater is loaded to transmit loads in the through thickness direction.	The margins of safety in these welds under loads experienced during lifting operations or accident conditions are quite large. The overpack baseplate welds to the inner shell, pedestal shell, and radial plates are only loaded during lifting conditions and have large safety factors during lifting.
HI-STORM overpack steel structure and HI-TRAC transfer cask steel structure	NF-3256 NF-3266	Provides requirements for welded joints.	Welds for which no structural credit is taken are identified as "Non-NF" welds in the design drawings by an "*". These non-structural welds are specified in accordance with the pre-qualified welds of AWS D1.1. These welds shall be made by welders and weld

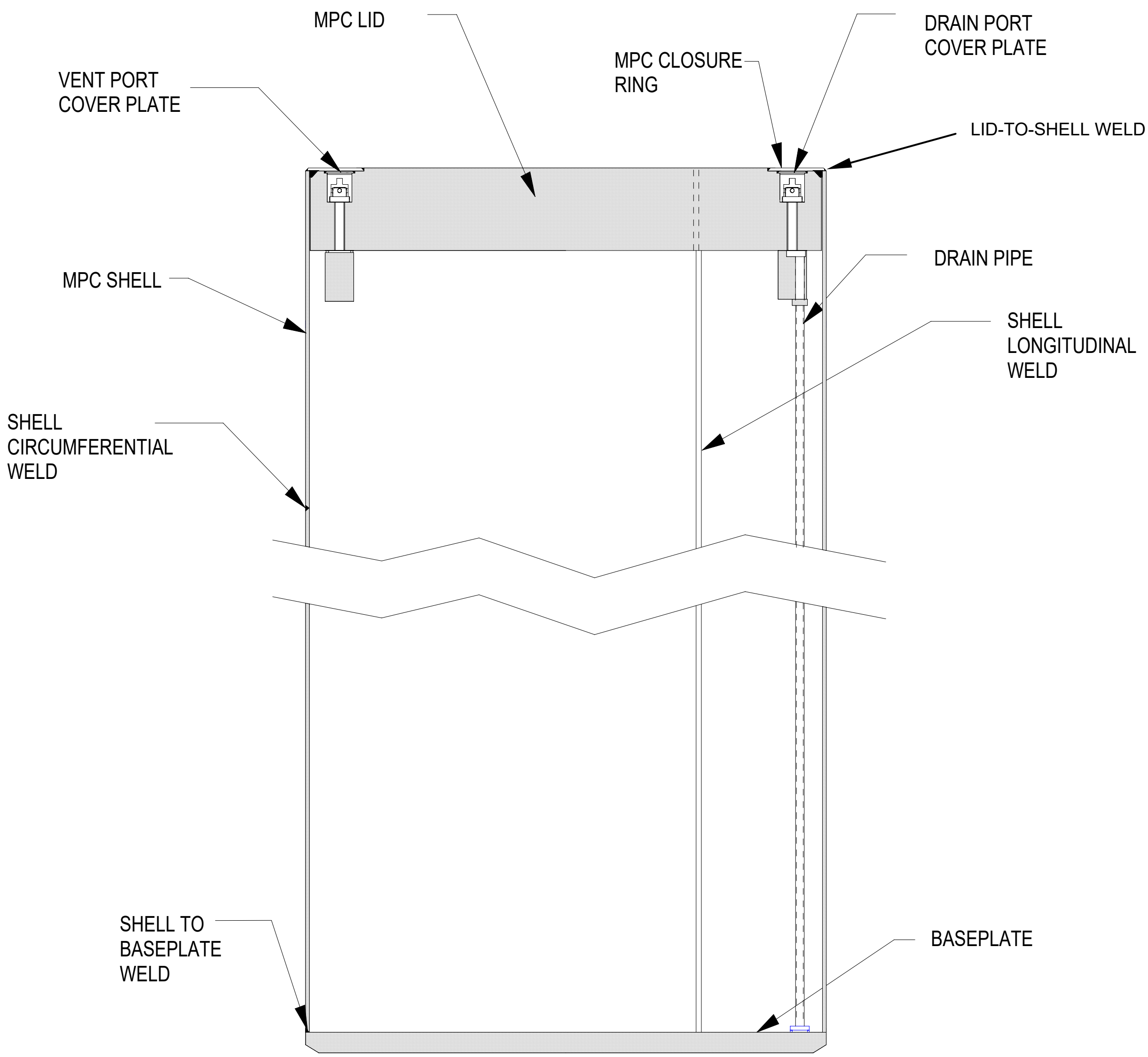
TABLE 3.4-6

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
			procedures qualified in accordance with AWS D1.1 or ASME Section IX. Welds for which structural credit is taken in the safety analyses shall meet the stress limits for NF-3256.2, but are not required to meet the joint configuration requirements specified in these Code articles. The geometry of the joint designs in the cask structures are based on the fabricability and accessibility of the joint, not generally contemplated by this Code section governing supports.
HI-STORM overpack and HI-TRAC transfer cask	NF-3320 NF-4720	NF-3324.6 and NF-4720 provide requirements for bolting.	These Code requirements are applicable to linear structures wherein bolted joints carry axial, shear, as well as rotational (torsional) loads. The overpack and transfer cask lid bolted connections in the structural load path are qualified by design based on the design loadings defined in the HI-STORM 100 FSAR. Bolted joints in these components see no shear or torsional loads under normal storage conditions. Larger clearances between bolts and holes may be necessary to ensure shear interfaces located elsewhere in the structure engage prior to the bolts experiencing shear loadings (which occur only during side impact scenarios). Bolted joints that are subject to shear loads in accident conditions are qualified by appropriate stress analysis. Larger bolt-to-hole clearances help ensure more

TABLE 3.4-6

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
			efficient operations in making these bolted connections, thereby minimizing time spent by operations personnel in a radiation area. Additionally, larger bolt-to-hole clearances allow interchangeability of the lids from one particular fabricated cask to another.

Note: Alternatives to the above table may be used when specifically authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee in accordance with 10 CFR 72.2 and as controlled by the Diablo Canyon ISFSI Technical Specifications.



FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 3.3-1 HI-STORM MPC CONFINEMENT BOUNDARY

CHAPTER 4

ISFSI DESIGN

CONTENTS

CHAPTER 4

ISFSI DESIGN

CONTENTS (Continued)

CHAPTER 4

ISFSI DESIGN

TABLES

CHAPTER 4

ISFSI DESIGN

FIGURES

CHAPTER 4

ISFSI DESIGN

FIGURES (continued)

CHAPTER 4

ISFSI DESIGN

This chapter provides descriptive design information for the ISFSI structures, systems, and components (SSCs). Emphasized are those design features that are important to safety, are covered by the quality assurance program, and are employed to withstand environmental and accident forces. The industrial codes used in the design of these SSCs are related to their design criteria and associated bases that are presented in Chapter 3.

4.1 LOCATION AND LAYOUT

The locations of the Diablo Canyon ISFSI storage site, cask transfer facility (CTF), and transport route from the DCPP fuel handling building/auxiliary building (FHB/AB) are shown in Figure 2.1-2. In addition, Figure 2.1-2 shows other facilities in the vicinity of the Diablo Canyon ISFSI storage site and the CTF, such as onsite roadways, buildings, water services, and transmission lines. None of these other facilities are related to the ISFSI. The Diablo Canyon ISFSI storage site, CTF, and the transport route are within the DCPP owner-controlled area. Travel distance from the FHB/AB via the transport route to the CTF and Diablo Canyon ISFSI storage site is approximately 1.2 miles. (See Section 4.3.3 for a discussion of the transport route.)

The storage casks are stored on concrete storage pads, which will be built as needed, within a protected area separate from that of DCPP. Each storage pad is designed to accommodate up to 20 storage casks in a 4-by-5 array as shown in Figure 4.1-1. Seven pads will be required to accommodate the fuel used during the duration of the plant's operating license period (up to 138 casks, plus 2 spare locations). Each loaded storage cask is approximately 11 ft in diameter, 20 ft high, and weighs about 360,000 lb. There is approximately 6 ft surface-to-surface distance between the casks. The seven storage pads will cover an area approximately 500 ft by 105 ft.

A security fence, with a locked gate, serves as the protected area boundary and circumscribes the storage pads. There is a minimum of 50 ft between the storage casks and the security fence on the north side of the storage pads, and a minimum of 40 ft between the storage casks and the security fence on the other three sides of the storage pads. This fence may form the restricted area fence, in compliance with 10 CFR 20, and ensures the dose rate at this boundary will be less than 10 CFR 20 requirements. Alternate restricted area boundary placement may be utilized for ISFSI operational needs, always assuring that 10 CFR 20 requirements are met. There is a second fence around the protected area that is approximately 100 ft from the storage casks. This second fence can be used as the restricted area boundary should additional distance to the storage casks be required to maintain dose rates less than 10 CFR 20 requirements.

As shown in Figure 4.1-1, the CTF is located outside the restricted area fence at about 100 ft off the northwest corner of the storage pads. The CTF is a below-ground structure where a loaded multipurpose canister (MPC) is transferred between the HI-TRAC transfer cask and the HI-STORM 100SA overpack.
The only utility associated with the Diablo Canyon ISFSI is electric power for the lights and communications for both the CTF and storage area and security equipment for the storage area. The source of this power is described in Section 4.4.4.

Loading and unloading of the MPCs takes place in the DCPP FHB/AB. These facilities are described in Chapter 9 of the DCPP Final Safety Analysis Report (FSAR) Update (Reference 1).

4.1.1 REFERENCES

1. Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update.

4.2 STORAGE SYSTEM

The design and analyses of the major components of the Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI), i.e., the cask storage pads, cask transfer facility, and the HI-STORM 100 System, are provided in this section.

4.2.1 STRUCTURES

Major important-to-safety ISFSI structures and their site locations are described in the following sections:

- Section 4.2.1.1 Cask Storage Pads
- Section 4.2.1.2 CTF
- Section 4.2.2 Site Layout
- Section 4.2.3 Storage Casks

See Figure 2.1-2 for the location of the Diablo Canyon ISFSI site in relation to the power block. See Figure 4.1-1 for the Diablo Canyon ISFSI site layout and the immediate surroundings.

4.2.1.1 Cask Storage Pads

The Diablo Canyon ISFSI storage site is designed to include seven cask storage pads in a row. Each pad will accommodate up to 20 HI-STORM 100SA storage casks. Figure 4.1-1 shows the layout of the pads with the surrounding security fence, nuisance fence, and approximate dimensions. Seven storage pads provide sufficient storage space for DCPP spent fuel through plant decommissioning. The seismic design criteria for the cask storage pads are described in Sections 3.2.3 and 3.3.2. Pad embedment design criteria are integrated with the storage cask pad design criteria, which is the primary focus of discussion in Section 3.3.2. A further discussion of the design criteria, analyses, and resulting design of the cask storage pads is provided here.

4.2.1.1.1 Function

The function of the cask storage pads is to provide a level, competent structural surface for placement of the loaded overpacks for all design-basis conditions of storage. The storage casks (overpacks) are to be anchored to the pad by 16, 2-inch diameter, SA 193 Gr. B7 studs.

4.2.1.1.2 Design Specifications

The cask storage pad design is based on a maximum, loaded-overpack weight of 360,000 lb each. This maximum weight bounds the maximum loaded weight of the overpacks used at the Diablo Canyon ISFSI. Prior to storing the MPC-24, MPC-24E, or MPC-24EF in this overpack, the MPCs would need to be modified and analyzed for compatibility with the rest of the cask handling system as was done for the MPC-32 (refer to Section 1.1). Maximum MPC weights are given in Table 4.2-1 of this FSAR and shown in Table 3.2-1 of the HI-STORM 100 System FSAR (Reference 1). See Section 3.3.2 for more details on the storage pad design criteria.

4.2.1.1.3 Plans and Sections

The site plan, which shows the locations of the concrete storage pads in relation to the power plant facility, is shown in Figure 2.1-2. A cross section of a typical concrete storage pad plan is shown in schematic Figure 4.2-1.

4.2.1.1.4 Components

- **Embedment Steel Assembly**: This assembly consists of structural steel plates and rods. The function of this assembly is to properly distribute the loads imposed on the surface (by the storage casks) to the entire structure (Figure 4.2-2).

- **Reinforced Concrete**: The steel-reinforced concrete is designed for a mix with a compressive strength of 5,000 psi at 90 days. The reinforcing steel bars have a minimum of 60,000-psi yield strength.

4.2.1.1.5 Design Bases and Safety Assurance

The cask storage pads are classified as important to safety in order to provide the appropriate level of quality assurance in the design and construction. This classification is consistent with the recommendation made in Section 2.0.4 of the HI-STORM 100 System FSAR for deployment of the anchored HI-STORM 100SA overpack at a high-seismic site. This ensures that the cask storage pads will perform their intended functions.

4.2.1.1.6 Storage Pad Design

The cask storage pads (total of seven) are structural units constructed of steel-reinforced concrete. Each concrete pad is approximately 68 ft wide by 105 ft long and 7.5 ft thick with longitudinal and transverse horizontal reinforcing bars near the top and bottom of the pads. The concrete compressive strength is 5,000 psi at 90 days. The reinforcement bars have a minimum yield strength of 60,000 psi (Reference 17).

Each pad accommodates a center-to-center spacing of 17 ft for the overpacks. Each of the cask storage pads accommodates up to 20 loaded overpacks (4 rows of 5). The sides of each storage pad are designed with an additional apron to provide maneuvering room for the cask transporter before it is driven onto the pad. The pads are nearly flush with grade to allow direct access by the cask transporter. The casks are installed on the pads in a prescribed loading sequence to assure pad stability for all design-basis accidents and to maintain design qualifications. The loading sequence is proceduralized.

The cask storage pad is designed with an embedded steel structure having a steel plate ring (Figure 4.2-1) at the surface of the concrete that mates with the bottom of the cask. Each cask is compressed against the embedment plate using 16 studs. Each stud is preloaded to approximately 157,000 lbf. The preload is achieved by threading the SA193-B7 studs into a coupling steel block located on the underside of the embedment plate, buried in the concrete. The seismic tensile/bending loads imposed on the pad are resisted by the long A-36 steel rods connected to the bottom base plate (Figure 4.2-2). The base plates are designed to provide sufficient bearing area onto the concrete so as to be able to transfer loads by bearing. Shear loads from each cask are carried through the embedment plate/coupling blocks into the concrete.

4.2.1.1.7 Storage Pad and Anchorage Analysis

The pad structural seismic analysis is performed by developing a finite-element model, using the ANSYS FEA Program (Reference 3) of a representative pad, which includes the casks and the supporting rock, to determine the potential for pad uplift and to calculate the stress fields in the concrete. The results of this static analysis are used in the design of the reinforcements to ensure that the bending moments are adequately carried by the pad, and that the stress limits set forth in ACI 349-97 and endorsed by Regulatory Guide 1.142 (Reference 34) are satisfied. Regulatory Guide 1.199 (Reference 35) endorses Appendix B to ACI 349-01 (Reference 22), and anchoring provisions of the pad were designed in accordance with this guide. The methodology used assumes the loading imposed on the pad embedment structures is similar to an inverted column. Specifically, the design-strength capacity of the embedded base plate, concrete bearing, and diagonal tension-shear capacity, computed in accordance with the NRC provisions stated in Regulatory Guide 1.199, all exceed the required ductile design strength of the embedded anchor stud. Furthermore, the ultimate tensile strength of the reduced section at the thread root of the anchor bar is approximately 125 percent of the yield strength of the unreduced gross section of the anchor bar. Anchor bars are made of A36 steel, which has a well-defined yield plateau. Thus, if any overload occurs, the anchor bars will yield before any less ductile failure could occur.

Lastly, the yield strength of the embedded anchor studs is more than 250 percent of the computed demand load on these bars to provide substantial margin against yielding. Reference 15 contains design and analysis information pertaining to the embedment support structure.

Supporting evidence that the concrete will not break out prior to failure of ductile metal members is provided in Reference 26. The tension tie-rods are not treated like anchors for the reasons stated above in the design philosophy. They are treated as inverted columns on base plates and are sized to have lower ultimate strength than the surrounding concrete strength in bearing and diagonal shear as set by Regulatory Guide 1.142. As such, the design ensures ductile behavior, which meets the intent of Regulatory Guide 1.199. Reference 15 provides various capacity calculations for different elements in the load path, thus providing the required evidence as stated above. Furthermore the design has substantial margin between the yield capacity of the weakest element (tension tie-rods) and the imposed tension pull-out demand load.

The load path for delivery of shear load into concrete is through the coupler at the top of the tension tie-rods. As such, the tension tie-rods are not relied on to deliver any shear load into concrete.

Lastly, the combination of the tension (pad flexure) and shear (pull-out) loading in the reinforcing steel and the minimum required steel area has been demonstrated and shown to have considerable margin.

The pad was evaluated for sliding. Section 8.2.1.2.3.2 describes the dynamic non-linear time history analysis that was performed to evaluate pad sliding. Overturning is not considered as a credible failure, considering the overall geometry of the structure.

4.2.1.1.7.1 Pad Static Analysis

A solid finite element model of the pad was developed (using the ANSYS FEA Program) to statically analyze the pad for loads imposed by the casks, as well as the pad-inertia loads, due to ZPA excitation from postulated bounding ground motions (Section 8.2.1.2.3.2). The static loading cases were performed for a range of ground/rock moduli of elasticity to account for variations in the rock properties. The earthquake loadings bound the other accidental loading conditions (for example, explosion and tower collapse) and natural phenomena accident conditions (for example, tornado and wind).

4.2.1.1.7.2 Cask Dynamic Analyses

The storage cask is analyzed by a nonlinear, time history analysis for bounding ground motions. The resulting anchorage loading at the concrete/embedment interface is used for the detailed analysis of the pad and the embedment steel (see Section 4.2.1.1.7.1 for a discussion of the pad static analysis). The cask dynamic analysis is explained further in Section 8.2.1.2.3.1.

4.2.1.1.8 Storage Pad Settlement

No pad settlement is anticipated as a result of the facility placement on the rock site (See Section 2.6.4.4 for more discussion).

4.2.1.1.9 Slope Stabilization Measures

The following sections discuss slope stabilization and rock fall mitigation measures being taken to ensure the storage casks are not adversely affected by debris flow and rock falls.

4.2.1.1.9.1 Cut Slope, Stabilization Design

As discussed in Sections 2.6.5.2.1 and 2.6.5.2.2, rock blocks exposed after cut-slope excavation have the potential to fall into the excavation under both static and seismic loading conditions. After excavation, cut-slope faces will be protected from weathering and minor raveling by a wire-mesh-reinforced shotcrete facing to stabilize the cut slope and prevent or minimize potential failures from occurring. To stabilize larger rock blocks, potentially prone to failure during seismic loading, rock anchors (Reference 18) will be installed in approximately 2- to 3-inch diameter holes on approximately 5-ft centers and drilled subhorizontally approximately 30 ft deep from the cut-slope faces (Figure 4.2-3). As shown in Figure 4.2-3, there will be a square shotcrete buildup formed around each anchor hole to distribute anchor loads to the rock surface. High-strength, corrosion-protected bar anchors will be inserted into the holes, grouted and stressed. Each bar will be installed and proof-tested as recommended by the Post Tensioning Institute (PTI). Additional holes, one approximately every fifth anchor, will be drilled between anchor holes and lined with PVC drainpipe to ensure the slope remains free for draining. The actual pattern will be adjusted during construction, based on the conditions found. All pattern adjustments must result in a dynamic safety factor of greater than or equal to 1.3.

4.2.1.1.9.2 Mitigation of Potential Displacements along Clay Beds

As discussed in Section 2.6.5.1.3, potential rock mass displacements along clay beds due to seismic ground motions are calculated to range from 1 to 3 ft on the clay beds located on the natural slope above the ISFSI site, and 1 ft to 2 ft on the clay beds inferred to daylight in the cut slope or pass just below the ISFSI site. The effects of these potential displacements are mitigated, as described below.

Rocks dislodged by displacements along any of the several clay beds on the natural slope above the ISFSI site are prevented from reaching the ISFSI site by a rockfall barrier constructed at the top of the ISFSI cut slope. This barrier is designed to absorb and dissipate the kinetic energy of the rockfall and is constructed of articulated steel posts, bundled wire ring steel nets, friction brake elements, anchoring and retaining ropes, and rock anchors.

The rockfall barrier fence constructed at the top of the ISFSI cutslope is a commercially available rockfall fence system specifically designed for the possible site loading conditions. PG&E's rockfall analysis suggests that such a commercially available high impact fence (design load of 295 ft-tons) is suitable for the ISFSI. The fence height of approximately 8 ft provides a substantial margin of safety against all possible block sizes and forces.

Rocks offset by displacements along clay beds daylighting in the cut slope will be prevented from dislodging from the cut slope face by the wire-mesh-reinforced shotcrete facing and rock anchor system described in Section 4.2.1.1.9. The orientation of clay beds in the region of the cut slope is approximately parallel to the preferred rock anchor orientation, thereby minimizing the potential for damage to the anchors as a result of displacements along the clay beds. In the unlikely event that rock blocks are completely dislodged from the cut-slope face during a seismic event, the midslope bench width and offset distance from the slope base to the ISFSI pads are sufficient to accommodate the largest rock blocks as defined in Section 2.6.5.2.2.

In the event displacements occur along clay beds inferred to pass beneath the site, it is expected that any displacements propagating upward will do so through the weaker rock surrounding the massive, heavily reinforced concrete pads, and not impose significant additional loads or displacements on the pads themselves.

The design basis criteria and analysis of potential slope instability mitigation features are further described and discussed in References 27 and 28.

4.2.1.2 CTF Support Structure

The CTF concrete support structure is a cylindrical, steel-lined structure, embedded in the rock, underground and made-up of steel-reinforced slabs and walls (Reference 38). The facility is designed with a sump for incidental water collection. When not in use, the facility is enclosed with a cover for personnel safety and protection of the structure from the environment.

The cask transporter lateral restraint system is designed to apply external restraint loading to the cask transporter structure. As discussed in References 21 and 29, the restraints are steel struts or similar equipment suitably sized to restrain the transporter by transferring the restraint loading to the ground adjacent to the CTF support structure. The transporter tie down locations immediately adjacent to the CTF support structure are shown in PG&E drawing 6021750, Sheet 312 (Reference 39). The tie downs are supported by rock anchor installations into the ground. Holtec Drawing 4480, showing the CTF shell structure, is provided in Figure 4.4-3.

The CTF structure is fully embedded in the ground. The top of the structure has a one inch high lip above grade and around the top of the CTF shell to prevent entry of liquids into the CTF pit. The bottom of the concrete base slab is approximately 20 ft below the surface of the adjacent competent rock (see PG&E drawing 6021750, Sheet 310 (Reference 38)). Once the base slab is poured, the main shell steel structure is placed, plumbed and anchored to the base slab. Concrete is placed between the exterior surface of the main shell and the surrounding competent rock. Following concrete placement, the main shell remains embedded in the concrete.

The concrete portion of the facility is designed to transfer all loads to the rock in direct bearing of the concrete on the rock. The analysis demonstrates that all stresses in the concrete and the rock remain less than the allowable limits under all design conditions. Therefore, it is not necessary to anchor the concrete structure to the rock.

The design of the CTF is described in References 24 and 25, which demonstrate that the concrete structure is capable of resisting all applied loads and adequately transferring these loads to the surrounding rock. This includes all applicable loads from the transporter, the CTF structure and the fully loaded cask. References 24 and 25 consider all operating loads in addition to other applicable loads including seismic. Removable seismic restraints in the form of wedge assemblies at the top and bottom of the CTF shell provide lateral support in the gap between the overpack and the CTF main shell (Reference 25 and Figure 4.4-3).

Holtec Calculation HI-2053370 (Reference 25) demonstrates the feasibility of the CTF conceptual design by modeling major components and developing the loads transmitted to the concrete support structure. In addition to the information provided in HI-2053370, Drawing 4480 (Figure 4.4-3) provides materials of construction and major dimensional information for the CTF.

Table 3.4-5 specifies that ASME Section III, Subsection NF, Appendix F, ACI-349-97, and Appendix B to ACI 349-01, as endorsed by Regulatory Guides 1.142 and 1.199, respectively are the governing codes for the design of the CTF. These codes provide requirements for design, materials, welding, inspection, brittle fracture testing, etc., which are reflected in the final design and procurement documents. Fabrication, assembly, and test procedures are developed in accordance with the design criteria and specifications, drawings, and applicable codes after final design is complete. In addition, the CTF installation is in accordance with ACI 439-01.

For added documentation, PG&E submitted the Holtec-proprietary design criteria document for the CTF (HI-2002570) to the NRC (Reference 23), which provides additional detail on codes and standards, as well as performance requirements. The aforementioned documents provide the complete set of information available regarding the design of the CTF.

4.2.1.2.1 Function

The function of the CTF support structure is to provide a flat, concrete pad at the bottom of the facility to accommodate installation of the CTF steel shell and to provide a rigid, concrete pad on the surface for the cask transporter.

4.2.1.2.2 Design Specifications

The structure has provisions for a sump and sump pump to allow for removal of incidental rainwater. The CTF and its supporting structure are qualified to withstand the design earthquake (DE), double-design earthquake (DDE), Hosgri earthquake (HE), and LTSP earthquakes. The earthquake loading bounds the other accidental loading conditions (for example, tower collapse) and natural phenomena accident conditions (for example, tornado and wind). See Section 3.3.4 for a discussion of the CTF design criteria.

4.2.1.2.3 Static Analysis

The reinforced concrete was designed and evaluated for a transporter on top of the facility and the overpack in the CTF during the MPC transfer operation. The structure is designed for appropriate vertical and lateral loads imposed during the DE, DDE, HE and LTSP earthquakes. The concrete and the reinforcing steel have been designed in accordance with the NRC positions stated in Regulatory Guide 1.142, which endorses ACI 349-97 (Reference 4). A static, seismic analysis was performed on the CTF shell (Section 8.2.1).

4.2.1.2.4 CTF Structure Layout

The structure is located on the ISFSI site approximately 100 ft from the concrete storage pads (Figure 4.1-1).

4.2.2 SITE LAYOUT

A plan view of the ISFSI storage site layout is shown in Figure 4.1-1. This figure shows the functional features of the storage site, including the locations of the CTF, the security and nuisance fences, and the access road that leads up from the DCPP. A section view of the ISFSI storage site is shown in Figure 4.2-5. This figure provides separation distances from the pad to nearby features, including the cut-slope hillside to the south and east of the pad.

As shown in Figures 4.1-1 and 4.2-5, a removable fence is located between the security fence and the raw water reservoirs. This fence provides protection against false security alarms due to authorized personnel, who are working in the raw-water-reservoir area, inadvertently stepping into an alarmed zone. If work activities in the raw-water-reservoir area require the fence to be temporarily removed, it can be with the appropriate, accompanying security compensating measures.

4.2.3 STORAGE CASK DESCRIPTION

The HI-STORM 100 System is used to store spent fuel and associated nonfuel hardware in a dry configuration at the Diablo Canyon ISFSI storage site. At Diablo Canyon, the shortened and anchored version of the standard HI-STORM 100 System overpack will be used. This system is referred to as the HI-STORM 100SA and has been certified by the NRC in Amendment 1 to the HI-STORM Certificate of Compliance (CoC) 1041-1 for general use at applicable onsite ISFSIs operated by a 10 CFR 50 license holder. Holtec Drawing 4461, sheet 14 showing the HI-STORM 100SA overpack-to-ISFSI pad (anchor stud/sector lug) arrangement is provided in Figure 4.2-7.

4.2.3.1 Function

As discussed in Section 3.2, the HI-STORM 100 System is designed to store spent nuclear fuel and associated nonfuel hardware from DCPP under Diablo Canyon ISFSI site-specific normal, off-normal, and accident conditions of service, including the most severe design-basis natural phenomena in accordance with 10 CFR 72 (Reference 5). The HI-STORM 100 System design is summarized in Chapter 1 of this FSAR and described in more detail in Chapters 1 and 2 of the HI-STORM 100 System FSAR.

The HI-STORM 100 System is designed to permit testing, inspection, and maintenance of the systems. The acceptance test and maintenance programs of the HI-STORM 100 System are specified in Chapter 9 of the HI-STORM 100 System FSAR. Because of the passive nature of the HI-STORM 100 System, onsite inspection and maintenance requirements are minimal. Surveillance requirements associated with operational control and limits are described in Chapter 10. Inspection and testing of important-to-safety components are performed in accordance with the Holtec International or PG&E Quality Assurance Program, as applicable.

Each of the HI-STORM 100 System components is described in further detail in the following sections. Figures, or reference to figures, in the HI-STORM 100 System FSAR are provided to illustrate the components and their functions.

4.2.3.2 Description

In its final storage configuration, the HI-STORM 100 System consists of the following major components considered important to safety:

- Holtec multi-purpose canister
- Holtec damaged fuel container (DFC)
- HI-STORM 100SA overpack

The following sections provide a summary of the HI-STORM 100 System MPC, DFC, and overpack design bases and design relative to the storage requirements of the Diablo Canyon ISFSI. The Diablo Canyon onsite transporter is described in Section 4.3. Detailed operating guidance for MPC loading, onsite transport, and transfer of the MPC from the transfer cask to the HI-STORM overpack is provided in Sections 5.1 and 10.2 of this FSAR. Design drawings for generic HI-STORM 100 System components, except the DFC, are contained in Section 1.5 of the HI-STORM 100 System FSAR. A figure depicting the DFC is contained in Section 2.1 of the HI-STORM 100 System FSAR.

The HI-TRAC 125 transfer cask is used to provide the necessary structural support, shielding, heat removal, and missile protection as well as the means to transfer the loaded MPC between the transfer cask and the HI-STORM 100SA overpack. The transfer cask is not used in the final storage configuration of the HI-STORM 100 System at the storage pads. Design drawings for a standard transfer cask are provided in Section 1.5 of the HI-STORM 100 System FSAR.

4.2.3.2.1 MPC

The MPC provides for confinement of radioactive materials, criticality control, and the means to dissipate decay heat from the stored fuel. It has the structural capability to withstand the loads created by all design basis accidents and natural phenomena. The MPC is a totally welded structure of cylindrical profile with flat ends. It consists of a honeycomb fuel basket, baseplate, MPC shell, MPC lid, vent and drain port cover plates, and closure ring. The MPCs, with different internal arrangements, can accommodate intact spent fuel, damaged fuel, fuel debris, and nonfuel core components, as discussed in Sections 3.1 and 10.2. The MPC lid provides top shielding and provisions for lifting the loaded MPC during transfer operations between the transfer cask and the overpack. The MPC fuel-basket assembly provides support for the fuel assemblies as well as the geometry and fixed neutron absorbers for criticality control. The MPC is made entirely of stainless steel, except for the neutron absorbers, and an aluminum seal washer or port plug with thread protector in both the vent and drain port assemblies. An alternative vent and drain port plug configuration may be used, which does not contain aluminum washers. A summary of the nominal physical characteristics of the MPC is provided in Table 4.2-1. Figure 4.2-13 shows the MPC enclosure vessel assembly and MPC-32 fuel basket supports.

4.2.3.2.2 DFC

The DFC is used to contain fuel assemblies classified as damaged fuel or fuel debris as required by the Diablo Canyon ISFSI TS and Section 10.2. Damaged fuel may be stored in both the MPC-24E and the MPC-24EF, however, storage of fuel debris is only allowed in the MPC-24EF. Storage of damaged fuel or fuel debris is not permitted in the MPC-24 or the MPC-32. The HI-STORM 100SA overpack currently used at the Diablo Canyon ISFSI can contain one MPC-32. Storing the MPC-24, MPC-24E or MPC-24EF would require modifications and analyses similar to those done for the MPC-32 (refer to Section 1.1). The DFC is a long, square, stainless-steel container with screened openings at the top and bottom. Each DFC is inserted into a designated storage cell within the MPC. The function of each DFC is to retain the damaged fuel or fuel debris in its storage cell and provide the means for ready retrievability. The DFC permits gaseous and liquid media to escape into the interior of the MPC, but minimizes dispersal of gross particulates during all design basis conditions of storage, including accident conditions. The total quantity of fuel debris permitted in a single DFC is limited to the equivalent weight and special nuclear material quantity of one intact fuel assembly. HI-STORM 100 System FSAR Figure 2.1.2B shows the general arrangement of the MPC-24E/EF DFC.

The lifting device at the top of the DFC is designed to meet the requirements of ANSI N14.6 (Reference 6) in accordance with applicable guidelines of NUREG-0612 (Reference 7). As discussed in the Holtec LAR 1014-1, Appendix 3 the DFC is designed to meet ASME Section III, Subsection NG (Reference 8) allowables for normal handling and ASME Section III, Appendix F allowables for loadings experienced during a postulated, cask-drop accident.

4.2.3.2.3 HI-STORM 100SA Overpack

The HI-STORM overpack is a rugged, heavy-walled, cylindrical, steel and concrete structure. The structure is made of inner and outer concentric carbon-steel shells, a baseplate, and a bolted lid (fabricated as a steel-encased concrete disc). The bottom baseplate diameter is increased with gusseted weldments to provide a bolt circle with 16 holes for anchor studs to fasten the overpack to its ISFSI pad anchorage embedment. Either field-installed shims or a permanent circumferential shim plate weldment is used to ensure the proper pre-load is obtained in each anchor stud. The spacing of the carbon-steel inner and outer shells provides approximately 30 inches of annular space that is filled with unreinforced concrete for radiation shielding. The overpack is designed to permit natural circulation of air around and up the exterior shell of the MPC, via the chimney effect, to provide for the passive cooling of the spent fuel contained in the MPC. The cask has 4 air inlet ducts located at 90-degree spacing in the base of the cask and 4 air outlet ducts located in the top lid of the overpack. The cooling air enters the inlet ducts, absorbs heat from the MPC surface, and flows upward in the annulus between the MPC and exits at the outlet ducts.

A summary of the nominal physical characteristics of the overpack is provided in Table 4.2-2. Figure 4.2-7 shows the HI-STORM overpack assembly.

4.2.3.2.4 HI-TRAC 125 Transfer Cask

The transfer cask is used to facilitate transport of the loaded MPC from the FHB/AB to the CTF and transfer of the loaded MPC into the overpack for storage at the ISFSI storage pad. It provides the necessary structural, shielding, and heat removal design features to protect the spent fuel and personnel during fuel loading, MPC preparation, and MPC transfer operations. The transfer cask is a rugged, heavy-walled, cylindrical steel vessel comprised of inner and outer concentric shells, a bolted bottom lid, a top lid, and an outer circumferential water jacket. The annulus between the inner and outer steel shells is filled with lead. The water jacket is filled with water for shielding before movement of the transfer cask to the SFP for fuel loading. The lead and the water in the jacket provide gamma and neutron shielding for personnel working on or near the loaded MPC to ensure occupational exposures are as low as is reasonably achievable (ALARA) during operations. The transfer cask is designed for transient use, to contain the MPC, and to be submerged in the SFP to support fuel loading. It includes lifting trunnions to allow the loaded transfer cask and MPC to be placed into and removed from the SFP for decontamination and preparation of the MPC for storage. The maximum design weight of the transfer cask is 125 tons, including a fully loaded MPC-32 with water in the MPC cavity and water in the water jacket. Additional physical characteristics of the transfer cask are provided in Table 4.2-3. Figure 4.2-8 shows the HI-TRAC transfer cask assembly. Further detailed descriptions, design drawings, and a summary of the design criteria for the transfer cask are provided in Sections 1.2.1.2.2, 1.5, and 2.0.3, respectively, of the HI-STORM 100 System FSAR.

An optional design of the HI TRAC 125 transfer cask was developed by Holtec International and was implemented under the provision of 10 CFR 72.48 for generic use with the HI-STORM 100 System. Holtec proprietary Drawing 3438 was provided to the NRC under separate cover (see Reference 9). A non-proprietary drawing was included in Revision 1 to the HI-STORM 100 System FSAR. This optional HI-TRAC 125D design was further modified for use at the DC ISFSI. Figure 4.2-8 shows the Diablo Canyon HI-TRAC transfer cask. The HI-TRAC design changes made under 10 CFR 72.48 are summarized below.

The key differences between the DC ISFSI modified HI-TRAC 125D and the generic HI-TRAC 125 design described above are as follows:

(1) The HI-TRAC 125D design has been shortened by 9 inches to allow continual vertical orientation throughout the loading and transport operations.

(2) The lower pocket trunnions have been removed as they are not needed to accommodate the Diablo Canyon ISFSI lifting and handling operations.

(3) The use of the HI-STORM mating device eliminates requiring the replacement of the HI-TRAC bottom lid with a separate transfer lid while in the FHB/AB, thus reducing personnel dose. This mating device design allows for the removal of the bottom lid to facilitate MPC transfer at the CTF. The transfer cask is only handled in a vertical configuration while in the FHB/AB and while traveling to the CTF where it is attached to the mating device. In that configuration, the bottom lid provides adequate shielding with no additional shielding required.

(4) The bottom baseplate diameter has been increased and an additional bolt circle added with 24 holes to accommodate the HI-TRAC bottom lid, and to add additional strength to the HI-STORM mating device connection. Gussets have also been added to the baseplate to provide additional strength.

(5) The water jacket design has been changed from channel-and-plate design to a rib-and-shell design to better facilitate fabrication and reduce the number of welded joints.

(6) The bottom lid and drain line have been slightly modified to improve the quality of the bolted joint and improve the operability of the drain.

4.2.3.3 Design Bases and Safety Assurance

The governing codes used for the design and construction of the HI-STORM 100 System steel components are listed in HI-STORM 100 System FSAR, Table 2.2.6, and are summarized below. Clarifications on the applicability of ACI 349-85 (Reference 10) to the unreinforced concrete used in the HI-STORM 100 overpack are provided in Appendix 1.D to the HI-STORM 100 System FSAR (Reference 41). Table 3.4-6 provides a list of ASME Code alternatives for the HI-STORM System.

•	MPC	
	Pressure boundary	ASME Code Section III, Subsection NB
	Fuel Basket	ASME Code Section III, Subsection NG
•	DFC	
	Lifting Bolts	ANSI N14.6 per applicable guidelines of NUREG-0612, Section 5.1.6
	Steel Structure	ASME Code Section III, Subsection NG
•	Overpack	
	Steel	ASME Code Section III, Subsection NF
	Unreinforced Concrete	ACI-349-85, NUREG-1536
•	Transfer Cask	
	Steel Structure	ASME Code Section III, Subsection NF
	Lifting Trunnion Blocks	ASME Code Section III, Subsection NF and ANSI N14.6 per applicable guidelines of NUREG-0612, Section 5.1.6
	Lifting Trunnions	ANSI N14.6 per applicable guidelines of NUREG-0612, Section 5.1.6

The safety classification of the components comprising the HI-STORM 100 System was determined using NUREG/CR-6407 (Reference 11) as a guide. Section 4.5 provides the safety classification of the HI-STORM 100 System components and additional detail on safety classification of components used at the Diablo Canyon ISFSI.

4.2.3.3.1 System Layout

In its storage configuration, the HI-STORM 100 System consists of a fully-welded MPC placed inside of a vertical concrete overpack. Each MPC holds either 24 or 32 PWR spent fuel assemblies in an internal basket, depending on the particular MPC model. The specifics of the material approved for storage in the HI-STORM 100 System at the Diablo Canyon ISFSI storage site are discussed in Sections 3.1.1 and 10.2 and the Diablo Canyon ISFSI TS.

The HI-STORM 100 System overpack, MPC shell, and MPC basket are illustrated in Figures 4.2-7, 4.2-13, and 4.2-14, respectively. Cross-sections of the PWR MPC baskets and an outline of the DFC are shown in the figures contained in Sections 1.2 and 2.1, respectively, of the HI-STORM 100 System FSAR.

The transfer cask is designed for repetitive, transient use to contain one MPC during fuel loading, MPC preparation for storage, and transfer of the sealed MPC to the CTF. The transfer cask provides necessary shielding, heat removal, and structural integrity during the short time it contains the loaded MPC. The transfer cask is shown in Figure 4.2-8.

4.2.3.3.2 Structural Design

The structural evaluation for the HI-STORM 100 System is contained in HI-STORM 100 System FSAR Chapter 3 and in the accident analyses in Chapter 8 of this FSAR. Structural evaluations and analyses of the HI-STORM 100 System components have been performed for all design basis normal, off-normal, and accident conditions and for design basis natural phenomena conditions in accordance with 10 CFR 72, Subpart L. The structural evaluations confirm that the structural integrity of the HI-STORM 100 System is maintained under all design-basis loads with a high level of assurance to support the conclusion that the confinement, criticality control, radiation shielding, and retrievability criteria are met.

The following discussion verifies that the Diablo Canyon ISFSI site-specific criteria are enveloped by the HI-STORM 100 System design.

4.2.3.3.2.1 Dead and Live Loads

Dead loads are addressed in the HI-STORM 100 System FSAR, Section 2.2.1.1. The dead load of the overpack includes the weight of the concrete and steel cask and the MPC loaded with spent fuel. As identified in HI-STORM 100 System FSAR Table 2.1.6, the dead load of the overpack with the loaded MPC is calculated assuming the heaviest PWR assembly (B&W 15-by-15 fuel assembly type, wt = 1,680 lb, including nonfuel hardware) that bounds the Diablo Canyon fuel dead load (1,621 lb). The stresses calculated for the dead loads of the MPC and the overpack are shown to be within applicable Code allowables and, therefore, meet the Diablo Canyon ISFSI design criteria in Section 3.2.5.

The overpack is designed for two live loads, both of which act on the top of the overpack: (a) snow loads, and (b) the mating-device and transfer cask weight (during transfer operations) containing a fully loaded MPC. The HI-STORM 100 System FSAR uses a conservative, worst-case ground snow load of 100 lb/ft^2 as shown in HI-STORM 100 System FSAR Table 2.2.8, which exceeds any anticipated Diablo Canyon ISFSI site snow load. The live load capacity of the overpack is shown in HI-STORM 100 System FSAR, Section 3.4.4.3.2.1, to exceed the live load imposed by the loaded transfer cask. Since the live loads used in the HI-STORM 100 System generic analysis meet or exceed those that would be expected at the Diablo Canyon ISFSI, the HI-STORM 100 System FSAR analysis bounds the Diablo Canyon ISFSI design criteria specified for live loads in Section 3.2.

As described above, the transfer cask dead load includes the weight of the cask plus the heaviest loaded MPC. The stresses calculated for the dead loads of the MPC and the transfer cask are shown to be within applicable Code allowables and, therefore, meet the Diablo Canyon ISFSI design criteria in Section 3.2.5 for dead loads.

4.2.3.3.2.2 Internal and External Pressure

Internal and external pressure loads are addressed in the HI-STORM 100 System FSAR, Sections 3.4.4.3.1.2 and 3.4.4.3.1.7, respectively. The normal and off-normal condition design pressures for the MPC are 100 psig for internal pressure and 0 psig (ambient) for external pressure as shown in Table 2.2.1 of the HI-STORM 100 System FSAR. For accident conditions, the design pressure for the MPC is 200 psig for internal pressure and 60 psig for external pressure. Table 4.2-4 provides the maximum calculated MPC pressures for two normal conditions; no fuel rods ruptured and 1 percent fuel rods ruptured, as calculated in Reference 36. The resultant pressure for the 10 percent rods rupture off-normal condition is provided in Section 8.1.1 and is below the 100 psig design pressure. The calculations assumed design basis heat load and bounding maximum fuel rod off-gas and internal pressure for DCPP fuel, considering a site-specific bounding value for fuel rod internal pressure. The internal pressure calculations for the MPC-32 bound those for the MPC-24, MPC-24E, and MPC-24EF because there is less free volume and more fuel inside the MPC-32 cavity, which creates higher pressures for the scenarios analyzed.

The MPCs loaded up through 2013 were backfilled with helium during fuel loading operations to a nominal pressure of 31.3 psig (maximum 33.3 psig) at a reference temperature of 70°F. Future MPCs with maximum heat load up to 28.74 kW are backfilled with helium during the loading operation to a normal pressure of ≥ 34 psig and ≤ 40 psig at a reference temperature of 70°F. The internal pressure rises in proportion to the rise in MPC cavity gas absolute temperature due to the decay heat emitted by the stored fuel and as temperatures equilibrate to those associated with the normal conditions day/night annual average ambient temperatures evaluated in the thermal analysis. This normal condition ambient temperature is higher than, and is therefore bounding for, the average day/night ambient temperature at the Diablo Canyon ISFSI site (Reference 12, Section 1.2.1.3).

MPC internal pressures were also evaluated for postulated accident conditions, including 100 percent fuel rod cladding rupture, assuming all rod fill gas and a conservative fraction of fission product gases are released from the failed rods into the MPC. The resultant pressure from the 100 percent fuel rod rupture is provided in Section 8.2.14 and is below the MPC accident design pressure of 200 psig.

The stresses resulting from the internal and external pressure loads were shown to be within Code allowables. The Diablo Canyon ISFSI TS and Section 10.2 ensure that the characteristics of the DCPP fuel to be loaded in a HI-STORM 100 System are consistent with the bounding fuel limits for array/class 17x17A and 17x17B fuel assemblies in Appendix B to the HI-STORM 100 System CoC (Reference 13). The pressure evaluations have appropriately accounted for the gas volume produced by burnable poison rod assemblies and integral fuel burnable absorber (IFBA) rods.

4.2.3.3.2.3 Thermal Expansion

Thermal expansion-induced mechanical stresses due to non-uniform temperature distribution are identified in Section 3.4.4.2 of the HI-STORM 100 System FSAR. There is adequate space (gap) between the MPC basket and shell, and between the MPC shell and overpack or transfer cask, to ensure there will be no interference during conditions of thermally induced expansion or contraction. This was confirmed for the Diablo Canyon ISFSI specific MPC-32 derivative, when updating to a three-dimensional (3-D) computational fluid dynamics (CFD) model in support of LA 2 (Reference 40).

Table 4.4.15 of the HI-STORM 100 System FSAR provides a summary of HI-STORM 100 System component temperature inputs for the structural evaluation, consisting of temperature differences in the basket periphery and MPC shell between the top and bottom portions of the HI-STORM PWR MPC (MPC-24, MPC-24E, MPC-24EF, and MPC-32). The temperature gradients were used to calculate resultant thermal stresses in the MPC that were included in the load combination analysis. The stresses resulting from the temperature gradients were shown to be within Code allowables. Section 3.4.4.2 of the HI-STORM 100 System FSAR provides a discussion of the analysis and results of the differential thermal expansion evaluation. The Diablo Canyon ISFSI TS and Section 10.2 ensure that the characteristics of the DCPP fuel to be loaded in a HI-STORM 100 System meet the limits delineated in Section 3.1.1. These limits are consistent with the bounding fuel limits for array/class 17-by-17A and 17-by-17B fuel assemblies in Appendix B to the HI-STORM 100 System CoC. Therefore, the thermal expansion evaluation, discussed above, in the HI-STORM 100 System FSAR bounds the conditions at the Diablo Canyon ISFSI.

4.2.3.3.2.4 Handling Loads

Handling loads for normal and off-normal conditions are addressed in the HI-STORM 100 System FSAR, Sections 2.2.1.2, 2.2.3.1, and 3.1.2.1.1.2. The normal handling loads that were applied included vertical lifting and transfer of the overpack with a loaded MPC through all movements. The MPC and overpack were designed to withstand loads resulting from off-normal handling assumed to be the result of a vertical drop. In the case of Diablo Canyon, however, the vertical drop during onsite transport, outside the FHB/AB, is precluded with the use of a cask transporter that is designed, fabricated, inspected, maintained, and tested in accordance with the applicable guidelines of NUREG-0612. Likewise, drops are precluded while the cask is lifted at the CTF since the transporter lifting mechanism is designed, fabricated, inspected, operated, maintained, and tested in accordance with NUREG-0612. This approach is consistent with the provisions in the HI-STORM 100 System CoC described in Section 4.2.3.3.2.5 below. The preclusion of drop events was chosen as a design strategy to accommodate the anchored HI-STORM 100SA overpack, which requires a robust pad to ensure that the anchor studs and embedment structure remain fixed during postulated earthquake and tornado events.

The transfer cask is designed to withstand the loads experienced during routine handling, including lifting and transfer to the CTF with a loaded MPC. Loads were increased by 15 percent in the analyses to account for dynamic effects from lifting operations (hoist load factor). The lifting trunnions, trunnion blocks, and load-bearing connection points (that is, bottom lid bolted connections) were analyzed for normal handling loads, as described in Section 3.4.3.7 of the HI-STORM 100 System FSAR.

4.2.3.3.2.5 Overpack/Transfer Cask Tipover and Drop

Outside the FHB/AB, tipover of a loaded overpack is a noncredible accident since the HI-STORM 100SA used at the Diablo Canyon ISFSI storage site is anchored to the ISFSI pad. When not on the ISFSI pad, the overpack will be either in the CTF or attached to the cask transporter (as described in Chapter 5). The cask transporter is designed to preclude cask drops. The anchored HI-STORM 100SA overpack has been designed to withstand the worst-case, design-basis, seismic ground motion without failure of the anchor studs or the embedment. In addition, the anchored overpack has been analyzed for site-specific: (a) explosions, and (b) tornado wind concurrent with the impulse force of a large, design-basis, tornado-borne missile to verify that the anchorage design can resist the resultant overturning moment (Reference 19). The design criteria for the concrete storage pad and cask anchors are described in Section 3.3.2. The design and analysis of the concrete storage pad and anchorage embedment are discussed in Section 4.2.1.1.7. The analysis of the cask/pad interface under seismic loadings is described in Section 8.2.1

The cask transporter is designed, fabricated, inspected, operated, maintained, and tested in accordance with the applicable guidelines of NUREG-0612. Thus, there is no need to establish lift height limits or to postulate cask-drop events during transport to the pad, including activities at the CTF.

The cask lifting assembly on the transporter is a horizontal beam that is supported by towers at each end with hydraulic lifting towers. During movement of the transporter with the cask in a fixed elevation, a redundant load support system is used. This is further described in Section 4.3 and in Chapter 5.

4.2.3.3.2.6 Tornado Winds and Missiles

Design criteria for tornado wind and missile impact are discussed in Section 3.2.1. The HI-STORM 100 System is designed to withstand pressures, wind loads, and missiles generated by a tornado, as described in Section 2.2.3.5 of the HI-STORM 100 System FSAR. In Section 8.2.2, the analysis of the Diablo Canyon ISFSI site design-basis tornado, including pressures, wind loads, and missiles is discussed. The MPC confinement boundary remains intact under all design-basis, tornado-wind, and missile-load combinations.
Tornado-wind and missile loads are evaluated for the overpack and the transfer cask. In the case of the transfer cask, the loaded transfer cask is always maintained in a restrained condition by the handling equipment while it is in a vertical position. Tipover or instability due to tornado-wind or missile impact is therefore a noncredible accident for the transfer cask (HI-STORM 100 System FSAR, Sections 2.2.3.1 and 3.4.8). However, missile penetration effects on the transfer cask and overpack have been evaluated.

4.2.3.3.2.7 Flood

Flooding is addressed in Sections 3.2.2 and 8.2.3 of this FSAR and in Sections 3.1.2.1.1.3 and 3.4.6 of the HI-STORM 100 System FSAR. The MPC is designed to withstand hydrostatic pressure (full submergence) up to a depth of 125 ft and horizontal loads due to water velocity up to 15 fps without tipping or sliding. The Diablo Canyon ISFSI and CTF are above probable maximum flood conditions; therefore, the HI-STORM 100 System FSAR evaluation bounds conditions at the Diablo Canyon ISFSI storage and CTF sites. Thus, the requirements of 10 CFR 72.122(b) are met with regard to floods.

4.2.3.3.2.8 Earthquake

Design criteria for earthquake loads at the Diablo Canyon ISFSI are discussed in Section 3.2.3. The results of the seismic analyses are discussed in Section 8.2.1. Analyses were performed using the DE, DDE, HE, and LTSP ground motions to verify that the Diablo Canyon ISFSI SSCs (including components of the HI-STORM 100 system) meet their design requirements of 10 CFR 72.122(b) with regard to earthquakes. Although not considered a licensing basis, PG&E has evaluated the effects of recent data (ILP ground motions, Section 2.6.2.4.2) to ensure appropriate design margins are maintained.

4.2.3.3.2.9 Explosion Overpressure

Explosion overpressure loads are addressed in Sections 3.3.1.6 and 8.2.6.2.1 of this FSAR and in Sections 3.4.7.2 and 11.2.11 of the HI-STORM 100 System FSAR. The HI-STORM 100 System MPC is analyzed and designed for accident external pressures up to 60 psig. The transfer cask overpressure design limit is 384 psig. The overpack is designed for steady-state and transient external pressures of 5 psig and 10 psig, respectively. As shown in Section 8.2.6, the Diablo Canyon ISFSI is not subject to credible explosions (that is, transient external pressures) that are in excess of 1 psig or are risk significant in accordance with Regulatory 1.91. Therefore, the HI-STORM 100 System bounds the expected overpressure due to explosions at the Diablo Canyon ISFSI, as required per 10 CFR 72.122(c).

4.2.3.3.2.10 Fire

Design criteria for fire loads are addressed in Section 3.3.1.6 and in the HI-STORM 100 System FSAR, Section 11.2.4. The HI-STORM 100 System was analyzed for a fire of 50 gallons of combustible fuel from the cask transporter encircling the cask, resulting in temperatures up to 1,475°F and lasting for a period of 3.6 minutes. The analysis also evaluated the post-fire temperatures of the system for the duration of 10 hours. The evaluation of this fire and its effect on both the loaded overpack and the loaded transfer cask is discussed in Section 11.2.4 of the HI-STORM 100 System FSAR. The results of the analysis show that the intense heat from the fire only partially penetrated the concrete-cask wall. This fire would cause less than 1 inch of concrete to exceed temperature limits, and would have a negligible effect on shielding or MPC and fuel temperatures.

For the Diablo Canyon ISFSI, the threat of fire was evaluated for a variety of potential sources in addition to the transporter fire, including a vehicle fuel tank, other local fuel tanks, other combustible materials, and a vegetation fire. The results of this evaluation are discussed in Section 8.2.5.

The HI-STORM 100 System design meets the Diablo Canyon ISFSI design criteria for accident-level thermal loads as required per 10 CFR 72.122(c).

4.2.3.3.2.11 Lightning

A lightning strike of the HI-STORM 100 System at the Diablo Canyon ISFSI is addressed in Sections 3.2.6 and 8.2.8. The lightning strike accident is also discussed in the HI-STORM 100 System FSAR, Sections 2.2.3.11 and 11.2.12. The analysis shows that the lightning will discharge through the steel shell of the overpack or the transfer cask to ground through a ground connector. The lightning current will discharge through the affected steel structure and will not affect the MPC, which provides the confinement boundary for the spent fuel.

Therefore, the HI-STORM 100 System design meets the Diablo Canyon ISFSI design criteria in Section 3.2.6 for lightning protection, as required in 10 CFR 72.122(b).

4.2.3.3.2.12 500-kV Line Drop

The Diablo Canyon ISFSI storage site is located underneath and adjacent to 500-kV transmission lines. The HI-STORM 100 System design criteria for a 500-kV transmission line dropping and striking the HI-STORM 100 overpack or transfer cask is similar to the lightning strike. Section 8.2.8 discusses the analysis of this accident and demonstrates that the MPC remains protected. The HI-STORM 100 System, therefore, meets the requirements of 10 CFR 72.122(b) for the 500-kV line break.

4.2.3.3.3 Thermal Design

The environmental thermal design criteria for the Diablo Canyon ISFSI are discussed in Section 3.2.7. Thermal performance for the HI-STORM 100 System is addressed in Chapter 4 of the HI-STORM 100 System FSAR. The HI-STORM 100 System is designed for long-term storage of spent fuel and safe thermal performance during onsite loading, unloading, and transfer operations. The HI-STORM 100 System is also designed to minimize internal stresses from thermal expansion caused by axial and radial temperature gradients. The thermal model and its benchmarking with full size cask test data is described in Reference 20.

The HI-STORM 100 System is designed to transfer decay heat from the spent fuel assemblies to the environment. The MPC design, which includes the all-welded honeycomb basket structure, provides for heat transfer by conduction, convection, and radiation away from the fuel assemblies, through the MPC basket structure and internal region, to the MPC shell. The internal MPC design incorporates top and bottom plenums, with interconnected downcomer paths, to accomplish convective heat transfer via the thermosiphon effect. The MPC is pressurized with helium, which assists in transferring heat from the fuel rods to the MPC shell by conduction and convection. Gaps exist between the basket and the MPC shell to permit unrestrained axial and radial thermal expansion of the basket without contacting the shell, thus minimizing internal stresses. The stainless-steel basket conducts heat from the individual spaces for storing fuel assemblies out to the MPC shell.

The HI-STORM 100SA overpack design provides an annular space between the MPC shell and the inner steel liner of the overpack for airflow up the annulus. Relatively cool air enters the four inlet ducts at the bottom of the overpack, flows upward through the annulus removing heat from the MPC shell by convection, and exits the four outlet ducts at the top of the cask.

The thermal analysis, discussed in Chapter 4 of the HI-STORM 100 System FSAR was performed using the ANSYS and FLUENT (Reference 14) computer codes. The HI-STORM PWR MPCs (MPC-24, MPC-24E, MPC-24EF, and MPC-32) were evaluated to determine the temperature distribution under long-term, normal storage conditions, assuming the MPCs are loaded with design basis PWR fuel assemblies.

Maximum-assembly, decay-heat-generation rates for fuel to be loaded into these two MPC models are specified in Section 10.2.

The thermal analysis assumed that the HI-STORM overpacks are in an array, subjected to an 80°F-annual-average ambient temperature, with full insolation. The annual-average temperature takes into account day-and-night and summer-and-winter temperatures throughout the year. The annual-average temperature is the principal design parameter in the HI-STORM 100 System design analysis, because it establishes the basis for demonstration of long-term spent nuclear fuel integrity. The long-term integrity of the spent fuel cladding is a function of the average-ambient temperature over the entire storage period, which is assumed to be at the maximum annual-average temperature in every year of storage for conservatism. The results of this analysis are presented in Tables 4.4.9, 4.4.26 and 4.4.27 of the HI-STORM 100 System FSAR for MPC-24, MPC-24E, MPC-24EF, and MPC-32, respectively. The results, summarized in HI-STORM 100 System FSAR Table 4.2-3, indicate that temperatures of all components are within normal condition temperature limits. These results bound the Diablo Canyon ISFSI site since the average-annual temperature at the site is only 55°F (Section 2.3.2).

Section 11.1.2 of the HI-STORM 100 System FSAR discusses the temperatures of the HI-STORM 100 System for a maximum off-normal, daily-average ambient temperature of 100°F, which is an increase of 20° F from the normal conditions of storage discussed above. The maximum off-normal temperatures were calculated by adding 20° F to the maximum normal temperatures from the highest component temperature for the MPC-24, MPC-24E, MPC-24EF, and MPC-32. All of the maximum off-normal temperatures are below the short-term peak fuel cladding temperature limits (HI-STORM 100 System FSAR Table 2.2.3). Therefore, all components are within allowable temperatures for the 100° F-ambient-temperature condition. Since the highest hourly temperature recorded at the Diablo Canyon Site is 97° F (Section 2.3.2), the HI-STORM 100 System FSAR evaluation bounds the Diablo Canyon ISFSI site.

The thermal analysis in the HI-STORM 100 System FSAR discussed above includes the following global assumptions: (a) the concrete pad is assumed to be an insulated surface (that is, no heat transfer to or from the pad is assumed to occur), (b) adjacent casks are assumed to be sufficiently separated from each other (that is, cask pitch is sufficiently large) so that their ventilation actions are autonomous, and (c) the cask is assumed to be subject to full solar insolation on its top surface as well as view-factor-adjusted solar insolation on its lateral surface, based on 12-hour insolation levels recommended in 10 CFR 71 (800g-cal/cm^2 averaged over a 24-hour period as allowed in NUREG-1567). The evaluation of insolation is further discussed in Section 4.4.1.1.8 of the HI-STORM 100 System FSAR.

Ambient-temperature and incident solar radiation (insolation) values applicable to the ISFSI site are summarized in Section 2.3.2. The highest and lowest hourly recorded temperature, as recorded at one of the recording stations at the Diablo Canyon site, is 97° F in October 1987 and 33°F in December 1990, respectively. The annual-average temperature is approximately 55° F. The maximum insolation values for the ISFSI site are estimated to be 766 g-cal/cm^2 per day for a 24-hour period and 754 g-cal/cm^2 for a 12-hour period.

Second-order effects such as insolation heating of the concrete pad, heating of feed air traveling downward between casks and entering the inlet ducts of the reference cask, and radiative heat transfer from adjacent spent fuel casks were not explicitly modeled in the HI-STORM 100 System FSAR analysis.

Within a loaded transfer cask, heat generated in the MPC is transported from the contained fuel assemblies to the MPC shell. A small, diametrical air gap exists between the outer surface of the MPC and the inner surface of the transfer cask. Heat is transported across this gap by the parallel mechanisms of conduction, natural convection, and thermal radiation. Assuming that the MPC is centered and does not contact the transfer cask walls conservatively minimizes heat transport across this gap. Additionally, thermal expansion that would minimize the gap is conservatively neglected.

Heat is transported through the cylindrical wall of the transfer cask by conduction through successive layers of steel, lead, and steel. A water jacket, which provides neutron shielding for the transfer cask, surrounds the cylindrical steel wall. The water jacket is composed of a carbon steel shell attached to the outer shell of the transfer cask by radial fins. Conduction heat transfer occurs through both the water cavities and the fins. While the water jacket openings are sufficiently large for natural convection loops to form, this mechanism is conservatively neglected. Heat is passively rejected to ambient from the outer surface of the transfer cask by natural convection and thermal radiation.

In the vertical position, the bottom face of the transfer cask is in contact with a supporting surface. This face is conservatively modeled as an insulated surface. Because the transfer cask is not used for long-term storage in an array, radiative heat blocking does not need to be considered. The transfer cask top lid is modeled as a surface with convection, radiative heat exchange with air, and a constant, maximum-incident solar heat flux load. Insolation on cylindrical surfaces is conservatively based on 12-hour levels prescribed in 10 CFR 71 and averaged on a 24-hour basis. Concise descriptions of these models are described in Section 4.5 of the HI-STORM 100 System FSAR.

The HI-STORM 100 System was analyzed for an extreme hot ambient temperature of 125°F averaged over a 72-hour time period. Section 8.2.10 of this FSAR and Section 11.2.15 of the HI-STORM 100 System FSAR provide discussions of the analysis of this extreme temperature condition. The ambient temperature is applied coincident with full solar insolation. Resulting fuel cladding temperatures are well below their short-term temperature limit. The balance of the HI-STORM 100 System structure remains insignificantly affected. Since the extreme hot ambient temperature at the Diablo Canyon site is 104°F, the extreme hot ambient temperature evaluation in the HI-STORM 100 System FSAR bounds the conditions at Diablo Canyon.

The HI-STORM 100 System was also evaluated for a -40°F, extreme-low ambient temperature condition, as discussed in Section 4.4.3 of the HI-STORM 100 System FSAR. Zero decay heat generation from spent fuel, and no solar insolation were conservatively assumed. All materials of construction for the MPC and overpack will perform their design function under this extreme cold condition. Since the minimum temperature at the Diablo Canyon site is greater than 24°F (Table 3.4-1), the extreme low ambient temperature evaluation in the HI-STORM 100 System FSAR bounds the conditions at Diablo Canyon.

At Diablo Canyon, the thermal performance of the MPC to limit fuel cladding temperature inside the transfer cask during welding, draining, drying, and helium backfill operations, and during transportation of the loaded transfer cask to the CTF is bounded by the thermal evaluation performed with the MPC helium filled and in the transfer cask with the annulus void of water. This condition is bounding for the other transient operational conditions mentioned above, because it maximizes the resistance to heat transfer from the MPC shell to the environment. In the other conditions, there are temperature controls on either the helium or water in the MPC cavity to limit the cladding temperature. The maximum cladding temperature for this bounding condition is well below the 1058°F short term limit.

When a modified MPC-32 was developed for use at the Diablo Canyon ISFSI, a site specific thermal evaluation (Reference 36) was performed to verify that the modified design was in compliance with the limits established for the HI-STORM 100 system. This analysis demonstrated that for all conditions of system operation with a design basis heat load, the required temperature limits were met.

In support of LA 2, the site specific thermal analysis was updated to a 3-D CFD analysis (Reference 40), and the analysis was modified to address the storage of HBF in accordance with the requirements of ISG-11, Rev. 3. That analysis covers uniform loading of HBF up to a 24 kilowatt heat load limit. In support of LA 3, an additional site specific thermal analysis was performed using the same methodology allowing up to a 28.74 kilowatt heat load for uniform loading and 25.572 kilowatt heat load for regionalized loading (Reference 43). This analysis demonstrated that fuel cladding temperatures met the requirements for all conditions, although a supplemental cooling system (SCS) was required for a helium filled MPC loaded with HBF in the HI-TRAC while temporary shielding is installed on the transfer cask, or while unloading an MPC-32 loaded under Amendment 2 of this license.. The SCS is used to maintain the temperature of the MPC shell at a temperature that ensures that the maximum temperature of the fuel cladding does not exceed its long term limits. As part of this new analysis, some of the individual component temperature limits were updated to those authorized in later HI-STORM Amendments (through Amendment 5) and two normal ambient temperatures were assumed based on the system configuration. A normal ambient temperature of 65°F was assumed for a loaded MPC contained in a HI-STORM overpack on the ISFSI pad and for a loaded MPC contained located in a HI-STORM overpack within CTF. All transport configurations with a loaded MPC contained within the HI-TRAC assumed a normal ambient temperature of 100°F.

The above discussion demonstrates that the HI-STORM 100 System as deployed at the Diablo Canyon ISFSI meets the requirements of 10 CFR 72.122(h), 72.128(a)(4), and 72.236(f) and (g) for thermal design.

4.2.3.3.3.1 HI-STORM Overpack at the CTF

The site-specific design of the Diablo Canyon CTF involves transferring a loaded MPC into the overpack with the overpack located below grade in a vault. The thermal implications of the difference between a loaded overpack located in a vault and one located at grade level have been evaluated.

Under normal conditions, the loaded overpack remains in the vault only for the time it takes to remove the transfer cask from atop the overpack, retrieve and install the overpack lid, and raise the overpack out of the vault with the transporter. This is expected to take less than 4 hours and has an insignificant effect on heat removal and fuel cladding temperatures.

Under off-normal conditions, such as a transporter failure affecting the CTF lift operation, the condition could last several hours, depending upon the time it takes to complete corrective actions to restore the transporter, or to provide an alternate lift capability. The effect of a loss of function of the transporter on the ability of the overpack to transfer the heat from the MPC to the environs is discussed in Section 8.1.7. The evaluation shows that ISG-11 Rev. 3 cladding temperature limits are not exceeded, and the MPC can remain in this configuration for as long as necessary to allow restoration of transporter function, as described in Section 8.1.7.

4.2.3.3.4 Shielding Design

Shielding design and performance for the HI-STORM 100 System is addressed in Section 3.3.1.5.2 and Chapter 7 of this FSAR specifically for the Diablo Canyon ISFSI, and in Chapter 5 of HI-STORM 100 System FSAR (Reference 42) for the HI-STORM 100 System generically. The HI-STORM 100 System is designed to maintain radiation exposure ALARA in accordance with 10 CFR 72.126(a). The concrete overpack is designed to limit the average external contact dose rates (gamma and neutron) to 135 mrem/hr on the sides, 60 mrem/hr on top, and 135 mrem/hr at the air inlets and outlets based on HI-STORM design basis fuel.

The overpack is a massive structure designed to provide gamma and neutron shielding of the spent fuel assemblies stored within the MPC. Most of the side shielding is provided by the overpack, although the MPC structure is credited in the shielding model. The overpack steel inner shell, the concrete-filled annulus, and the steel outer shell provide radiation shielding for the side of the overpack. The steel MPC lid and the overpack lid provide axial shielding at the top. The MPC lid is approximately 10 inches thick and is stainless steel. The overpack lid consists of a 4-inch thick steel top plate and steel-encased concrete. The lid shield configurations differ between the HI-STORM 100 and the HI-STORM 100S designs as shown on the respective drawings in Section 1.5 of the HI-STORM 100 System FSAR. In both designs, particular emphasis is placed on providing overpack lid shielding above the annulus between the MPC and the overpack inner shell, which is a streaming path.

The configuration of the inlet and outlet ducts in relation to the MPC prevents a direct radiation-streaming path from the MPC to outside the cask. The duct dose rates are further reduced by the installation of duct photon attenuators to minimize scatter (Figure 4.2-7). The HI-STORM 100 System design allows for necessary personnel access during inspection and maintenance operations, while keeping dose rates ALARA. The HI-STORM 100 System FSAR (Reference 42), Section 5.1.1 provides generic calculated dose rates around the sides and top of the HI-STORM 100S overpack. Predicted Diablo Canyon ISFSI dose rates and site-specific dose evaluations are presented in Chapter 7 for the HI-STORM 100 System, and meet the requirements of 10 CFR 72.104 and 72.106.

The transfer cask provides shielding to maintain occupational exposures ALARA in accordance with 10 CFR 20, while also maintaining the maximum load on the FHB crane hook to 125 tons or less. The dose rates for a transfer cask loaded with design basis fuel which are used to perform the occupational exposure estimate for MPC loading, closure, and transfer operations, are described in Chapter 7. The actual dose rates from a loaded transfer cask during operations in support of loading fuel for the Diablo Canyon ISFSI will be lower because the actual MPCs to be loaded will not contain design-basis fuel in every fuel storage location. Occupational exposures during transfer cask operations are monitored and maintained ALARA in accordance with the DCPP radiation protection program and the requirements of 10 CFR 20.

The above discussion demonstrates that the HI-STORM 100 System as used at the Diablo Canyon ISFSI meets the requirements of 10 CFR 72.104, 72.106, 72.128(a)(2), and 72.236(d) for shielding design.

4.2.3.3.5 Criticality Design

Criticality of the HI-STORM 100 System is addressed in Section 3.3.1.4 of this FSAR and Chapter 6 of the HI-STORM 100 System FSAR. The HI-STORM 100 System is designed to maintain the spent fuel subcritical in accordance with 10 CFR 72.124(a) and (b) with the MPC materials and geometry. The acceptance criterion for the prevention of criticality is that k_{eff} remain below 0.95 for all normal, off-normal, and accident conditions.

Criticality safety of the HI-STORM 100 System depends upon the following four principal design parameters:

- Administrative limits on the maximum fuel assembly enrichment and physical properties acceptable for storage in the MPC

- The inherent geometry of the fuel basket designs within the MPC, including the flux-traps in the MPC-24, MPC-24E, and MPC-24EF (water gaps for loading fuel into submerged MPCs)

- The incorporation of permanent, fixed, neutron-absorbing panels (Boral or Metamic) in the fuel basket structure to assist in control of reactivity

- Administrative controls requiring minimum concentrations of soluble boron in the MPC water during fuel loading and unloading, depending upon MPC model and fuel enrichment

The criticality analysis performed for the HI-STORM 100 System assumes only fresh fuel with no credit for burnup as a conservative bounding condition. In addition, no credit is taken for fuel-related burnable neutron absorbers, and it is assumed that the Boron-10 content in the Boral is only 75 percent of the manufacturer's minimum specified content, and the Boron-10 content of Metamic is only 90 percent of the manufacturer's minimum. Boral or Metamic panels are intended to have no significant flaws. However, to account for manufacturing deviations occurring during installation of the panels into the MPC fuel basket, neutron absorber damage up to the equivalent of a 1-inch diameter hole in each panel has been analyzed and found to be acceptable (Appendix H of Reference 37). Other assumptions made to ensure the results of the analysis are conservative are identified in Section 6.1 of the HI-STORM 100 System FSAR.

In its storage configuration, the HI-STORM 100 System is dry (no moderator), and the reactivity is very low (k_{eff} less than 0.515). At the Diablo Canyon ISFSI, the fuel is always in a dry, inert-gas environment. It is sealed within a welded MPC, and no credible accident will result in water entering the MPC. The limiting reactivity condition occurs in the SFP during fuel loading, where assemblies are loaded into the MPC in close proximity to each other, with moderator between assemblies. All fuel loaded into the MPC-32, regardless of enrichment, requires a certain amount of soluble boron in the MPC during loading to preserve the assumptions of the criticality analyses. Higher enriched fuels loaded into the MPC-24, MPC-24E, or MPC-24EF also require soluble boron in the MPC during loading operations. The Diablo Canyon ISFSI TS ensure that soluble boron is appropriately maintained during fuel loading operations.

The results of the criticality analyses of different fuel types are shown in Chapter 6 of the HI-STORM 100 System FSAR for the MPC-24, MPC-24E, MPC-24EF, and MPC-32. The results confirm that the maximum reactivities of the MPCs are below the design criteria (k_{eff} less than 0.95) for fuels with specified maximum allowable enrichments, considering calculational uncertainties. The PWR fuel types for which these analyses were performed are shown in Table 2.1.3 of the HI-STORM 100 System FSAR. All DCPP fuel is bounded by array/classes 17x17A and 17x17B. No credit is taken for neutron poison in the form of gadolinium in the fuel pellets or in the IFBA rods; therefore, fuel assemblies containing these poisons are acceptable for loading.

Accident conditions have also been considered, and no credible accidents have been identified that would result in exceeding the regulatory limit on reactivity. In Section 6.1 of the HI-STORM 100 System FSAR Holtec determined that the physical separation between overpacks due to the large diameter and cask pitch, and the concrete and steel radiation shields, are each adequate to preclude any significant neutronic coupling between HI-STORM 100 Systems.

Section 6.4.4 of the HI-STORM 100 System FSAR discusses the results of criticality analyses on MPCs storing damaged fuel in a Holtec damaged fuel container. Analyses were performed for three possible scenarios. The scenarios are:

- Lost or missing fuel rods, calculated for various numbers of missing rods in order to determine the maximum reactivity.

- Fuel assembly broken with the upper segments falling into the lower segment creating a close-packed array. For conservatism, the array was assumed to retain the same length as the original fuel assemblies.

- Fuel pellets lost from the assembly and forming powdered fuel dispersed through a volume equivalent to the height of the original fuel, with the flow channel and cladding material assumed to disappear.

Results of these analyses confirm that, in all cases, the maximum reactivity of the HI-STORM 100 System with design-basis failed fuel in the most adverse post-accident condition will remain well below the regulatory limit within the enrichment range analyzed.

The HI-STORM 100 System is designed such that the fixed neutron absorber (Boral or Metamic) will remain effective for a storage period greater than 20 years, and there are no credible means to lose the Boral or Metamic effectiveness. As discussed in Section 6.3.2 of the HI-STORM 100 System FSAR, the reduction in Boron-10 concentration due to neutron absorption from storage of design-basis fuel in a HI-STORM 100SA overpack over a 50-year period is expected to be negligible. Further, the analysis in Appendix 3.M of the HI-STAR 100 System FSAR demonstrates that the sheathing, which affixes the Boral or Metamic panel, remains in place during all credible accident conditions, and thus the Boral or Metamic panel remains fixed for the life of the Diablo Canyon ISFSI. Therefore, verification of continued efficacy of the Boral or Metamic neutron absorber is not required. This is consistent with the requirements of 10 CFR 72.124(b).

For MPCs filled with pure water, the reactivity of any PWR assembly with nonfuel hardware inserted into the guide tubes is bounded by (that is, lower than) the reactivity of the same assembly without the inserts. This is because the inserts reduce the amount of moderator, while the amount of fissile material remains unchanged. In the presence of soluble boron in the water, especially for higher-required soluble boron concentrations, it is possible that the nonfuel hardware in the PWR assembly results in an increase of reactivity. This is because the insert not only replaces water, but also replaces the neutron absorber in the water with a nonpoison material. To account for this effect, analyses with and without nonfuel hardware in the assemblies were performed for higher soluble boron concentrations in support of Holtec LAR 1014-1. The highest reactivities for either case are used as the basis of the criticality evaluation. Section 6.4.8 of the HI-STORM 100 System FSAR provides additional discussion of the criticality effect of nonfuel hardware stored with PWR spent fuel assemblies.

During development of the DC ISFSI License Application, PG&E identified that a criticality analysis had not been performed for the VANTAGE 5 option of annular pellets in the axial blanket region of the fuel assemblies, should the annular pellets be flooded with water. Holtec subsequently performed the analysis and documented it in Appendix R of Holtec Report HI-2012771, Revision 12, "HI-STAR 100 and HI-STORM 100 Additional Criticality Calculations", dated April 30, 2006. This analysis concluded that up to 12 inches of annular pellets, of various IDs, in the axial blanket regions of a fuel assembly show no significant reactivity effects, even if the annular region is flooded with pure water. All Holtec criticality calculations for PWR fuel assemblies have been performed using solid pellets along the entire length of the active fuel region, and the results are directly applicable to those PWR assemblies with annular pellets. This analysis was accepted by the NRC during the HI-STORM CoC Amendment 3 review, as documented in the associated SER. As such, there is no need for an administrative restriction to the VANTAGE 5 fuel allowed for loading and storage at the DC ISFSI based on the use of the annular pellet option.

During cask loading and unloading activities in the FHB/AB, criticality monitoring requirements of 10 CFR 72.124(c) are met using a combination of installed and portable radiation monitoring instrumentation, in accordance with GDC-63 (to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions). As discussed in PG&E letter DCL-97-058, dated April 3, 1997, the radiation monitoring instrumentation generally conforms to the guidance of Regulatory Guide 8.12, “Criticality Accident Alarm Systems,” and ANSI/ANS 8.3-1979, “Criticality Accident Alarm System.” As discussed in DCPP FSAR Update Section 9.1.2.3.5, spent fuel pool radiation monitors RM-58 and RM-59 provide personnel protection and general surveillance of the spent fuel pool area. As discussed in DCL-97-058, portable radiation monitors are placed in the cask washdown area to provide personnel protection and general surveillance of this area. On November 12, 1997, the NRC granted PG&E an exemption from the requirements of 10 CFR 70.24 concerning criticality monitors. In DCL-02-044 dated April 15, 2002, which submitted License Amendment Request 02-03, Spend Fuel Cask Handling, PG&E requested an exemption from the 10 CFR 72. 124(c) criticality monitoring requirement by requesting an extension of the NRC’s November 12, 1997, exemption for the FHB/AB to envelop the activities associated with the Diablo Canyon ISFSI FSAR.

In PG&E letter DCL-02-117, “Change in Licensing Basis Compliance from 10 CFR 70.24 to 10 CFR 50.68(b),” dated October 2, 2002, PG&E informed the NRC that PG&E would revise the DCPP licensing basis to reflect compliance with 10 CFR 50.68(b) in lieu of 10 CFR 70.24 and that the exemption request in PG&E letter

DCL-02-044 would be revised to request a similar exemption from 10 CFR 50.68(b) in lieu of 10 CFR 70.24.

10 CFR 50.68(b)(1) prohibits the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water. Specifically, the regulation ensures a subcritical condition will be maintained without credit for soluble boron. For an MPC loaded with fuel having the highest permissible reactivity, soluble boron credit is necessary to ensure the MPC remains subcritical in the DCPP SFP. Therefore, PG&E requested an exemption from 10 CFR 50.68(b)(1) to allow MPC loading, unloading, and handling operations without meeting the requirement of being subcritical under the most adverse moderation conditions feasible by unborated water.

In the exemption request (Reference 30), PG&E evaluated the possibility of an inadvertent criticality during MPC loading, unloading, and handling in the DCPP SFP. Based on the alarms, procedures, administrative controls, assumption of zero burnup fuel, and availability of trained operators described in Reference 30, the NRC granted an exemption (Reference 31) from the criticality requirements of 10 CFR 50.68(b)(1) during loading, unloading, and handling of the MPC in the DCPP SFP.

The Holtec design, associated procedural controls, the proposed Diablo Canyon ISFSI Technical Specifications (TS) and Section 10.2 preclude accidental criticality when the spent fuel has been properly placed in the storage cask confinement system and the confinement system has been adequately drained, dried, inerted, and sealed.

The analysis of a fuel assembly drop onto the racks, and the drop of a fuel cask in the SFP, shows criticality is prevented and is also addressed in the 10 CFR 50 spent fuel cask handling LAR and license amendments (References 32 and 33, respectively).

The above discussion demonstrates that the HI-STORM 100 System as deployed at the Diablo Canyon ISFSI meets the requirements of 10 CFR 72.124 and 72.236(c) for criticality design.

4.2.3.3.6 Confinement Design

Confinement design for the HI-STORM 100 System is addressed in Chapter 7 of the HI-STORM 100 System FSAR. The confinement vessel of the HI-STORM 100 System is the MPC, which provides confinement of all radionuclides under normal, off-normal, and accident conditions in accordance with 10 CFR 72.122(h). The MPC consists of the MPC shell, bottom base plate, MPC lid, vent and drain port cover plates, and the MPC closure ring, which form a totally welded vessel for the storage of spent fuel assemblies. The MPC requires no valves, gaskets, or mechanical seals for confinement. All components of the confinement system are classified as important to safety.

The MPC is a totally welded pressure vessel designed to meet the stress criteria of ASME Section III, Subsection NB. No bolts or fasteners are used for closure. All factory welds are examined per ASME Section III and helium leak tested to ensure conformance to the offsite dose analysis. All closure welds are examined using the liquid-penetrant method. Two penetrations are provided in the MPC lid for draining, drying, and backfilling during loading operations. Following loading operations, vent and drain port cover plates are welded to the MPC lid and helium leak tested to ensure their integrity. A closure ring, which covers the penetration cover plates and welds, is welded to the MPC lid to provide redundant closure of the MPC vessel. The loading and welding operations are performed inside the DCPP FHB/AB. There are no confinement boundary penetrations required for MPC monitoring or maintenance during storage.

For those MPCs to be loaded with HBF, the confinement boundary will be considered leak tight. The factory shell welds and the vent and drain port cover plate welds will be helium leakage tested to the “leaktight” criteria of ANSI N14.5-1997. The lid-to-shell (LTS) weld is a large, multi-pass weld which is placed and inspected in accordance with ISG-15; therefore, in accordance with ISG-18, leakage from this weld is considered non-credible.

The confinement features of the HI-STORM 100 System meet the requirements of 10 CFR 72.122(h).

4.2.4 INSTRUMENTATION SYSTEM DESCRIPTION

Monitoring of the loaded casks on the storage pad is necessary to ensure that the passive, air- cooled, convective heat transfer system for the MPC and overpack remains operable. Rather than install an active temperature monitoring system, PG&E has chosen to visually monitor overpack inlet and outlet air duct perforated plates (screens), as required by the Diablo Canyon ISFSI TS, to verify the perforated plates (screens) are free of blockage and intact.

4.2.5 COMPLIANCE WITH GENERAL DESIGN CRITERIA

Table 4.2-5 provides a tabular presentation of the locations in this SAR and/or the HI-STORM 100 System FSAR where compliance with the General Design Criteria of 10 CFR 72, Subpart F, is shown to be met.

4.2.6 REFERENCES

1. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision1A, January 2003.

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3. ANSYS Finite Element Modeling, ANSYS Inc., Southpointe 275 Technology Drive; Canonsburg, PA.

4. ACI-349-97, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, 1997.

5. 10 CFR 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

6. ANSI N14.6, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More, American National Standards Institute, 1993 Edition.

7. Control of Heavy Loads at Nuclear Power Plants, USNRC, NUREG- 0612, July 1980.

8. Boiler and Pressure Vessel Code, Section III, Division I, American Society of Mechanical Engineers, 1995 Edition including 1996 and 1997 addenda.

9. Submittal of Holtec Proprietary and Non-Proprietary Drawing Packages, PG&E Letter to the NRC, DIL-01-008, dated December 21, 2001.

10. ACI 349-85, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute.

11. Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, USNRC, NUREG/CR-6407, February 1996.

12. Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update.

13. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Revision 0, May 2000.

14. FLUENT Computational Fluid Dynamics Software, Fluent, Inc., Centerra Resource Park, 10 Cavendish Court, Lebanon, NH 03766.

15. PG&E Calculation No, 52.27.100.705 (PGE-009-CALC-001), “Embedment Support Structure.”

16. Calculation PGE-009-CALC-006, “ISFSI Cask Storage Pad Concrete Shrinkage and Thermal Stresses.”

17. Calculation PGE-009-CALC-007, ISFSI Cask Storage Pad Steel Reinforcement.”

18. PG&E Calculation 52.27.100.718 (GEO.DCPP.01.08), “Determination of Rock Anchor Design Parameters for DCPP ISFSI Cutslope.”

19. Holtec International Report No. HI-2002474, “Analysis of the Loaded HI-STORM 100 System Under Drop and Tipover Scenarios,” Revision 2.

20. Holtec International Report No. HI-992252, “Topical Report on the HI-STAR/HI-STORM Thermal Model and its Benchmarking with Full-Size Cask Test Data,” Revision 1.

21. PG&E Calculation OQE-017, “Cask Transfer Facility Seismic Restraint Configuration.”

22. ACI 349-01, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, 2001.

23. Holtec International Report No. HI-2002570, “Design Criteria Document for the Diablo Canyon Cask Transfer Facility,” Revision 5.

24. PG&E Calculation 52.27.100.708 (PGE-009-CALC-002), “Cask Transfer Facility (Reinforced Concrete).”

25. Holtec International Report No. HI-2053370, “Structural Analysis of CTF at DCNP Under Design Basis Loads,” Revision 2.

26. PG&E Letter DIL-03-003 to the NRC, Revised Response to NRC Request for Additional Information 5-1 for the Diablo Canyon ISFSI Application, March 27, 2003.

27. PG&E Letter DIL-03-004 to the NRC, Supplemental Slope Stability Response to Additional NRC Questions for the Diablo Canyon ISFSI Application, March 27, 2003.

28. PG&E Letter DIL-03-007 to the NRC, Supplemental Slope Stability Design Mitigation Features Information to Additional NRC Questions for the Diablo Canyon ISFSI Application, May 6, 2003.

29. PG&E Letter DIL-03-015 to the NRC, Additional Information on Cask Transfer Facility Cask Transporter Lateral Restraint System, December 4, 2003.

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

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

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

34. Regulatory Guide 1.142, Safety Related Concrete Structure for Nuclear Power Plants (Other than Reactor Vessels and Containment), USNRC, November 2001.

35. Regulatory Guide 1.199, Anchoring Components and Structural Supports in Concrete, USNRC, November 2003.

36. Holtec International Report No. HI-2053376, “Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System Design,” Revision 5.

37. Holtec International Report No. HI-2012771, “HI-STAR 100 and HI-STORM 100 Additional Criticality Calculations,” Revision 12.

38. Drawing 6021750, Sheet 310, DCPP ISFSI Cask Transfer Facility (CTF) Concrete Sections and Detail, Revision 2.

39. Drawing 6021750, Sheet 312, DCPP ISFSI Transporter Seismic Restraint Anchor Block, Revision 2.

40. Holtec International Report No. HI-2104625, "Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System Design," Revision 10.

41. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 7, August 2009.

42. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 5, June 2007.

43. Holtec International Report No. HI-2125191, “Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System with up to 28.74 kW Decay Heat,” Revision 6.

4.3 TRANSPORT SYSTEM

The cask transporter is designed and used to safely lift, handle, and transport a HI-TRAC transfer cask or a HI-STORM 100SA overpack, loaded with spent fuel and associated nonfuel hardware, between the DCPP FHB/AB, the cask transfer facility (CTF), and the Diablo Canyon ISFSI storage pad site as described below. The movement is conducted exclusively on the DCPP site as shown in Figure 2.1-2. Due to its important-to-safety classification, the transporter is licensed under 10 CFR 72 (Reference 1). The cask transporter is designed to withstand all design-basis, natural-phenomena events while lifting, handling, and moving the loaded transfer cask or overpack without impairing its ability to safely hold the load.

4.3.1 FUNCTION

The functions of the cask transporter considered important to safety are:

- Transporting the loaded transfer cask, in the vertical orientation, between the FHB/AB and the CTF.

- Lifting the loaded transfer cask and placing it atop the overpack at the CTF.

- Facilitating the transfer of the loaded MPC between the transfer cask and the overpack.

- Lifting the loaded overpack at the CTF.

- Transporting the loaded overpack between the CTF and its storage location on the Diablo Canyon ISFSI storage pad.

The cask transporter is capable of traveling over all of the road surfaces on the transport route. The road surfaces and underground facilities (see Section 4.3.3) will be evaluated to ensure the capability to support the weight of a cask transporter plus a loaded transfer cask or overpack.

4.3.2 COMPONENTS

This section describes the components used to lift, handle, and transport the loaded transfer cask and overpack to the CTF and Diablo Canyon ISFSI storage pad. Sections 3.3.3 and 3.4 provide discussion of the design criteria for the cask transportation system. Section 8.2.1 summarizes the results of the stress analyses under seismic loading, which bound the normal operation loads. Table 4.3-1 summarizes the functions of, and applicable design codes for, the transport system components that are considered important to safety and covered by an approved 10 CFR 72 quality assurance program.

4.3.2.1 Cask Transporter

4.3.2.1.1 Description

The cask transporter is a self-propelled, open-front, tracked vehicle used for handling and onsite transport of overpacks and the transfer cask with an MPC contained therein. It is nominally 27 $^{1}/_{2}$ ft long, 19 $^{1}/_{6}$ ft wide, and weighs approximately 95 tons, unloaded. It is designed with two steel tracks to spread out the load on the transport route surface as a distributed pressure load. These tracks provide the means to maneuver the cask transporter around the site. On top of the main structure is a lifting beam supported by two lifting towers that use hydraulic cylinders to provide the lifting force. The industrial-grade hydraulic cylinders are made of carbon steel to ensure high strength and ductility for all service conditions. The cask transporter is diesel-powered and is limited to a fuel volume of 50 gallons to comply with the Diablo Canyon ISFSI Technical Specifications (TS). The functional specification for the transporter is provided in Reference 6.

4.3.2.1.2 Design

The cask transporter is custom-designed for conditions at the Diablo Canyon ISFSI site, including the transport route with its maximum grade of approximately 8.5 percent. It will remain stable and will not overturn, experience structural failure, or leave the transport route should a design-basis seismic event occur while the loaded transfer cask is being moved to the CTF; while transferring an MPC at the CTF; or while moving a loaded overpack from the CTF to the storage pad. In addition, the cask transporter is designed to withstand DCPP design-basis tornado winds and tornado-generated missiles without overturning, dropping the load, or leaving the transport route. Other natural phenomena, such as lightning strikes, floods and fires have been evaluated and accounted for in the cask transporter design. All design criteria for natural phenomena used to design the cask transporter are specific to the Diablo Canyon ISFSI site (see Sections 3.2 and 3.4 for detailed information).

A lightning strike on the cask transporter would not structurally affect the transporter's ability to hold the load. Due to the massive amount of steel in the structure, the current would be transmitted to the ground without significantly damaging the transporter. However, should the lightning strike result in a loss of electrical or engine function, the vehicle will be automatically stopped (when moving) and the brakes applied or load movement will be stopped and mechanical locks applied to hold the load (when lifting or lowering). Tthe driver may be affected by a lightning strike. Therefore, the transporter design includes fail-safe features to automatically stop the vehicle (when moving) or stop load movement and apply mechanical locks to hold the load (when lifting or lowering) if the operator is incapacitated for any reason.

Flooding is not a concern on the transport route for reasons discussed in Section 3.2.2. Sources of fires and explosions have been identified in Sections 2.2 and 3.3.1.6 and in Table 3.4-1, and have been evaluated with respect to cask integrity in Sections 8.2.5

and 8.2.6. Fixed sources of fire and explosion are sufficiently far from the transport route to be of no concern. Mobile sources of fire and explosion, such as fuel tanker trucks, will be kept at a safe distance away from the transporter during loaded cask movement through the use of administrative controls.

The cask transporter is capable of forward and reverse movement as well as turning and stopping. It includes an on-board engine that is capable of supplying enough power to perform its design functions. The cask transporter includes fail-safe service brakes that automatically engage in any loss of power (i.e., a loss of hydraulic or electrical) and an independent parking brake on each tractor motor. The brake system is capable of stopping and holding a fully loaded cask transporter on the maximum design grade. The cask transporter is also equipped with an automatic drive brake system that applies the brakes if there is a loss of hydraulic pressure (e.g., spring set tractor motor brake) or decelerates if the drive controls are released (e.g., hydraulic system pressure relief). Additionally, the fully loaded cask transporter is not capable of coasting on a 10 percent downward grade with the brakes disengaged, due to the passive resistance in the drive system that is inherent in the design of each multi-stage, planetary gear, tractor drive transmission.

The cask transporter is equipped with a control panel that is suitably positioned on the transporter frame to allow the operator easy access to the controls located on the control panel and, at the same time, allow an unobstructed view of the cask handling operations. The control panel provides for all-weather operation or will be adequately protected. The control station includes controls for all cask transporter operations including speed control, steering, braking, load raising and lowering, cask restraining, engine control and “dead-man” and emergency stop switches. Additional emergency stop switches are located at ground level both in the front and rear of the transporter. A radio-remote control module can alternatively be used to control the drive and lift functions.

The cask transporter works with certain other ancillary components to facilitate the lifting and movement of the transfer cask and the overpack. Each ancillary component is described in Sections 4.3.2.2 through 4.3.2.7. Transfer cask vertical handling, using the cask transporter, is performed only in the vertical orientation using the transfer cask lift links. Likewise, overpack handling is performed only with the overpack in the vertical orientation using the HI-STORM lifting brackets.

The cask transporter and associated lifting components are classified important to safety, purchased commercial grade, and qualified for MPC and overpack transfer operations at the CTF by testing prior to service. These lifting components are defined as those components in the load path of the supported load. Special lifting devices, defined as any suspended load-bearing component below the integral load links, are designed in accordance with ANSI N14.6 (Reference 2) per the applicable guidance of NUREG-0612 (Reference 3). Table 4.3-1 provides a summary of the design code(s) applicable to each of the lifting and handling components.

On top of the main structure of the transporter is a lifting beam supported by two lifting towers that use hydraulic cylinders to provide the lifting force. Mechanical design features and administrative controls provide a defense-in-depth approach to preventing load drops during lifting and handling. The primary load-retaining devices of the cask transporter are the hydraulic cylinders. In combination, the hydraulic system is designed to carry the rated load, including a 15 percent hoist factor, or 1.15 times the rated load (425,500 lb). Calculated safety margins for cylinder buckling and hoop stress are a minimum 2:1 versus the buckling load limit and material yield strength, respectively (Reference 6).

Once the cask is raised to its travel height by the cylinders, a redundant load support system is used. This system consists of wedge locks-mechanical spring engagement/hydraulic disengagement. Wedge locks, by their shape, limit tower movement to the lift (up) direction only. Any failure of the lifting hydraulics will not result in an uncontrolled lowering of the load. The wedge locks are operable at all times when a load is being lifted or lowered. To remove the wedge locks, the cylinder must first be extended slightly to take the load off the wedge. The load may then be lowered using the lifting cylinders. Requiring the cylinders to take the load ensures that they are operational before lowering the load. Any failure of the hydraulic system at this time is mitigated by the cylinder safety systems as described below.

The cask transporter hydraulic system wedge lock design prevents uncontrolled lowering of the load upon a loss of hydraulic fluid. A minimum amount of hydraulic fluid system pressure is required to disengage the wedge locks to allow movement of the load. A loss of hydraulic fluid would drop the pressure in the system and engage the wedge locks, preventing further movement of the load until corrective actions can be implemented.

The cask transporter is used to lift and place the loaded transfer cask atop the overpack for MPC transfer. During the MPC transfer process, the transfer cask trunnion connections to the cask transporter (that is, lift links) must be disconnected to provide access for the MPC downloader. Prior to disconnecting the lift links, the transporter is restrained. The restraint limits movement of the cask transporter during the time the cask transporter is disengaged from the transfer cask trunnions. Section 5.1 provides additional detail on storage system operations.

The design of the cask transporter includes a lateral cask restraining system to secure the load during transport operations. The restraint system is designed to prevent lateral and transverse swinging of the load.

The cask transporter structure is designed to accommodate external loading from a lateral restraint system at the CTF to preclude seismic interaction with the cask system during MPC transfer operations in the CTF. As discussed in Reference 7, the restraints are steel struts or similar equipment suitably sized to restrain the transporter by transferring the restraint loading to the ground adjacent to the CTF support structure.

4.3.2.1.3 Radiation Protection

The driver of the cask transporter, when using the vehicle-mounted control panel, is the only person in proximity to the transfer cask during onsite transfer operations who requires specific radiation protection consideration. Dose rate and accumulated dose estimates for the driver during cask transport operations are included in Section 7.4 using DCPP design-basis spent fuel source terms. All necessary radiation protection measures are determined by DCPP radiation protection personnel at the time of fuel loading based on the actual dose rates in the immediate vicinity of the loaded transfer cask.

4.3.2.1.4 Functional Testing and Inspection

As part of normal storage system operations, the cask transporter is inspected for operating conditions prior to each ISFSI loading campaign typically consisting of several casks. During the operational testing of this equipment, procedures are followed that will affirm the correct performance of the cask transporter features that provide for safe fuel-handling operations.

4.3.2.2 Low Profile Transporter

The low profile transporter (LPT) is used to move the loaded transfer cask in a vertical orientation through the FHB/AB door. The LPT travels on a rail system that runs from inside the FHB/AB to the access road located outside the FHB/AB roll-up door. The loaded transfer cask mounted on the LPT exits the FHB/AB to the east and travels approximately 60 ft to the cask transporter. The route is level and straight. The rail system distributes the load to selected areas of the roadway. The LPT is pulled along the rails by a winch.

The LPT is designed as a dedicated-use multi-roller heavy haul device capable of supporting and moving the HI-TRAC transfer cask. The LPT has a wide rectangular frame fitted with four high-capacity rollers. The transfer cask is secured to the LPT baseplate through bolts along the bottom flange of the transfer cask. The LPT is also equipped with guide bumpers to resist lateral loads, tow points at both ends, HI-TRAC alignment pins, and lift points. The LPT is classified as important to safety. A structural evaluation of the LPT is provided in Holtec Report HI-2053390 (Reference 8).

4.3.2.3 Lift Links

The lift links are load-bearing, structural steel components used to connect the cask transporter lift points to the lifting trunnions on the transfer cask and the HI-STORM lifting brackets. The lift links transfer the force of the loaded transfer cask from the lifting trunnions to the cask transporter lifting points through connector pins. The lift links are also used to retrieve a loaded overpack from the CTF. The lift links are important to safety, and are designed in accordance with ANSI N14.6 per the guidance of NUREG-0612, Section 5.1.6.

4.3.2.4 MPC Downloader Slings

The MPC downloader slings are used to lower (or raise) the loaded MPC during MPC transfer operations between the transfer cask and the overpack. The MPC downloader slings transmit the force of the loaded MPC from the MPC lift cleats to the MPC downloader. The MPC downloader slings are important to safety, and are designed in accordance with ASME B30.9 per the guidance of NUREG-0612, Section 5.1.6.

4.3.2.5 MPC Lift Cleats

The MPC lift cleats are ancillary devices temporarily attached to the MPC lid and used during transfer of the loaded MPC between the transfer cask and the overpack. The MPC lift cleats transmit the weight of the loaded MPC to the MPC downloader slings. The MPC lift cleats are classified as important to safety. The MPC lift cleats are special lifting devices that are designed in accordance with ANSI N14.6 per the guidance of NUREG-0612, Section 5.1.6. The MPC Lift Cleat nuts are only required to be tightened to wrench tight to perform their intended function, as documented in Holtec Report HI-992234 (Reference 9).

4.3.2.6 HI-STORM Lifting Brackets and Slings

The HI-STORM lifting brackets are load-bearing, structural steel components used to connect the cask transporter lifting points to the lid studs on the overpack. The HI-STORM lifting brackets transfer the weight of the loaded overpack from the lid studs to the cask transporter lift points through connector pins. The HI-STORM lifting brackets are special lifting devices that classified as important to safety, and are designed in accordance with ANSI N14.6 per the guidance of NUREG-0612, Section 5.1.6.

The HI-STORM lifting slings are used to raise the loaded overpack out of the CTF and are designed to ASME B30.9 per NUREG-0612, Section 5.1.6.

4.3.3 CASK TRANSPORT ROUTE

The cask transport route between the FHB/AB and the CTF and the Diablo Canyon ISFSI storage pads is shown in Figure 2.1-2. The route begins in the radiological control area (RCA) behind the FHB/AB, extends through the protected area past the Unit 2 cold machine shop (U2 CMS), along Plant View Road near Parking Lot 6, Shore Cliff Road (the main access road), and Patton Cove Bypass (Sections 2.6.1.3 and 2.6.1.12.1) and then up along Reservoir Road. The route descends a maximum 8.5 percent grade (for approximately 200 ft) from the RCA to the U-2 CMS and then along Plant View Road, which is essentially flat. From the intersection of Plant View Road and Shore Cliff Road, there begins an approximate 8 percent uphill grade (for approximately 600 ft) and then an approximate 6 to 8 percent grade (for approximately 3,000 ft) that continues along Reservoir Road. The route ends with a right-hand turn to the CTF and ISFSI storage pad areas. This route consists of an asphalt roadway. The transport route has a 2 percent transverse slope into the hill from the southeast entry outside the plant protected area and south along Plant View Road up to where the road joins the main plant access road. The main plant road has a 2 percent crown for about 50 to 100 ft until the Patton Cove Bypass Road. The Patton Cove Bypass Road will have a 2 percent transverse slope towards its radius until it joins Reservoir Road at which the transverse slope is 2 percent into the uphill side of the road. The transport route is built to AASHTO H-20 and HS-20 pressure ratings, except for the turntables as discussed below. The roadway capacity to withstand the transporter weight has been verified. The underground utilities and structures are evaluated and temporarily reinforced with steel plates, cribbing, and/or shoring as necessary to withstand the load from the loaded cask transporter. The transporter position on the road is controlled to ensure an adequate standoff distance is maintained from potential hazards. The following is a discussion of underground utilities along the transport route.

Underground utilities and related valve boxes, pull boxes, catch basins, concrete pipeways, and the retaining wall east of the DCPP Unit 2 CMS are rated for H-20 traffic loads. Administrative controls are established to preclude the transporter traversing the turntables that are located on the 115-ft Elevation. The turntables, used in the transfer of resin containers from the AB to the radwaste storage building, are only rated for 30 tons. Most pipes and conduits are buried 3 ft deep, except for utilities installed during the plant construction period and ground grid which are shallower, generally 1.5-ft deep. Pipes and conduits are generally nonmetallic, for example, asbestos-cement or polyvinyl chloride (PVC). Firewater line fittings are of ductile or cast iron. Valves are most commonly metallic.

None of the water lines or drains to be crossed are safety related for the 10 CFR 50 power plant. Firewater lines are 10 CFR 50 nonsafety-related, but they are subject to prescribed quality assurance requirements. Radwaste and makeup water lines in the RCA are encased in concrete. 10 CFR 50 safety-related or nonsafety-related circuitry passing beneath the route are contained in plastic conduits and are protected by a concrete cap or encasement.

Inside the RCA, the cask transporter will cross: makeup water; radwaste drainage; firewater; storm drains and pipeway drains; hydrogen and nitrogen gas lines in pipeways; electrical, lighting, and security system conduits and grounding; related concrete pipeways, valve boxes and pullboxes; and embedded rails.

From the Elevation 115-ft bench to the protected area (PA) gate near the Unit 2 CMS, utilities that cross the path include: 12-kV conduits near the road to the main warehouse; a drainage pipeway near the access road to the warehouse; domestic water and sanitary sewer lines near the CMS; and electrical and security conduits near the PA fence.

Along Plant View Road, from the PA to Area 10, and along Shore Cliff Road to Warehouse B, utilities include: PVC domestic water lines and asbestos cement pipe

(ACP) raw water lines that run along the edge of the road; shallow steel water lines that cross the road near the south end of Building 201; electrical and telephone lines that run along the other lane; culverts; firewater lines, electrical and telecommunications conduits run from the Area 10 intersection near Warehouse B.

Utilities on Reservoir Road include: an ACP raw water pipeline, a fiberglass seawater reverse osmosis permeate pipeline with combination air valves, and electrical and telecommunications conduits. Abandoned sanitary sewer lines cross the road near the leach field. Culverts cross the road at various locations.

As the transporter ascends the hill along Reservoir Road, it passes beneath the Unit 2 500-kV transmission lines, which are approximately 55 ft above the road surface. To ensure there remains an electrically safe working distance between the transporter and the transmission lines, the maximum height of the lifting beam on the transporter will be administratively controlled in accordance with plant procedures.

4.3.4 DESIGN BASES AND SAFETY ASSURANCE

The design criteria and associated design bases for the transporter are presented in Section 3.3.3. The components of the transportation system in the direct load support path while the load is suspended (lifting points) are considered important to safety. The design and construction of important-to-safety items are conducted under an approved 10 CFR 72 quality assurance program. The design approach to classify certain load path members as important to safety with enhanced safety factors is taken to render all hypothetical transfer cask and overpack drop events outside the FHB/AB not credible. Section 8.2.4 describes this approach in more detail. As a defense-in-depth measure, however, the transportation system design and administrative controls are such that the transfer cask and overpack will be lifted only to those heights necessary for cask handling operations. These transporter design bases and administrative controls are in compliance with 10 CFR 72.128 (a) with regard to ensuring adequate safety under normal and accident conditions.

4.3.5 REFERENCES

1. 10 CFR 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

2. ANSI N14.6, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More, American National Standards Institute, 1993 Edition.

3. Control of Heavy Loads at Nuclear Power Plants, USNRC NUREG-0612, July 1980.

4. Deleted in Revision 2.

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6. Holtec International Report No. HI-2002501, “Functional Specification for the Diablo Canyon Cask Transporter,” Revision 8.

7. PG&E Letter DIL-03-015 to the NRC, Additional Information on Cask Transfer Facility Cask Transporter Lateral Restraint System, December 4, 2003.

8. Holtec International Report No. HI-2053390, “Structural Evaluation of the Low Profile Transporter,” Revision 4.

9. Holtec International Report HI- 992234, “Stress Analysis of MPC Lift Cleat,” Revision 5

4.4 OPERATING SYSTEMS

4.4.1 LOADING AND UNLOADING SYSTEM

The dry storage cask handling systems are provided to lift, move, handle, and otherwise prepare an MPC loaded with DCPP spent fuel for storage at the Diablo Canyon ISFSI. Equipment is also available to unload an MPC in the unlikely event this becomes necessary. This section provides an overview of the functions and design of the equipment used to deploy the HI-STORM 100 System at the Diablo Canyon ISFSI for normal, off-normal, and accident conditions. Regulatory Guide 3.62 uses the term “emergency conditions.” This FSAR uses the term “accident conditions” for consistency with more recent regulatory guidance (that is, NUREG-1567). Movement of spent fuel assemblies between the spent fuel racks and the MPC is conducted in accordance with existing plant equipment and procedures, which are modified, as necessary, to meet handling requirements and commitments as described in the DCPP 10 CFR 50 LAR and license amendments (References 11 and 12, respectively), and is not specifically addressed here. Chapter 5 provides detailed operating guidance regarding use of the structures, systems, and components to perform the various cask-handling activities.

Personnel radiation exposures occurring as a result of dry storage operations are planned and monitored in accordance with the DCPP radiation protection program (Section 7.1).

4.4.1.1 Function

The function of the loading system is to safely accomplish the following major objectives while maintaining occupational doses ALARA:

- Place the empty MPC and HI-TRAC transfer cask into the DCPP SFP using the FHB/AB crane.
- Load the MPC using 10 CFR 50 fuel handling equipment.
- After fuel loading, place the MPC lid on the MPC.
- Remove the loaded MPC and transfer cask from the SFP and place the assemblage down in the cask washdown area in the FHB/AB.
- Weld the MPC lid to the MPC shell.
- Helium leak test the MPC.
- Drain, dry, and backfill the MPC with helium.
- Weld the vent and drain port cover plates and closure ring to the MPC lid and shell.

- Install the transfer cask top lid.
- Lift and place the loaded transfer cask onto the low profile transporter (LPT).
- Move the loaded transfer cask out of the FHB/AB vertically.
- Move the loaded transfer cask from just outside the FHB/AB to the CTF using the cask transporter.
- Pre-stage an empty HI-STORM 100SA overpack for MPC transfer at the CTF.
- Place the loaded transfer cask atop the empty overpack at the CTF using the cask transporter.
- Transfer the loaded MPC from the transfer cask to the overpack.
- Remove the empty transfer cask and place it in its designated storage area.
- Install the overpack lid.
- Move the loaded overpack to a storage pad using the cask transporter and place it in its designated position.

The same lifting and handling equipment is used in reverse order to return the loaded MPC to the cask washdown area in the FHB/AB in the unlikely event that an MPC needs to be unloaded. Loading and unloading operations are summarized below, including descriptions of the equipment used in performing these operations.

4.4.1.2 Major Components and Operating Characteristics

Detailed operational guidance is provided in Section 5.1. The following discussion provides an overview of the cask loading and unloading operations, including normal, off-normal, and accident conditions.

4.4.1.2.1 Component Arrival and Movement to the Preparation Area

The MPC is a cylindrical, stainless steel pressure vessel containing an internal honeycomb fuel basket that is designed to house the spent fuel assemblies chosen for storage at the Diablo Canyon ISFSI. The nominal thicknesses of the MPC shell, lid, and baseplate are 1/2 inch, 9-1/2 inches, and 2-1/2 inches, respectively. See Section 4.2.3.2.1 for detailed description of the MPC.

The MPC is shipped to the DCPP site with the fuel basket having been installed at the fabrication facility. Upon arrival at the site, the MPC is removed from the delivery vehicle, receipt inspected, and cleaned, as necessary, prior to being declared ready for installation into the transfer cask.

The transfer cask is used to lift and move the MPC located inside it. It is used both before and after the MPC has been loaded with spent fuel assemblies. The transfer cask is designed to provide radiation shielding while maintaining the total weight of the loaded MPC and transfer cask within the load rating of the FHB crane (125 tons). The 125-ton transfer cask design includes a nominal 3/4-inch thick inner shell and a 1-inch thick outer shell, both made of carbon steel. The approximately 4-1/2 inch wide annulus between the inner and outer shells is filled with lead for gamma shielding. A water jacket attached to the outer shell provides a radial dimension of approximately 5.4 inches of water for neutron shielding. The top lid is composed of 2 carbon steel plates with a combined thickness of approximately 1-1/2 inches. Between the plates are 3-1/4 inches of Holtite neutron shielding material. The bottom lid is composed of two carbon steel plates with a combined thickness of approximately 3 inches. Between these plates are 2-1/2 inches of lead. The bottom lid also includes a drain to remove water during preparation activities. The top lid is bolted to allow reuse and has a nominal 27-inch diameter hole in the center. This hole and the bolted connection of the bottom lid allow raising and lowering of the loaded MPC during transfer operations between the overpack and the transfer cask, as described below. The transfer cask is designed for repetitive, transient use to facilitate the movement of the MPC between the overpack and the SFP. All surfaces exposed to the SFP water are coated with coatings compatible with the SFP water chemistry and any uncoated items are compatible with the SFP water chemistry. The Holtec proprietary drawings for the original HI-TRAC 125D transfer cask design that was to be used at the Diablo Canyon ISFSI have been provided to the NRC (Reference 1) and non-proprietary drawings for the standard HI-TRAC 125D are included in Section 1.5 of Revision 1A to the HI-STORM 100 System FSAR (Reference 2). Section 4.2.3.2.4 describes the modified version of the HI-TRAC 125 transfer cask to be used for Diablo Canyon ISFSI operations. Figure 4.2-8 shows this modified design of the HI-TRAC 125D assembly to be used at the Diablo Canyon ISFSI.

Like the MPC, upon arrival onsite, the transfer cask is removed from the delivery vehicle, inspected, cleaned as necessary, and upended to the vertical position with a lifting device such as a mobile crane. The bottom lid is bolted to the bottom flange and the transfer cask is declared ready for use. The transfer cask lid is removed and the empty MPC is lifted and placed inside the transfer cask using the four lift lugs welded to the inside of the shell. The combined empty MPC and transfer cask assemblage is then attached to the LPT and moved into the FHB/AB through the roll-up door on the east side of the building. The empty MPC may also be loaded in the FHB/AB. All of the outdoor lifts of nonfuel bearing components are performed with suitably designed, commercial-grade lifting and rigging equipment or the transporter.

A lift yoke, custom designed for compatibility with the DCPP FHB/AB crane, the transfer-cask lifting trunnions and SFP water chemistry, is used to lift the transfer cask/MPC for the fuel loading operations.

4.4.1.2.2 Cask Preparation and Fuel Loading

Once in the FHB/AB, the transfer cask with the empty MPC inside, is moved to the Unit 2 cask washdown area. While in the FHB/AB, the transfer cask is restrained to preclude an unanalyzed tip-over. The cask is secured in the Unit 2 cask washdown area seismic restraint structure, the lift yoke is disconnected from the cask, and the cask is prepared for movement to the SFP.

The MPC is then filled with water of the proper boron concentration, as required by the Diablo Canyon ISFSI Technical Specifications (TS). The annulus overpressure system is attached. The annulus between the transfer cask and the MPC is filled with uncontaminated, demineralized water and an inflatable annulus seal is installed to prevent contamination of the outer MPC shell while it is submerged in the SFP. The annulus overpressure system is a defense-in-depth measure to ensure that any breach of the annulus seal or bottom lid seal will force leakage of clean water into the SFP, and not contaminated SFP water into the annulus. The lift yoke is reconnected and the transfer cask is lifted above and traversed over the SFP wall by the crane into position over the cask recess area of the SFP. The FHB crane lowers the cask into the SFP transfer cask restraint.

The lift yoke is disconnected and the selected fuel assemblies are loaded into the MPC in accordance with plant procedures.

The drain line is attached to the MPC lid and, after fuel loading is complete, the MPC lid is lowered into position on top of the MPC lift lugs. The lift yoke is attached to the transfer cask, the cask is lifted out of the SFP, and the annulus overpressure system is disconnected.

4.4.1.2.3 MPC and HI-TRAC Preparation for Storage

The loaded transfer cask and MPC are lowered to the Unit 2 cask washdown area inside a seismic restraint structure and the cask is decontaminated.

The water level in the MPC is lowered slightly, and the MPC lid is welded to the MPC shell using the automated welding system (AWS) augmented by manual welding as necessary. Liquid penetrant (PT) examinations will be performed on the root and final weld layers, and each approximately 3/8-inch of weld depth. For the MPC-24, -24E, and –32, which have a 3/4-inch deep MPC lid-to-shell weld, this will require one or two intermediate PT examinations. For the MPC-24EF, which has a 1-1/4 inch deep lid-to-shell weld, four intermediate PTs will be required. The examinations are performed in accordance with the commitments in the HI-STORM 100 System FSAR.

After MPC-lid welding, the water in the MPC is raised again and a hydrostatic test is performed. Upon successful hydrostatic test completion, the MPC is completely drained of water using the MPC blowdown system. The MPC to transfer cask annulus is drained of water, and the last of the water is removed from the MPC through the use of a forced helium dehydration (FHD) system. The design criteria for the FHD system are provided in Section 10.2. The Diablo Canyon ISFSI TS program controls and Section 10.2 specify the dryness acceptance criteria. After meeting the drying acceptance criteria, the MPC is backfilled with 99.995 percent pure helium to within a pressure range defined by Section 10.2.

When the MPC has been satisfactorily drained, dried, backfilled with helium, if the MPC contains any high burnup fuel assemblies (> 45,000 MWD/MTU) and temporary shielding is utilized on the transfer cask, the supplemental cooling system (SCS) is placed in service by filling the transfer cask annulus and associated keep-full tank with demineralized water. The MPC vent and drain port cover plates are welded on, inspected, and helium leak tested in accordance with the commitments in the HI-STORM 100 System FSAR, including ANSI N14.5-1997 (Reference 3). Then, the MPC closure ring is welded in place and inspected in accordance with the HI-STORM 100 System FSAR. The inner diameter of the closure ring is welded to the MPC lid and the outer diameter is welded to the top of the MPC shell.

The transfer cask top lid is installed, the SCS, if in use, is removed, and the cask is released from the cask washdown area seismic restraint structure. The transfer cask is then lifted by the single failure proof FHB/AB crane. The height to which the transfer cask is lifted is carefully controlled to that needed to move the cask onto the LPT.

The transfer cask is then moved laterally to the LPT and attached to the LPT baseplate. The LPT with the loaded transfer cask is moved out of the FHB/AB on removable tracks to a position just outside the FHB/AB door. The transfer cask is positioned under the lift beam of the cask transporter and the transfer cask lift links are connected to the cask. The transfer cask is unbolted from the LPT, then raised and secured within the transporter for the trip to the CTF.

4.4.1.2.4 MPC Transfer and Overpack Storage at the ISFSI

Outside the FHB/AB, the loaded transfer cask is rigged to the cask transporter and moved to the CTF in the vertical position. These evolutions and the cask transport system design, including associated lifting components, are described in more detail in Sections 4.3 and 5.1. The design of the CTF is discussed in Section 4.4.5.

At the CTF, the empty overpack is prestaged in the subterranean vault with approximately the top 3 ft of the overpack extending above grade level. At this stage of the loading process, the overpack is supported by the CTF baseplate and fitted with a cask-mating device. When the cask transporter arrives at the CTF, it moves the transfer cask over the overpack, the annulus is drained, and the transfer cask is placed atop the cask mating device on the overpack.

After the transfer cask is placed atop the overpack, the MPC lift cleats are installed. The MPC downloader and MPC lift slings are used to lift the MPC by the lift cleats just enough to take the weight of the MPC off the transfer cask bottom lid. The MPC downloader system is integral to the cask transporter and is located on the bottom flange of the horizontal lift beam of the cask transporter. Once the weight of the loaded MPC is taken off the bottom lid, the bottom lid is unbolted and the cask-mating device is used to remove the lid, creating a clear path between the transfer cask and the overpack. The MPC is then lowered into the overpack using the MPC downloader slings and the slings are lowered onto the top of the MPC. The transfer cask is removed from the top of the overpack and placed out of the way, allowing the downloader slings and MPC lift cleats to be removed. The overpack lid is installed and the overpack is transported to the storage pad using the cask transporter.

4.4.1.2.5 Off-Normal and Accident Conditions

For off-normal and accident conditions, the necessary response is a function of the nature of the event. Chapter 8 describes the off-normal and accident events for which the cask system is designed and provides suggested corrective actions. The HI-STORM 100 System is designed to maintain confinement integrity under all design-basis, off-normal, and accident conditions, including natural phenomena and drop events. For Diablo Canyon, cask drops inside the FHB/AB are not considered credible since the FHB/AB crane is single failure proof in accordance with the criteria of NUREG-0612; cask drops outside the FHB/AB are not considered credible since the transporter is single failure proof in accordance with the criteria of NUREG-0612. Cask tipover events are precluded during transport of the loaded cask while on the LPT through the design of the LPT. Based on the circumstances of an actual event, plant personnel will take appropriate action ranging from inspections of the affected cask components to movement of the cask back into the SFP and unloading of the spent fuel assemblies.

4.4.1.2.6 Unloading Operations

To unload a HI-STORM 100 System, the loading operations are essentially performed in reverse order, using the same lifting and handling equipment. Should any MPCs loaded under Amendment 2 of this license require unloading, implementation of the supplemental cooling system is required. Once the transfer cask is returned to the cask washdown area in the FHB/AB, the MPC closure ring and vent and drain port cover plates are removed by cutting their attachment welds. Fuel cooldown is performed, if necessary, using the vent and drain and the helium cooldown system until the helium temperature is reduced to the maximum temperature specified in Section 10.2. Helium cooldown is required prior to reflooding the MPC with water (borated as necessary) to prevent flashing of the water and the associated pressure excursions. Once the fuel is sufficiently cool, the MPC is flooded with borated water and the lid weld is removed using the weld removal system. Then, the transfer cask and MPC are lowered into the

SFP using the lift yoke and FHB crane. Finally, the MPC lid is removed, and the fuel assemblies are transferred from the MPC to the spent fuel racks.

4.4.1.3 Safety Considerations and Controls

The MPC shell is designed in accordance with ASME Section III (Reference 4), Subsection NB. The MPC fuel basket is designed in accordance with ASME Section III, Subsection NG. As discussed in Reference 4, the MPC is designed to retain its confinement boundary integrity under all normal, off-normal, and accident conditions. The MPC is a fully welded vessel that does not require the use of mechanical seals or leakage monitoring systems. The cask system is completely passive by design and does not require the operability of any supporting systems to safely store the spent nuclear fuel at the ISFSI storage pads. The design features that ensure safe handling of the fuel are described in Section 4.4.1.2 and the ISFSI operations are provided in Section 5.1.

The transfer cask and overpack steel structures are designed in accordance with ASME Section III, Subsection NF with some of the NRC-approved Code exceptions applicable to DCPP (Table 3.4-6). Both the transfer cask and the overpack are designed to withstand the design-basis normal, off-normal, and accident loadings (including natural phenomena) for the Diablo Canyon ISFSI site. The transfer cask design includes shielding design features that keep dose rates ALARA during fuel loading operation and transport of the loaded cask to the storage pads.

The transfer cask shielding is optimized to provide the maximum practicable protection from radiation while staying within the size and weight limits necessary for compatibility with the DCPP facility and the capacity of the FHB/AB crane. Additionally, the design of the transfer cask includes as few pockets and crevices as practicable in the design to minimize the amount of radioactive crud that could be retained in these areas. The paint on the transfer cask is suitable for ready decontamination and removal of loose particles through the use of a standard decontamination practices. The overpack provides the maximum shielding possible while keeping the cask at a reasonable size and weight, compatible with commercially available crawler vehicles. Details of the transfer cask and overpack shielding design features are provided in Chapter 5 of the HI-STORM 100 System FSAR and Section 7.3.1 of this FSAR.

4.4.1.3.1 Considerations Inside the 10 CFR 50 Facility

NUREG-0612 provides guidelines to licensees to ensure the safe handling of heavy loads. The guidelines define acceptable alternatives for heavy load movements, which include using a single failure proof handling system or analyzing the effects of a load drop.

Inside the FHB/AB, the cask and any ancillary components are lifted, handled, and moved in accordance with DCPP procedures and the DCPP Control of Heavy Loads Program for lifting heavy loads, as applicable. The FHB/AB crane hoist is used with a lift yoke to perform all lifts of the cask inside the FHB/AB. The transfer-cask-lifting trunnions and the lift yoke are designed, fabricated, inspected, maintained, and tested in accordance with NUREG-0612 to ensure that structural failures of these items are not credible. PG&E's Control of Heavy Loads Program controls the design of special lifting devices in accordance with ANSI N14.6 (Reference 5). This program is fully described in DCPP FSAR Update, Section 9.1.4.3.5. The existing FHB/AB crane has been upgraded to be single-failure proof, as defined in NUREG-0612, Section 5.1.2 (Reference 6). Therefore, cask drops inside the FHB/AB when using the FHB/AB crane are not considered credible.

In the Unit 2 cask washdown area, a seismic restraint structure secures the transfer cask while preparing the transfer cask and empty MPC for fuel loading in the SFP, preparing the loaded MPC and transfer cask for transport to the CTF, and, if necessary, preparing the transfer cask and loaded MPC for fuel unloading in the SFP. The seismic restraint structure is located in the corner of the cask washdown area and consists of a wall mounted platform with a restraining strap and a floor mounted restraining plate. The capability of the seismic restraint structure to prevent tip-over or damage to the HI-TRAC during postulated seismic events is demonstrated in analyses provided in Holtec Report HI-2063593 (Reference 15).

Cask tipover events are precluded during transport of the loaded cask while on the LPT through the design of the LPT. A structural analysis of the cask on the LPT during seismic events was performed in Holtec Report HI-2053390 (Reference 16), which measured the peak displacements of the top of the cask relative to the ground. The analysis results show that overturning of the loaded cask during seismic events is not credible. During a seismic event the maximum potential longitudinal sliding along the tracks for a loaded LPT is on the order of 45 inches. The LPT and loaded transfer cask would not experience accelerations greater than the 45 g design basis limit during any sliding event.

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

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

When high-burnup fuel is loaded in the MPC, and temporary shielding is being utilized on the transfer cask, the SCS is required to maintain cladding temperatures within limits, following draining of the MPC, and when the MPC is not being recirculated by the forced helium dehydration system. The SCS may be out of service for short periods of time (per the associated TS) to perform necessary evolutions. A loss of supplemental cooling is evaluated as an accident in Chapter 8.

4.4.1.3.2 Considerations Outside the 10 CFR 50 Facility

Cask drop events are precluded during transport of the loaded cask from the FHB/AB to the CTF, and from the CTF to the storage pad, through the design of the cask transport system, including the cask transporter (Section 4.3). Drop events are precluded by lift devices designed, fabricated, operated, inspected, maintained, and tested in accordance with NUREG-0612. The cask transport system is designed in accordance with these requirements and appropriate design codes and standards to preclude drop events on the transport route. The cask transporter is also designed to withstand applicable, site-design-basis natural phenomena, such as seismic events and tornadoes, without dropping the load or leaving the transport route. The load-path parts of the cask transporter are designed as specified in Section 4.3.2.1. The cask transporter is procured commercial grade and is qualified by functional testing prior to service for MPC and overpack transfer operations at the CTF. Uncontrolled movement of the cask transporter is prevented by the automatic drive brake system and emergency stop and deadman design features, as discussed in Section 4.3.2.1.2; these components also are procured commercial grade and are qualified by functional testing prior to service.

4.4.2 DECONTAMINATION SYSTEM

Standard decontamination methods are used to remove surface contamination, to the extent practicable, from the transfer cask and accessible portions of the MPC (that is, the lid) resulting from their submersion in the SFP. The cask and MPC lid are rinsed with clean water while over the SFP. Final decontamination of the transfer cask and MPC lid is performed in the cask washdown area in the FHB/AB. Decontamination is typically performed manually. While the entire MPC is submerged in the SFP during fuel loading, the annulus seal and annulus overpressure system prevent contaminated water from coming in contact with the sides of the MPC, leaving the MPC lid as the only exterior surface of the HI-STORM 100 System at the ISFSI storage pad that has been exposed to SFP water.

4.4.3 STORAGE CASK REPAIR AND MAINTENANCE

Chapter 9 of the HI-STORM 100 System FSAR describes the required maintenance for the storage cask system. The HI-STORM 100 System is totally passive by design. There are no active components or monitoring systems required to ensure the performance of its safety functions in the final storage configuration. As a result, only minimal maintenance is required over its lifetime, and this maintenance primarily results from cask handling and weathering effects in storage. Typical of such maintenance is the reapplication of corrosion inhibiting materials on accessible external surfaces. Visual inspection of the overpack inlet and outlet air duct perforated plates (screens) is required by the Diablo Canyon ISFSI TS to ensure that they are free from obstruction, including clearing of debris, if necessary.

Repairs and maintenance are performed by maintenance personnel either in-situ or in another appropriate location, based on the nature of the work to be performed. Radiation protection personnel provide input to and monitor as necessary these maintenance work activities through the work control process.

4.4.3.1 Structural and Pressure Parts

PG&E anticipates that it will use a cask loading campaign where multiple storage casks are loaded in an essentially continuous work effort. Prior to each transfer cask fuel loading, a visual examination is performed on the transfer-cask-lifting trunnions. The examination consists of inspections for indications of overstress such as cracking, deformation, or wear marks. Repair or replacement is required if unacceptable conditions are identified. The transfer-cask trunnions are maintained and inspected in accordance with ANSI 14.6.

As described in the HI-STORM 100 System FSAR, Chapters 7 and 11, there are no credible normal, off-normal, or accident events that can cause the structural failure of the MPC. Therefore, periodic structural or pressure tests on the MPCs, following the initial acceptance tests, are not required as part of the storage maintenance program.

4.4.3.2 Leakage Tests

There are no seals or gaskets used on the fully welded MPC confinement system. Therefore, confinement boundary leakage testing is not required as part of the storage system maintenance program.

4.4.3.3 Subsystem Maintenance

The HI-STORM 100 System does not include any subsystems that provide auxiliary cooling in its final storage configuration. Normal maintenance and calibration testing is required on the forced helium drying, helium backfill, recirculation and cooldown, supplemental cooling, and leakage testing systems. Rigging, remote welders, cranes, and lifting beams are inspected prior to each loading campaign to ensure this equipment is ready for service.

4.4.3.4 Pressure Relief Valves

The pressure relief valves used on the water jacket for the transfer cask require calibration on an annual basis (or prior to the next transfer cask use if the period the transfer cask is out of use exceeds 1 year) to ensure the pressure relief setting is within tolerance as controlled by PG&E's DCPP Maintenance Program.

4.4.3.5 Shielding

The gamma and neutron shielding materials in the overpack, transfer cask, and MPC degrade negligibly over time or as a result of usage. Radiation monitoring of the ISFSI provides ongoing evidence and confirmation of shielding integrity and performance. If the monitoring program indicates increased radiation doses, additional surveys of the overpacks are performed to determine the cause of the increased dose rates.

The Boral or Metamic panels installed in the MPC baskets are not expected to degrade. The use of Boral or Metamic as the fixed neutron absorber is discussed in Section 4.2.3.3.5. Therefore, no periodic verification testing of neutron poison material is required on the HI-STORM 100 System.

4.4.3.6 Thermal Performance

In order to ensure that the HI-STORM 100 System continues to provide effective thermal performance during storage operations, surveillance of the passive heat removal system is performed in accordance with the Diablo Canyon ISFSI TS. This involves a periodic inspection to verify that the air duct perforated plates (screens) are not blocked.

4.4.4 UTILITY SUPPLIES AND SYSTEMS

Electric power is provided for CTF and storage-pad-area lighting, and the storage-pad-area security system. As the HI-STORM 100 System is a passive system, no other utilities are required for ISFSI operation. As a minimum, electrical and pneumatic power supplies are provided for the mating device hydraulic and airbag systems.

4.4.4.1 Electrical Systems

Electric power is required to support functions of the mating device hydraulic and airbag systems. Normal power is supplied from the nonsafety-related 12-kV distribution system for the CTF and the storage-pad-area normal lighting. Power for the storage-pad-area security equipment is provided by the DCPP security power system. There are no motorized fans, dampers, louvers, valves, electronic monitoring systems, and no electrically operated cranes. Electric and pneumatic power supplies are provided for the mating device hydraulic and airbag systems. In the event of a loss of power, power will not be supplied to the ISFSI components, except for the security loads. A discussion of the normal and emergency power for security equipment is provided in the Physical Security Plan (Section 9.6). Section 8.1.6 describes recovery actions to mitigate a loss of power event.

4.4.4.1.1 Normal Power Supplies

The existing DCPP 12-kV distribution system is connected to the DCPP power distribution system in the existing DCPP 12-kV startup buses. Either DCPP Unit 1 or Unit 2 can supply the 12-kV system. The 12-kV underground distribution system is connected to the 12-kV startup bus by existing 3-way switches. The existing 12-kV underground distribution system is routed throughout the DCPP site, including near the location of the CTF and ISFSI storage pad area. A combination of new and existing switches and 12-kV/480-V transformers are used to connect the CTF and ISFSI storage-pad-area loads.

4.4.4.1.2 Grounding

The ISFSI storage pad area, perimeter fencing, lighting and poles, and security equipment are located below the DCPP Unit 1 500-kV transmission lines. The existing DCPP station-to-switchyard ground grid below the ISFSI location is maintained. The ISFSI area is provided with a ground grid, and is connected to the station-to-switchyard ground grid. Each storage cask is grounded to the ISFSI-area ground grid.

4.4.5 CASK TRANSFER FACILITY

The design criteria for the CTF are provided in Section 3.3.4. Holtec CTF drawing 4480 is provided in Figure 4.4-3. The site-specific structural details of the CTF design and analysis for the Diablo Canyon ISFSI are provided in Section 4.2.1.2. The mechanical design aspects are discussed below.

4.4.5.1 CTF Function

The function of the CTF is to facilitate transfer of a loaded MPC between the transfer cask and the overpack. These operations are discussed in Sections 4.4.1.2.4 and 5.1.1.3.

4.4.5.2 CTF Design

Design criteria for the CTF are provided in Section 3.3.4 of this FSAR, Section 2.3.3.1 of the HI-STORM 100 System FSAR, and in Reference 7. The CTF is used in conjunction with the Diablo Canyon cask transporter to permit MPC transfers between the transfer cask and the overpack. The CTF is designed to position an overpack sufficiently below grade where the transfer cask can be mated to the overpack using a cask transporter and a suitably designed mating device.

The analysis of the concrete structure housing the cask transfer facility (CTF) utilized the static and dynamic rock properties provided in References 8 and 9. Reference 8 provides the ultimate and design allowable values for lateral resistance of the rock while Reference 9 provides the ultimate and design allowable values for bearing on the rock at the bottom of the CTF.

4.4.5.2.1 Functional/Technical Requirements

The CTF is designed to operate in conjunction with the cask transporter. Together, the cask transporter and the CTF designs ensure that there will be no uncontrolled lowering of the lifted load under all design-basis conditions of service, including environmental phenomena.

4.4.5.2.2 Main Shell

As discussed in Sections 4.2.1.2 and 4.4.5.2.3.3, the main shell is equipped with a sump to collect water from the CTF cylinder.

4.4.5.2.3 Sump

During periods of nonusage, the CTF has a cover installed to prevent water entry. To collect any accumulated water, the CTF is equipped with a sump. Any sump water is collected, sampled for radioactivity, and processed in accordance with applicable administrative procedures.

4.4.5.3 CTF Analysis

The load path parts of the CTF are conservatively designed in accordance with the ASME Code, Section III, Subsection NF. The CTF was purchased ITS-B and is qualified for MPC and overpack transfer operations.

Analyses have been performed to verify that, during MPC transfer from the HI-TRAC to the HI-STORM overpack, the main shell of the CTF and its surrounding foundation are sufficient to maintain the overpack in the vertical position.

There are no impact factors considered in the CTF analysis. After the empty overpack is positioned in the CTF, any radial gaps between the CTF shell and the body of the overpack just below the top of the CTF are closed to the extent practical by adding metallic wedge assemblies at the top of the CTF shell. Small gaps may still remain even after the addition of wedge assemblies to close the gap. These very small gaps (compared to the scale of the structure) may give rise to high frequency impact forces upon contact. However, since the structural analysis for Code qualification focuses on the response to low frequency loads from a seismic loading, any high frequency impact loads arising from the existence of any remaining very small gaps after wedge assembly installation have been omitted.

After MPC transfer, the HI-TRAC transfer cask and mating device are removed, the lid is installed on the loaded overpack and the wedge assemblies are removed. The actual time between MPC transfer into the overpack and raising the overpack out of the CTF is expected to be less than an operating shift, or 8 hours. With the CTF wedge assemblies in place between the loaded overpack and the CTF walls, there is still some convective heat transfer through the overpack, albeit not at a rate commensurate with the conditions on the ISFSI pad. The thermal analyses (HI-2053376, Reference 17, HI-2104625, Reference 18 and HI-2125191, Reference 19) demonstrate that the overpack and MPC can remain in the CTF indefinitely and fuel cladding temperature limits for long term storage are not exceeded.

The analysis performed in Holtec Report HI-2053370 (Reference 10) evaluates the CTF under design basis loads. Loadings involving seismic events were considered only for the longer duration scenario when the loaded stack was supported by the base of the CTF. In this configuration, the lowest frequencies are associated with lateral bending of the stacked configuration as a beam-like structure. The vertical frequency of the stacked casks is in the rigid range, so no amplifier is used for vertical loads when the system is resting on the base of the CTF.

4.4.6 REFERENCES

1. Submittal of Holtec Proprietary Design Drawing Packages, PG&E Letter to the NRC, DIL-01-008, December 21, 2001.

2. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 1A, January 2003.

3. ANSI N14.5, Leakage Tests on Packages for Shipment, American National Standards Institute, 1997 Edition.

4. Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, American Society of Mechanical Engineers, 1995 Edition including 1996 and 1997 addenda.

5. ANSI N14.6, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More, American National Standards Institute, 1993 Edition.

6. Control of Heavy Loads at Nuclear Power Plants, NUREG-0612, USNRC, July 1980.

7. Holtec International Report No. HI-2002570, “Design Criteria Document for the Diablo Canyon Cask Transfer Facility,” Revision 5.

8. PG&E Calculation No. 52.27.100.716 (GEO.DCPP.01.06), “Development of Lateral Bearing Capacity for DPCP CTF Stability Analysis.”

9. PG&E Calculation No. 52.27.100.713 (GEO.DCPP.01.03), “Development of Allowable Bearing Capacity for DCPP ISFSI Pad and CTF Stability Analysis.”

10. Holtec International Report No. HI-2053370, “Structural Analysis of CTF at DCNP Under Design Basis Loads,” Revision 2.

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

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

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

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

15. Holtec International Report No. HI-2063593, “Dynamic Analysis of the HI-TRAC in Cask Washdown Area When Restrained,” Revision 5.

16. Holtec International Report No. HI-2053390, “Structural Evolution of the Low Profile Transporter,” Revision 4.

17. Holtec International Report No. HI-2053376, “Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System Design,” Revision 7.

18. Holtec International Report No. HI-2104625, “Three-Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System Design,” Revision 10.

19. Holtec International Report No. HI-2125191, “Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System with up to 28.74 kW Decay Heat,” Revision 6.

4.5 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

The structures, systems, and components (SSCs) comprising the Diablo Canyon ISFSI are classified as important to safety (ITS) or not important to safety (NITS). The criteria for selecting the classification for particular SSCs are based on the following definitions:

- Important to Safety

  A classification from 10 CFR 72.3 for any SSC whose function is to maintain the conditions required to safely store spent fuel, prevent damage to the spent fuel or spent fuel container during handling and storage, or 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.

- Not Important to Safety

  A classification for SSCs that do not meet the criteria for classification as ITS.

Major Diablo Canyon ISFSI SSCs are classified as ITS if at least one subcomponent comprising the major component is classified ITS. SSCs classified ITS are subject to the Quality Assurance (QA) Program described in Chapter 11. The importance to safety for each ITS SSC is further refined into three QA classification categories based on the guidance contained in NUREG/CR-6407 (Reference 1). The categories are intended to standardize the QA control applied to activities involving spent fuel storage systems. These classifications are defined as follows:

- Classification Category A – Critical to Safe Operation

  Category A items include SSCs 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 containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control.

- Classification Category B – Major Impact on Safety

  Category B items include SSCs 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 the failure of an additional item, could result in an unsafe condition.

- Classification Category C – Minor Impact on Safety

  Category C items include SSCs whose failure or malfunction would not significantly reduce the cask system effectiveness and would not be likely to create a situation adversely affecting public health and safety.

The major SSCs that are classified ITS are discussed in the following sections. A safety classification for these SSCs establishes the requirements that satisfy the 10 CFR 72.122(a) general design criteria, which specify that SSCs that are classified ITS be designed, fabricated, erected, and tested to quality standards. The safety classification of the subcomponents and the determination of the ITS category of each item is administratively controlled by PG&E via design and procurement control procedures with input from the storage cask vendor.

Table 4.5-1 lists the safety classification and QA Category for the major SSCs.

4.5.1 SPENT FUEL STORAGE CASK COMPONENTS

The major ITS components comprising the HI-STORM 100 System are described below with a brief description as to why each is classified as ITS. Table 4.5-1 lists the major storage cask components by QA Category, based on the highest QA category of any subcomponent comprising the major component.

4.5.1.1 Multi-Purpose Canister and Fuel Basket

The multi-purpose canister (MPC) is classified ITS because it serves as the primary confinement structure for the fuel assemblies and is designed to remain intact under all normal, off-normal, and accident conditions. The fuel basket inside the MPC is classified ITS, because it ensures the correct geometry of the stored fuel assemblies and provides the fixed neutron absorber between fuel cells to prevent criticality.

4.5.1.2 Damaged Fuel Container

The damaged fuel container (DFC) is classified ITS because it maintains fuel classified as damaged fuel or fuel debris in a safe geometry and enables retrieval of the damaged fuel assembly or fuel debris. The DFC also prevents the gross dispersal of particulates, including loose fuel pellets.

4.5.1.3 Overpack

The overpack is classified ITS because it is designed to remain intact under all normal, off-normal, and accident conditions and serves as the primary component for protecting the MPC during storage. It provides structural protection to prevent damage to the spent fuel and ensures fuel retrievability. It also provides radiation shielding and allows for MPC heat rejection to the environment.

4.5.1.4 HI-TRAC Transfer Cask

The transfer cask is classified ITS because it is designed to support the MPC during transfer and lift operations and to provide structural integrity, missile protection, radiation shielding, and MPC heat rejection during the short time it contains the loaded MPC.

4.5.2 CASK STORAGE PADS

The cask storage pads are classified ITS because they provide the necessary embedment for the anchored overpack to prevent sliding and tipover during a design basis seismic event.

4.5.3 CASK TRANSFER FACILITY

The cask transfer facility (CTF) is classified ITS because the load-bearing components prevent damage to the spent fuel and spent fuel storage cask system components during lifting and MPC transfer operations under all normal, off-normal, and accident conditions. The CTF was purchased ITS-B and qualified by functional testing prior to use.

4.5.4 CASK TRANSPORT SYSTEM

The cask transport system is classified ITS because the load-bearing components prevent damage to the spent fuel and spent fuel storage cask system components during transport, lifting, and MPC transfer operations under all normal, off-normal, and accident conditions. The cask transporter "dead-man" and emergency stop features and the setting brakes are classified as ITS. The transport system is designed to prevent uncontrolled lowering of the load. In addition, the cask transporter is designed to maintain stability on the transport route between the FHB/AB and the CTF, and between the CTF and the cask storage pads. The cask transporter is purchased commercial grade and qualified by functional testing prior to use.

4.5.5 ANCILLARY EQUIPMENT

Ancillary equipment is comprised of those SSCs, not described above, that are used to lift, handle, and move the cask and prepare the MPC for storage operations. Table 4.5-1 lists the major ancillary equipment. Any additional ancillary equipment not included on the list will be classified and categorized in accordance with the PG&E design and procurement control procedures with input from the storage cask vendor.

4.5.6 DESIGN CRITERIA FOR SSCs NOT IMPORTANT TO SAFETY

The design criteria for SSCs classified as NITS, but which have security or operational importance, are addressed in other sections of this FSAR (for example, security systems and portions of the cask transport system and ancillary equipment systems).

These SSCs are designed in accordance with applicable commercial codes and standards to ensure, where interfaces exist, that there is compatibility with SSCs that are ITS.

The Diablo Canyon ISFSI security system is classified as NITS because it does not have a design function directly related to the protection of public health and safety due to operation of the Diablo Canyon ISFSI. The primary function of the security system is to prevent and detect unauthorized access to the Diablo Canyon ISFSI. The Diablo Canyon ISFSI security system design meets the requirements of 10 CFR 72, Subpart H.

The electrical power system is classified as NITS because it is not ultimately relied upon to support a function necessary for the safe operation of the dry cask storage system. The HI-STORM 100 System is completely passive in design and requires no electric power to ensure safe, long-term storage of the spent nuclear fuel.

Portions of the cask transporter and ancillary equipment not having design functions directly related to protecting public health and safety, as defined by the ITS classification categories in Reference 1, are classified as NITS. Major NITS equipment of this type are provided in Table 4.5-1. New equipment and subcomponents of existing equipment not included in Table 4.5-1 will be classified in accordance with the PG&E administrative control process with input from the storage cask vendor.

4.5.7 REFERENCES

1. Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, USNRC, NUREG/CR-6407, February 1996.

4.6 DECOMMISSIONING PLAN

4.6.1 PRELIMINARY DECOMMISSIONING PLAN

Prior to the end of the Diablo Canyon ISFSI life, MPCs loaded with spent fuel will be transferred from storage overpacks into transportation casks and transported offsite. Since the MPCs are designed to meet Department of Energy guidance applicable to MPCs for storage, transport, and disposal of spent fuel, the fuel assemblies will remain sealed in the MPCs such that decontamination of the MPCs is not required. Following shipment of the MPCs offsite, the ISFSI will be decommissioned by identification and removal of any residual radioactive material, and performance of a final radiological survey. The specific methods and details of the Diablo Canyon ISFSI decommissioning will be included in a formal decommissioning plan that will be submitted to the NRC for review and approval prior to the commencement of decommissioning activities. The conceptual program for ISFSI decontamination and decommissioning is briefly described in this section.

4.6.2 FEATURES THAT FACILITATE DECONTAMINATION AND DECOMMISSIONING

The design features of the Diablo Canyon ISFSI provide for inherent ease and simplicity of decommissioning the ISFSI in conformance with 10 CFR 72.130.

4.6.3 COST OF DECOMMISSIONING AND FUNDING METHOD

10 CFR 72.30(b) requires that the proposed decommissioning plan include a decommissioning cost estimate, a funding plan, and a method of ensuring the availability of decommissioning funds.

The philosophy of operating the Diablo Canyon ISFSI is “start clean/stay clean.” Thus, the intention is to maintain the facility free of radiological contamination at all times. During the operational phase of the facility, all radioactive contamination will be removed, if possible, immediately upon its discovery.
A cost estimate for decommissioning was prepared, which includes the following assumptions:

- The overpacks will not have any interior or exterior radioactive surface contamination
- The ISFSI pad and cask transfer facility area will not be contaminated
- Radiological remediation of surficial soil will be performed in a limited area of approximately 1.2 acres adjacent to the footprint of the ISFSI pad (conservative assumption for cost estimating purposes)
- A final status survey will be performed; this will include a 100 percent survey of the concrete overpack surfaces, and a significant fraction of the ISFSI pad and the immediate area surrounding the pad

The assumptions regarding contamination will be verified by performing the necessary surveys. The cost estimate for decommissioning the ISFSI is part of the detailed cost estimate contained in the PG&E Decommissioning Funding Report submitted to the NRC (Reference 2), as required by 10 CFR 50.75(f)(1). The costs in the estimate are organized into 4 major scopes of work, including:

- Utilities and Structures Demolition – includes planning, decontamination (as needed), and removal of site ISFSI pad and associated features
- Soil Remediation – remediate residual radiologically contaminated soil to meet radiological cleanup criteria
- Final Site Survey – license termination surveys, independent surveys, and application for license termination
- Waste, Transportation, and Material Management – waste handling, packaging, transportation, and disposal fees

In addition to the direct costs associated with decontamination, demolition and restoration, the estimate also contains costs for PG&E's oversight staff, security, site operating costs, and the NRC (and NRC contractor). It is estimated that ISFSI demolition and site restoration will cost approximately $64.1 million (2017 dollars) including support costs and contingency– for the DECON alternative.

PG&E has established an external sinking trust fund account for decommissioning DCPP Units 1 and 2. As discussed in the Decommissioning Funding Report to the NRC (Reference 1), this account contains designated monies for decommissioning the Diablo Canyon ISFSI.

4.6.4 LONG-TERM LAND USE AND IRREVERSIBLE COMMITMENT OF RESOURCES

Following removal of all storage casks from the ISFSI and completion of decommissioning in accordance with NRC regulations the facility can be released for unrestricted use.

The security-related structures and the CTF could be dismantled and removed. The concrete storage pads and the concrete floor of the CTF could be sectioned and removed, or alternatively left in place. In either case, the storage pads and CTF areas could be covered with top soil and replanted with native vegetation; thus, returning the land to its original condition.

The long-term plan will be addressed further in the final decommissioning plan that will be submitted prior to ISFSI license termination.

4.6.5 RECORDKEEPING FOR DECOMMISSIONING

Records important to decommissioning, as required by 10 CFR 72.30(d), will be maintained until the ISFSI is released for unrestricted use. These records will be maintained at DCPP as part of the records management system.

4.6.6 REFERENCES

1. PG&E Letter DIL-18-019, Decommissioning Funding Plan, dated December 17, 2018.

2. PG&E Letter DCL-19-020, Decommissioning Funding Report for Diablo Canyon Power Plant, Units 1 and 2, dated March 26, 2019.

4.7 OPERATING ENVIRONMENT EVALUATION

In accordance with NRC Bulletin 96-04 and consistent with Interim Staff Guidance (ISG) 15 (References 1 and 2), a review of the potential for chemical, galvanic, or other reactions among the materials of the HI-STORM 100 dry storage system, its contents, and the operating environments, which may produce adverse reactions, has been performed.

4.7.1 MULTI-PURPOSE CANISTERS

The passive, non-cyclic nature of dry storage conditions does not subject the MPC to conditions that might lead to structural fatigue failure. Ambient temperature and insolation cycling during normal dry storage conditions and the resulting fluctuations in MPC thermal gradients and internal pressure is the only mechanism for fatigue. These low-stress, high-cycle conditions cannot lead to a fatigue failure of the MPC enclosure vessel or fuel basket structural materials, that are made from austenitic stainless steel, known as “Alloy X.” “Alloy X” is a fictitious stainless steel used in the design basis analyses of the MPC to ensure any of the permitted austenitic stainless steels used in MPC fabrication will be bounded by the analyses. (See Reference 6, HI-STORM 100 System FSAR, Section 1.2.1.1, for a detailed discussion of Alloy X.) A typical MPC construction material specification, ASME SA240-304 stainless steel, has a fatigue endurance limit well in excess of 20,000 psi. All other off-normal or postulated accident conditions are infrequent or one-time occurrences, which cannot produce fatigue failures. The MPC also uses materials that are not susceptible to brittle fracture.

The MPC enclosure vessel and fuel basket are in contact with air, helium, and spent fuel pool (SFP) water during various stages of use. The MPC is made entirely of stainless steel, except for the neutron absorbers, and an aluminum seal washer or port plug with thread protector in both the vent and drain port assemblies. An alternative vent and drain port plug configuration may be used, which does not contain aluminum washers. Aluminum heat conduction elements, offered as optional equipment in the HI-STORM 100 System generic MPC design (Section 1.2.1.1 of the HI-STORM FSAR), are not used in any of the MPCs deployed at the Diablo Canyon ISFSI. There is no significant chemical or galvanic reaction of stainless steel with air or helium. The aluminum seal washers used with the vent and drain port caps never are in contact with water, so combustible gas generation is not a concern. There are no coatings of any kind used in or on the MPC. The control of combustible gases generated by the interaction of the Boral or Metamic neutron absorber with the SFP water is discussed in Section 4.7.1.1.

The moisture in the MPC is removed during loading operations to a point where oxidizing liquids and gases are at insignificant levels. The MPC cavity is then backfilled with dry inert helium at the time of closure to maintain an atmosphere in the MPC that provides corrosion protection for the SNF cladding and MPC materials throughout the dry storage period. The specific limits on MPC moisture removal and helium backfilling are included in the technical specifications. Insofar as corrosion is a long-term time-dependent phenomenon, the inert gas environment in the MPC minimizes the incidence of corrosion during storage on the ISFSI to an insignificant amount.

4.7.1.1 Boral and Metamic Neutron Absorber

The Boral neutron absorber panels consist of a boron carbide powder-aluminum powder mixture sandwiched between two solid aluminum surfaces. The Metamic neutron absorber panel is an aluminum/ boron carbide metal matrix composite material. The corrosion-resistant characteristics of such materials for dry SNF storage canister applications, as well as the protection offered by these materials against other material degradation effects, are well established in the nuclear industry in both wet and dry spent fuel storage applications. The preservation of this non-corrosive atmosphere is assured by the inherent seal-worthiness of the MPC confinement boundary integrity (there are no gasket joints in the MPC).

The Boral or Metamic neutron absorber panels are submerged in borated water during fuel loading operations in the SFP, and during MPC lid welding and potential MPC lid cutting in the unlikely event the MPC needs to be unloaded. The aluminum in the as-manufactured Boral or Metamic panels reacts with water, producing hydrogen gas. Therefore, all Boral surfaces are pre-passivated or anodized before installation in the MPC to minimize the rate of hydrogen production and ensure a combustible concentration of hydrogen does not accumulate under the MPC lid prior to, or during MPC lid welding or cutting operations. Because of the composite nature of the Metamic, passivation is not required to minimize hydrogen generation.

Because the Boral or Metamic water reaction cannot be completely eliminated and the Boral or Metamic material in the MPC is under varying hydrostatic pressure levels (up to approximately 40 ft of water pressure during fuel loading or unloading in the SFP, and up to approximately 15 ft during lid welding or cutting), continued generation of limited quantities of hydrogen is possible. To address hydrogen generation from the Boral or Metamic water reaction, the operating procedures for the Diablo Canyon ISFSI include provisions to address combustible gas control in the MPC lid area, consistent with the controls discussed in Sections 8.1.5 and 8.3.3 of the HI-STORM 100 System FSAR (Reference 6), for loading and unloading operations, respectively.

4.7.2 HI-TRAC TRANSFER CASK

The HI-TRAC transfer cask is used in an air and borated water environment during the various stages of loading and unloading operations. The use of appropriate coatings and the controlled environment in which the transfer cask is used minimize damage due to direct exposure to corrosive chemicals that may be present during loading and unloading operations. The transfer cask is designed for repeated normal condition handling operations with high factors of safety, particularly for the lifting trunnions, to assure structural integrity. The resulting cyclic loading produces stresses that are well below the endurance limit of the trunnion material, and therefore, will not lead to a fatigue failure in the transfer cask. All other off-normal or postulated accident conditions are infrequent or one-time occurrences that do not contribute significantly to fatigue. In addition, the transfer cask utilizes materials that are not susceptible to brittle fracture during the lowest temperature permitted for loading.

The transient use and relatively low neutron fluence to which the transfer cask materials are subjected do not result in radiation embrittlement or degradation of the transfer cask's shielding materials that could impair its ability to perform its intended safety function. The transfer cask materials are selected for durability and wear resistance for their deployment.

The load-bearing portions of the transfer cask structure are fabricated from carbon steel. Other materials included in the transfer cask design are Holtite-A (in the top lid for neutron shielding); elemental lead (in the body and bottom lid for gamma shielding) and brass, bronze or stainless-steel appurtenances (pressure relief valves, drain tube, etc.). A complete description of materials is provided on the transfer cask drawing in Chapter 1 of the HI-STORM 100 System FSAR. The Holtite and lead shielding materials are completely enclosed by the welded steel construction of the transfer cask. Therefore, there will be no significant galvanic or chemical reactions between these shielding materials and the air or borated water. A detailed description of Holtite-A may be found in Section 1.2.1.3.2 of the HI-STORM 100 System FSAR.

The internal and external steel surfaces of the transfer cask, (except threaded plugs and holes, seal areas and trunnions) are sandblasted and coated with an epoxy-based coating system, qualified for borated water use, to preclude surface oxidation. Lid bolts are plated and the threaded holes in the top flange are plugged or sealed during water immersion to prevent borated water intrusion. The transfer cask coating system was chosen based on manufacturer's literature that confirms that the coatings are designed for use in the conditions that the transfer cask will experience. Table 4.7-1 provides the specific coatings to be used on the transfer cask. With the coating system in place, there is no significant galvanic or chemical interaction between the air or SFP water and the steel materials. Minor nicks and dings that may expose the underlying carbon steel are repaired by maintenance coating between uses of the transfer cask. The small size of any carbon steel exposed by the nicks and dings, the temporary nature of transfer cask use, the relatively short duration of exposure to borated water, and the coating repair maintenance program, combined, eliminate significant corrosion of the carbon steel as a concern.

In summary, significant chemical or galvanic reactions involving the transfer cask and the SFP water are not expected.

4.7.3 HI-STORM OVERPACK

The HI-STORM overpack is used only in an air environment during the various stages of loading and unloading operations. The overpack is never immersed in the SFP or any other source of water. It is subjected to the environment at the ISFSI, which includes saline water vapor and rain. The overpack consists of two concentric carbon steel cylinders separated by radial plates, with a carbon steel base plate and lid. The annulus between the two cylinders is filled with concrete. All exposed carbon steel surfaces of the overpack, including the anchor studs and nuts, are coated with an approved coating to prevent corrosion due to salinity or other airborne contaminants at the ISFSI. Table 4.7-1 provides the specific coatings to be used on the overpack. Concrete in the overpack body, lid, and pedestal is non-reinforced and completely encased in steel. Therefore, the potential of environmental-induced degradation in an oceanside environment such as the Diablo Canyon ISFSI, including spalling of concrete, are not possible for the overpack.

Under normal storage conditions, the bulk temperature of the overpack, because of its large thermal inertia, changes very gradually with time. Therefore, material degradation from rapid thermal ramping conditions is not credible for the overpack. As discussed in Appendix 1.D of the HI-STORM 100 System FSAR, the aggregates, cement and water used in the storage cask concrete are carefully controlled to provide high durability and resistance to temperature effects. The configuration of the storage overpack assures resistance to freeze-thaw degradation even though this degradation force is expected to be minimal at the site. All other off-normal or postulated accident conditions are infrequent or one-time occurrences that do not contribute significantly to fatigue. In addition, the overpack utilizes materials that are not susceptible to brittle fracture during the lowest temperature permitted for loading.

A maintenance program for coatings on accessible areas of the overpack ensures that nicks or dings that expose the carbon steel components underneath are repaired before any significant corrosion can occur.

The relatively low neutron flux to which the storage overpack is subjected cannot produce measurable degradation of the cask's material properties and impair its intended safety function.

In summary, the materials of construction of the overpack design are compatible with the environment in which the overpack will operate. These design features and the coating maintenance program ensures that the overpack can perform its design functions for the life of the ISFSI.

4.7.4 NEUTRON ABSORBER

The effectiveness of the fixed borated neutron absorbing material used in the MPC fuel basket design requires that sufficient concentrations of boron be present to assure criticality safety during worst case design basis conditions over the 40-year design life of the MPC. Information on the characteristics of the Boral neutron absorbing material used in the MPC fuel basket is provided in Subsection 1.2.1.3.1 of the HI-STORM 100 System FSAR. The relatively low neutron flux, which continues to decay over time, to which this borated material is subjected, does not result in significant depletion of the material's available boron to perform its intended safety function. In addition, the boron content of the material used in the criticality safety analysis is conservatively based on the minimum specified boron areal density (rather than the nominal), which is further reduced by 25 percent for Boral or 10 percent for Metamic for analysis purposes, as described in Section 4.2.3.3.5 of the Diablo Canyon ISFSI FSAR. An evaluation discussed in Section 6.3.2 of the HI-STORM 100 System FSAR demonstrates that the boron depletion in the Boral or Metamic is negligible over a 50-year duration. Thus, sufficient levels of boron are present in the fuel basket neutron absorbing material to maintain criticality safety functions over the 40-year design life of the MPC.

4.7.5 EMBEDDED CASK ANCHORAGE SYSTEM AND PAD

The embedded cask anchorage system (i.e., embedment ring, coupling, rods and embedded plates and jam nuts) is constructed of carbon steel material in accordance with the requirements of Appendix B to ACI 349-01, as endorsed by Regulatory Guide 1.199 (Reference 9). Refer to Figure 4.2-7 (Holtec Drawing 4461, sheet 14) for the cask anchor stud and drawing number PGE-009-SK-301 and –302 for the embedded anchorage in the concrete pad (See Appendix “DOC 1” to Reference 8).

The steel components exposed to the environment (such as the top exposed surface of the embedment ring), will be properly coated per DCPP coating specifications, similar to the components in the power plant also located in the outdoor environment. The ISFSI reinforced concrete pad is located approximately 1/4 mile from the coastline at approximately 300 ft elevation and is not subjected to the harsh saltwater atmosphere that exists at other marine structures (such as the intake structure) located at DCPP. Existing DCPP structures with similar construction (i.e., uncoated reinforcement with minimum concrete cover per ACI Code) and environmental exposure conditions, as proposed for the ISFSI pad (e.g., the containment structure, auxiliary building), have been in service for over 20 years at DCPP and have shown no evidence of adverse degradation due to embedded steel corrosion. In order to provide necessary corrosion protection for the given environmental exposure, construction requirements specified in ACI 349-97 (Reference 5, Part 3, Chapters 4 and 5), as endorsed by Regulatory Guide 1.142 (Reference 10), will be followed. These requirements include meeting the concrete durability requirement for the maximum water to cement ratio and a minimum compressive strength and providing the minimum concrete cover for the reinforcing steel based on placement. To provide added protection from the potential of reinforcing steel corrosion, the concrete pad surface is maintained with a penetrating, breathable, water-repellent sealer to protect the concrete surfaces exposed to weather and marine air.

No corrosion allowance was applied to the embedded anchorage as necessary measures are taken to minimize / prevent the possibility of water intrusion into the pad. The pad is also periodically visually inspected to monitor the materiel condition of the facility and its components.

4.7.6 MATERIALS SUMMARY

Table 4.7-1 provides a listing of the materials of fabrication for the HI-STORM 100 dry storage system and summarizes the performance of the material in the expected operating environments during short-term loading/unloading operations and long-term storage operations. As a result of this review, no operations were identified that could produce adverse reactions beyond those conditions already generically evaluated and approved in the licensing of the HI-STORM 100 System.

4.7.7 REFERENCES

1. USNRC Bulletin 96-04, Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks.

2. USNRC Interim Staff Guidance Document 15, Materials Evaluation.

3. Holtec International Dry Storage Position Paper DS-248, Revision 2, Chemical Stability of the Holtec MPC Internals During Fuel Loading and Dry Storage.

4. ACI 349-01, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, 2001.

5. ACI 349-97, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, 1997.

6. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 1A, January 2003.

7. Deleted in Revision 2.

8. PG&E Calculation No. 52.27.100.705 (PGE-009-CALC-001), "Embedment Support Structure."

9. Regulatory Guide 1.199, Anchoring Components and Structural Supports in Concrete, USNRC, November 2003.

10. Regulatory Guide 1.142, Safety Related Concrete Structure for Nuclear Power Plants (Other than Reactor Vessels and Containment), USNRC, November 2001.

TABLE 4.2-1

PHYSICAL CHARACTERISTICS OF THE HI-STORM MPC

PARAMETER	NOMINAL VALUE	REFERENCE
Outside Diameter	68 $^{1}/_{2}$ inches	Figure 4.2-13
Length	181 $^{5}/_{16}$ inches	Figure 4.2-13
Maximum Heat Load (Intact Fuel)	27.8 kW for MPC-24 28.2 kW for MPC-24E and MPC-24EF 28.74 kW for MPC-32 with fuel burnup ≤ 45,000 MWD/MTU (backfill ≥ 29.3 psig and ≤ 33.3) 28.74 kW for MPC-32 with fuel burnup > 45,000 MWD/MTU (backfill ≥ 34 psig and ≤ 40) 24 kW for MPC-32 with fuel burnup >45,000 MWD/MTU (backfill ≥ 29.3 psig and ≤33.3)	Table 1.2.2 of the HI-STORM 100 System FSAR, Revision 1A Report No. HI-2053376, Revision 7 Report No. HI-2125191, Revision 6, Table 3.1-2 Report No. HI-2104625, Revision 10, Table 3.1-2
Maximum Heat Load (Damaged fuel or fuel debris)	26.8 kW for MPC-24E 26.8 kW for MPC-24EF	Table 2.1.7 of the HI-STORM 100 System FSAR, Revision 1A
Material of Construction	Stainless Steel (except neutron absorber and an aluminum seal	Figure 4.2-13

	washer or port plug with thread protector in both the vent and drain port assemblies). An alternative vent and drain port plug configuration may be used, which does not contain aluminum washers.	
Maximum Weight with Fuel	79,987 lb for MPC-24 82,389 lb for MPC-24E & MPC-24EF 90,000 lb for MPC-32	Table 2.0.1 of the HI-STORM 100 System FSAR, Revision 1A
Internal Atmosphere	Helium	Table 2.0.1 of the HI-STORM 100 System FSAR, Revision 1A

TABLE 4.2-2

PHYSICAL CHARACTERISTICS OF THE
HI-STORM 100SA OVERPACK

PARAMETER	VALUE	REFERENCE
Height	229 $^{1}/_{2}$ inches	Figure 4.2-7
Outside Diameter	146 $^{1}/_{2}$ inches (bottom baseplate)	Figure 4.2-7
Capacity	One loaded MPC-32	Table 2.0.2 of the HI-STORM 100 System FSAR, Revision 1A
Material of Construction	Concrete (lid and side shielding) Carbon steel (lid and shell structure)	Table 2.2.6 of the HI-STORM 100 System FSAR, Revision 7
Maximum Weight with a loaded MPC	360,000 lb	Table 2.0.2 of the HI-STORM 100 System FSAR, Revision 1A
Design Life	40 years	Table 2.0.2 of the HI-STORM 100 System FSAR, Revision 1A

TABLE 4.2-3

PHYSICAL CHARACTERISTICS OF THE
HI-TRAC 125D TRANSFER CASK

PARAMETER	VALUE	REFERENCE
Height	192 $^{1}/_{2}$ inches (including top lid)	Figure 4.2-8
Outside Flange Diameter	104 inches	Figure 4.2-8
Capacity	One loaded MPC-32	Table 2.0.3 of the HI-STORM 100 System FSAR, Revision 1A
Material of Construction	Steel (shells, baseplate, lids, water jacket, bolting hardware) Lead (between inner and outer shells) Holtite-A (inside top lid)	Table 2.2.6 of the HI- STORM 100 System FSAR, Revision 1A
Maximum Weight	233,500 lb, including loaded MPC-32 (water jacket filled), pool lid and top lid	Table 3.2.2 of the HI- STORM 100 System FSAR, Revision 1A
Design Life	40 years	Table 2.0.3 of the HI-STORM 100 System FSAR, Revision 1A

TABLE 4.2-4

SUMMARY OF MPC-32 MPC CAVITY PRESSURES[(a)(b)]
FOR NORMAL CONDITIONS

Condition	Pressure (psig)
Initial Backfill (at 70°F) MPCs loaded under Amendments 0, 1, and 2	33.3 (Maximum)
Initial Backfill (at 70°F) MPCs loaded under Amendment 3 and later	40.0 (Maximum)
Normal Condition with no rod rupture	78.1
Normal Condition with 1% rods ruptured (storage)	79.1
Normal Condition with no rod rupture (transport)	80.9

(a) Per NUREG-1536, pressure analyses with ruptured fuel rods (including BPRAs) is performed with release of 100% of the ruptured fuel rod fill gas and 30% of the significant radioactive gaseous fission products.

(b) Calculated normal condition pressures are taken from HI-2125191, Revision 6.

TABLE 4.2-5

DIABLO CANYON ISFSI COMPLIANCE WITH
GENERAL DESIGN CRITERIA (10 CFR 72, SUBPART F)

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	FSAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.122 (a) Quality standards	Structures, systems, and components (SSCs) important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function.	• Section 4.5 provides the classification for SSCs important to safety. • Chapter 4 describes the ISFSI design features of SSCs that are important to safety. • HI-STORM 100 SYSTEM FSAR Tables 2.2.6 and 8.1.6 provide the safety classifications of cask and ancillary components, respectively. • Chapter 11 describes the Diablo Canyon ISFSI QA Program
72.122 (b) Protection against environmental conditions and natural phenomena	SSCs important to safety must be designed to accommodate the effects of and be compatible with site characteristics and environmental conditions and to withstand postulated accidents.	• Section 3.2 provides the design bases and criteria for environmental conditions and natural phenomena for the Diablo Canyon ISFSI. • Section 4.2 describes the design for the ISFSI pads, cask anchor studs, and cask structure for normal, off-normal and accident conditions, and environmental conditions and natural phenomena. • Section 3.3 describes the design criteria for the cask transporter, ISFSI pad, and cask transfer facility (CTF). • Section 4.2 describes the design of the ISFSI concrete storage pad and CTF. • Section 4.3 describes the design for the cask transport system. • HI-STORM FSAR Chapters 3 and 11 describe the details of the cask design, including normal, off-normal, and accident conditions of storage.

TABLE 4.2-5

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	FSAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.122 (c) Protection against fires and explosions	SSCs important to safety must be designed and located so that they can continue to perform their safety functions under credible fire and explosion exposure conditions.	• Section 3.3.1.6 describes the fire and explosion protection design criteria. • Sections 4.2.3.3.2.9 and 4.2.3.3.2.10 discuss cask design features as they relate to the capability of the cask system to withstand explosions and fires. • Sections 8.2.5 and 8.2.6 describe the evaluations and analyses related to fires and explosions.
72.122 (d) Sharing of SSCs	SSCs important to safety must not be shared between the ISFSI and other facilities unless it is shown that such sharing will not impair the capability of either facility to perform its safety functions.	• Section 1.2 discusses the shared SSCs between the Diablo Canyon ISFSI and DCPP. No important to safety SSCs are shared between the ISFSI and DCPP.
72.122 (e) Proximity of sites	An ISFSI located near other nuclear facilities must be designed and operated to ensure that the cumulative effects of their combined operations will not constitute an unreasonable risk to the health and safety of the public.	• Sections 2.1 and 4.1 discuss the location and layout of the Diablo Canyon ISFSI. • Chapter 7 discusses the combined radiation doses to the public from the concurrent operation of the ISFSI and DCPP.
72.122 (f) Testing and maintenance of systems and components	Systems and components that are important to safety must be designed to permit inspection, maintenance, and testing.	• Section 3.3.1.5.1 describes the expected need for access to the ISFSI to conduct maintenance and inspection activities. • Section 4.2 describes the design features of the ISFSI that accommodate inspection, maintenance, and testing. • Chapter 9 of the HI-STORM FSAR describes the limited amount of maintenance expected to be required for the cask system.
72.122 (g) Emergency capability	SSCs important to safety must be designed for emergencies. The design must provide accessibility to the equipment by onsite and available offsite emergency facilities and services.	• Section 2.1.2 describes the accessibility of the Diablo Canyon site. • Section 9.5 summarizes the Emergency Plan for the ISFSI.

TABLE 4.2-5

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	FSAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.122 (h) Confinement barriers and systems	The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined. Ventilation systems must be provided, where necessary, to ensure confinement of airborne particulates. Periodic monitoring is sufficient, consistent with cask design requirements.	• Section 3.3.1.2.1 describes the HI-STORM 100 System confinement barriers and systems. • Sections 3.3.1.5.3, 3.3.1.7.2, 7.3.3, and 7.7 discuss the absence of radioactive effluents from the HI-STORM 100 System, which eliminates the need for ventilation systems. • Section 4.2.3.3.6 describes how the design of the HI-STORM 100 System maintains confinement integrity under all normal, off-normal, and accident conditions of storage. • Chapter 3 of the HI-STORM 100 SYSTEM FSAR describes the structural evaluations performed to demonstrate confinement integrity under all conditions of storage. • The Diablo Canyon ISFSI Technical Specifications include a surveillance requirement to periodically monitor the passive heat removal system for operability.
72.122 (i) Instrumentation and control systems	Instrumentation and control systems must be provided in accordance with cask design requirements to monitor normal, off-normal, and accident conditions.	• Section 3.3.1.3.2 discusses the fact that the HI-STORM 100 System requires no instrumentation for normal or off-normal operation or for accidents.
72.122 (j) Control room or control area	A control room or control area, if appropriate, must be designed to permit occupancy and actions to be taken to monitor the ISFSI safely under normal conditions, and to provide safe control of the ISFSI under off-normal or accident conditions.	• Section 3.3.1.5.1 discusses why access to the ISFSI may periodically be required. • Section 5.2 discusses why a dedicated ISFSI control room/area is not required. • The ISFSI Physical Security Plan provides the details for ISFSI access control. Section 9.6 provides a brief non-safeguards summary.

TABLE 4.2-5

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	FSAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.122 (k) Utility or other services	Each utility service system must be designed to meet emergency conditions. The design of utility services and distribution systems that are important to safety must include redundant systems to maintain the ability to perform safety functions assuming a single failure. An ISFSI located on the site of another facility may share common utilities and services provided the sharing or physical connection does not significantly increase the probability or consequences of an accident or malfunctions of equipment important to safety; or reduce the margin of safety as defined in the technical specification bases for either facility.	• Section 4.4.4 discusses utility supplies and systems. No important to safety services are shared. • Section 8.1.6 discusses the evaluation of a loss of electrical power to the Diablo Canyon ISFSI.
72.122 (l) Retreivability	Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal.	• Section 5.1 discusses unloading of the HI-STORM 100 System and ready fuel retrievability for return to the DCPP spent fuel pool.
72.124 (a) Design for criticality safety	Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical.	• Sections 3.3.1.4, 3.3.1.7, and 4.2.3.3.5 summarize the design features and administrative controls used to ensure subcriticality of the spent fuel is maintained during all phases of fuel loading, cask preparation, and storage. • Chapter 6 of the HI-STORM 100 System FSAR provides a detailed discussion of the criticality analyses for the system.

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	FSAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.124 (b) Methods of criticality control	When practicable, the design of an ISFSI must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. The continued efficacy of the neutron absorbing material may be confirmed by demonstration or analysis before use, showing significant degradation over the life of the facility cannot occur.	• Sections 3.3.1.4 and 4.2.3.3.5 discuss the combination of geometry and fixed neutron poisons as the means of subcriticality control. • Section 6.3.2 of the HI-STORM 100 System FSAR describes the Boral neutron absorber used in the Holtec MPCs and provides information showing that significant degradation over the life of the facility will not occur and verification of continued efficacy is not required.
72.124 (c) Criticality monitoring	A criticality monitoring system shall be maintained in each area where special nuclear material (SNM) is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Monitoring of dry storage areas where SNM is packaged in its stored configuration under a 10 CFR-72 license is not required.	• Section 4.2.3.3.5 discusses a criticality monitoring exemption issued by the NRC from the requirements of 10 CFR 50.68(b)(1) during MPC loading, unloading, and handling operations.
72.126 (a) Exposure control	Radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials.	• Section 3.3.1.5 provides the radiological protection design criteria and the key cask system components relied upon for shielding. • Chapter 7 discusses the radiation protection program.
72.126 (b) Radiological alarm systems	Radiological alarm systems must be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given set point and of concentrations of radioactive material in effluents above control limits.	• Section 3.3.1.5.3 discusses the requirements for radiological alarm systems. • Section 7.3.4 describes the radiological monitoring program.

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	FSAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.126 (c) Effluent and direct radiation monitoring	As appropriate for the handling and storage system, means to measure effluents must be provided for normal and accident conditions. Areas containing radioactive materials must be provided with systems for measuring the direct radiation levels in and around these areas.	• Section 3.3.1.5.3 describes how the HI-STORM 100 System emits no solid, gaseous, or liquid effluents under normal or off-normal conditions of storage. • Section 4.2.3.3.6 describes how confinement integrity is maintained under normal, off-normal, and accident conditions. • Section 7.3.4 describes the radiological monitoring program.
72.126 (d) Effluent control	The ISFSI must be designed to provide means to limit to ALARA levels, the release of radioactive materials in effluents during normal operations and control the release of radioactive materials in effluents under normal conditions and to control the release of radioactive materials under accident conditions.	• Section 3.3.1.5.3 describes how the HI-STORM 100 System emits no gaseous or liquid effluents under normal or off-normal conditions of storage. • Section 4.2.3.3.6 describes how confinement integrity is maintained under normal, off-normal, and accident conditions. • Section 7.5 provides discussion of the doses to the public from a hypothetical leak in the confinement boundary during normal, off-normal, and accident conditions. All doses are shown to be within regulatory limits.
72.128 (a) Spent fuel storage and handling systems	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.	• Section 4.2 describes the design of SSCs that contain or handle radioactive material for normal and accident conditions.
72.128 (b) Waste treatment	Radioactive waste treatment facilities must be provided.	• Sections 3.3.1.5.3, 3.3.1.7.2, 7.3.3, and 7.7 discuss how no radioactive waste is produced by the HI-STORM 100 System. • Sections 4.4.2 and 6.2 describes the decontamination process during loading operations and the treatment of and waste that is created.

TABLE 4.2-5

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	FSAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.130 Criteria for decommissioning	The ISFSI must be designed for decommissioning. Provisions must be made to facilitate decontamination of structures and equipment, minimize the quantity of radioactive wastes and contaminated equipment, and facilitate the removal of radioactive waste at the time of decommissioning.	• Section 4.6 summarizes the ISFSI preliminary decommissioning plan. • Section 2.4 of the HI-STORM 100 System FSAR describes the cask design features as they relate to decommissioning. • The Preliminary Decommissioning Plan (License Application, Attachment F) presents an overall description of the decommissioning requirements.

TABLE 4.3-1

IMPORTANT-TO-SAFETY COMPONENTS OF THE CASK TRANSPORTATION SYSTEM

Component	Function	Applicable Design Codes
Cask Transporter	Lift, handle, and transport a loaded HI-TRAC transfer cask or a HI-STORM 100SA overpack	Purchased commercial grade and tested prior to use in accordance with NUREG-0612
Lift Links	Transmit the force of the lifted load from the transfer cask lifting trunnions to the cask transporter lift points during vertical lifts. Transmit the force of the lifted load from the overpack lifting brackets to the cask transporter lift points during vertical lifts under off-normal or accident conditions with a loaded overpack in the CTF.	ANSI N14.6 per NUREG-0612, Section 5.1.6
HI-STORM Slings	Transmit the force of the loaded overpack from the lift links and HI-STORM lift brackets to remove the overpack from the CTF.	ASME B30.9 Purchased commercial grade and tested prior to use in accordance with NUREG-0612
MPC Downloader Slings	Transmit the force of the loaded MPC from the MPC lift cleats to the MPC downloader	ASME B30.9 Purchased commercial grade and tested prior to use in accordance with NUREG-0612
MPC Lift Cleats	Provide a lift point for raising and lowering the loaded MPC between the transfer cask and overpack	ANSI N14.6 per NUREG-0612, Section 5.1.6
HI-STORM Lifting Brackets	Transmit the force of the lifted load from the overpack lid studs to the cask transporter lift points during vertical lifts	ANSI N14.6 per NUREG-0612, Section 5.1.6
Connector Pins	Connect the transfer cask lift links or the overpack lifting brackets to the cask transporter lift links	ANSI N14.6 per NUREG-0612, Section 5.1.6

TABLE 4.5-1

QUALITY ASSURANCE CLASSIFICATION OF
MAJOR STRUCTURES, SYSTEMS, AND COMPONENTS

IMPORTANT TO SAFETY[(a)]	NOT IMPORTANT TO SAFETY
Classification Category A Multi-Purpose Canister Fuel Basket Transfer Cask MPC Lift Cleats MPC Downloader Slings[(b)] HI-STORM Lifting Brackets HI-STORM Mating Device Bolts and Shielding Frame Lateral Restraints[(b)] (HI-TRAC and transporter at CTF) Lift Links **Classification Category B** HI-STORM Overpack ISFSI Storage Pads Overpack Anchorage Hardware CTF Upper or Lower Fuel Spacers Transporter Connector Pins Helium Fill Gas[(b)] Cask Transporter[(b)] LPT[(c)] **Classification Category C** HI-STORM Cask Mating Device (except bolts and shielding frame) Damaged Fuel Container	Security Systems Fencing Lighting Electrical Power Communications Systems Automated Welding System (AWS) MPC Helium Backfill System MPC Forced Helium Dehydration System Rockfall Fence Rock-bolted Cutslope Supplemental Cooling System

(a) Major cask system components are listed according to the highest QA category of any subcomponent comprising the major component. The safety classification of the subcomponents and the determination of the ITS category of each item is administratively controlled by PG&E via design and procurement control procedures with input from the storage cask vendor.

(b) Purchased commercial grade and qualified by testing prior to use.

(c) Refer to 10 CFR 50 Q-List Section 1.

TABLE 4.7-1

HI-STORM 100 SYSTEM MATERIALS SUMMARY

Material/Component	Fuel Pool (Borated Water)[(a)]	ISFSI Pad and CTF (Open to Environment)
Alloy X: MPC fuel basket MPC baseplate MPC shell MPC lid MPC fuel spacers	Stainless steels have been extensively used in spent fuel storage pools with both borated and unborated water with no adverse reactions or interactions with spent fuel.	The MPC internal environment is an inert (helium) atmosphere and the external surface is exposed to ambient air.
Boral or Metamic Neutron absorber in MPC fuel basket	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 non-combustible concentration during MPC lid welding or cutting operations. See Chapter 5 for additional requirements for combustible gas monitoring and actions for control of combustible gas accumulation under the MPC lid.	The MPC internal environment is an inert (helium) atmosphere.
Steels (Transfer Cask): SA350-LF2 SA350-LF3 SA203-E SA515 Grade 70 SA516 Grade 70 SA193 Grade B7 SA 106	All exposed steel surfaces (except seal areas, and lifting trunnions) will be coated with material specifically selected for performance in the operating environments. Lid bolts are plated and the threaded portion of the bolt holes are plugged or otherwise covered to seal the threaded area from exposure to borated water.	Exposed surfaces of the HI-TRAC transfer cask are coated and maintained between uses.

TABLE 4.7-1

Material/Component	Fuel Pool (Borated Water)[a]	ISFSI Pad and CTF (Open to Environment)
Steels (Overpack): SA516 Grade 70 SA515 Grade 70 SA203-E SA350-LF2 SA240 304 A36 SA 193 B7 SA 194 2H	HI-STORM 100 storage overpack is not exposed to fuel pool environment.	Internal and external carbon steel surfaces are coated (except for threaded bolts and holes). Accessible external surfaces, including cask anchor studs and nuts, are maintained with an approved coating.
Stainless Steels (Misc.): SA240 304 MPC Fuel Spacer MPC Seal Bolt Lock Washers SA193 Grade B8 MPC Upper Fuel Spacer Bolt 18-8 S/S Transfer Cask Lid Washers	Stainless steels have been extensively used in spent fuel storage pools with borated and unborated water with no adverse reactions.	Stainless steel has a long proven history of corrosion resistance when exposed to the atmosphere. These materials are used for washers.
Nickel Alloy: SB637-NO7718 Transfer Cask Lifting Trunnions	No adverse reactions with borated water.	Short-term exposure to saline air environment.
Brass/Bronze: Transfer cask water jacket Pressure relief valve	The pressure relief valves are removed from the transfer cask prior to placing into SFP. The relief valves will be installed after the transfer cask is removed from the SFP. No significant adverse impact identified.	Short-term exposure to saline air environment. Normal maintenance assures operability of valves.

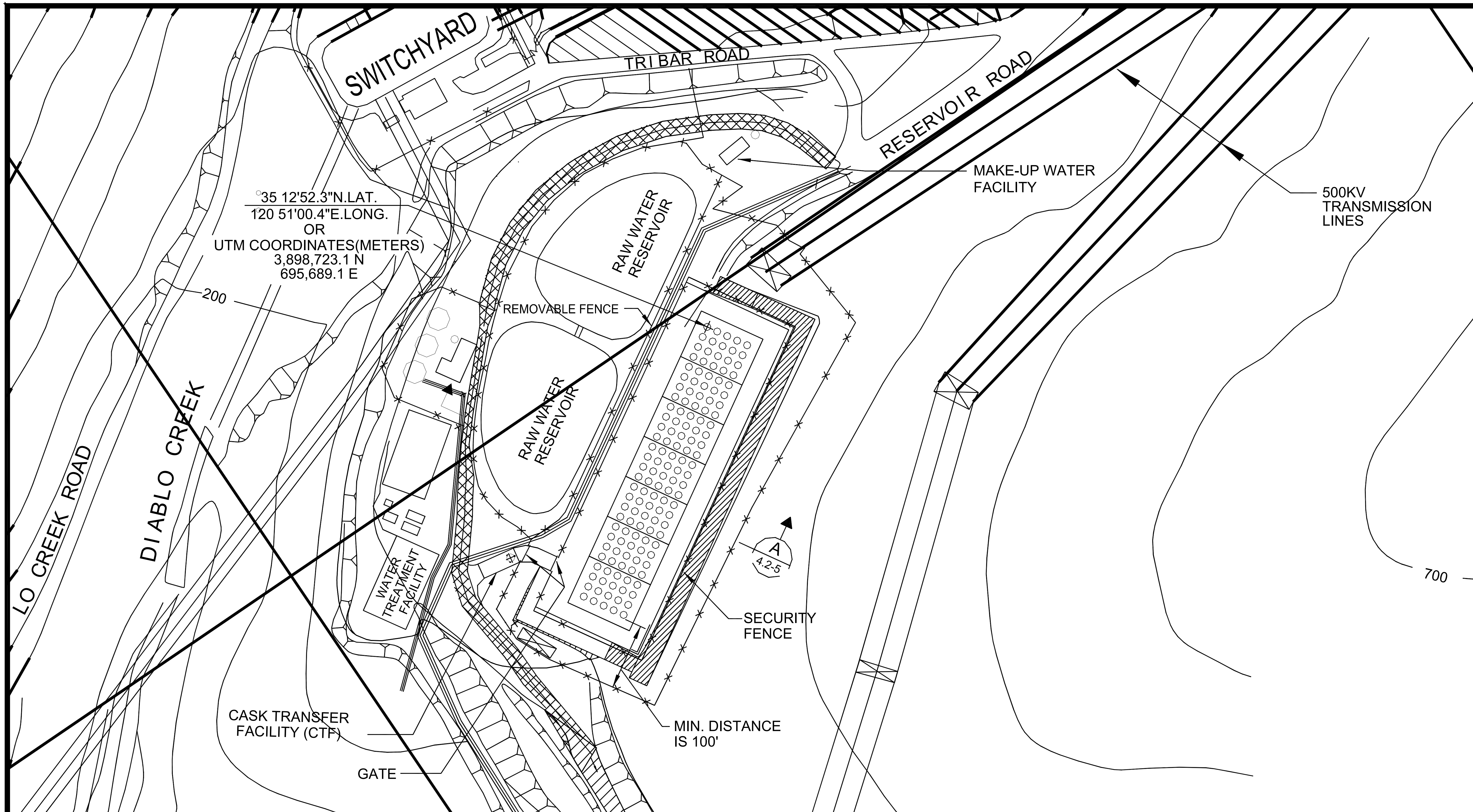
TABLE 4.7-1

Material/Component	Fuel Pool (Borated Water)[(a)]	ISFSI Pad and CTF (Open to Environment)
Holtite-A: Solid neutron shield in transfer cask lid structure	The neutron shield is fully enclosed in the top lid structural steel. The transfer cask top lid is not immersed in the spent fuel pool.	The neutron shield is fully enclosed in the top lid structural steel. Therefore, Holtite is not exposed to the environment.
Coatings: Bio-Gard 251 Carboline 890 Thermaline 450 Carbozinc 11/11HS Carboline 891 Bar-Rust 235 Exterior transfer cask and overpack carbon steel surface coatings DCPP Balance of Plant Coatings Program approved coatings	Carboline 890 or Bio-Gard 251 used for HI-TRAC transfer cask surfaces other than the inner shell for good decontamination properties and acceptable temperature resistance for the application. Acceptable performance for short-term exposure in borated pool water. Thermaline 450 selected for HI-TRAC transfer cask inner shell surfaces for excellent high temperature resistance properties. Will be exposed to borated water during in-pool operations as annulus is filled with clean borated water prior to placement in the spent fuel pool, and the inflatable seal prevents contaminated fuel pool water in-leakage. Manufacturer's data confirms that these coatings will perform adequately in these environments.	Coating products are used in a variety of corrosive external environments, including chemical industry. Good for resistance to oceanside saline environment. Thermaline 450 or Carbozinc 11/11HS used for HI-STORM overpack surfaces exposed to the environment for high temperature resistance and weathering capability. Includes exposed portions of cask anchorage (e.g., bottom flange). Manufacturer's data confirms that these coatings perform adequately in these environments. Carboline 891 or Bar-Rust 235 are applied to exposed portions of storage cask anchor studs, washers and nuts. The administratively controlled DCPP Balance of Plant Coatings Program may be used for field application of site-proven coating systems on ISFSI components.

TABLE 4.7-1

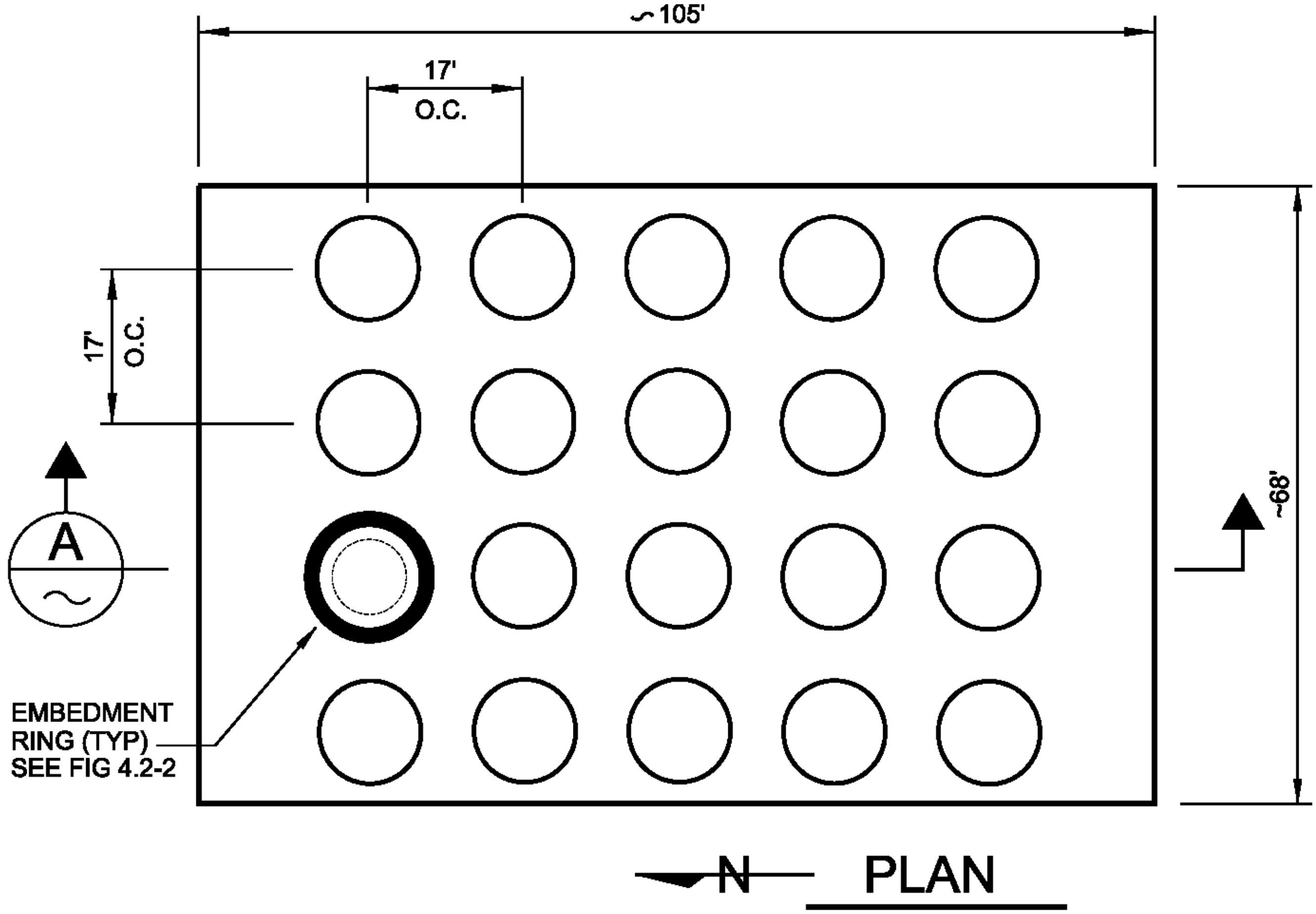
Material/Component	Fuel Pool (Borated Water)[a]	ISFSI Pad and CTF (Open to Environment)
Elastomer Seals: Transfer cask pool	Gasket is compressed between pool lid and transfer cask bottom flange to prevent spent fuel pool water inleakage. Gasket will be inspected periodically and replaced as necessary.	Leakage prevention function not required after the transfer cask is removed from the spent fuel pool and the annulus is drained.
Lead: Transfer cask body and lid gamma shield	Enclosed by carbon steel in transfer cask body and pool lid. Lead is not exposed to spent fuel pool water.	Enclosed by carbon steel in transfer cask body and lid. Lead is not exposed to ambient environment.
Concrete: Overpack body, lid, and pedestal shield	Storage overpack is not exposed to fuel pool water.	Concrete is enclosed by carbon steel in lid, pedestal, and overpack body in final storage configuration and not exposed to ambient environment.

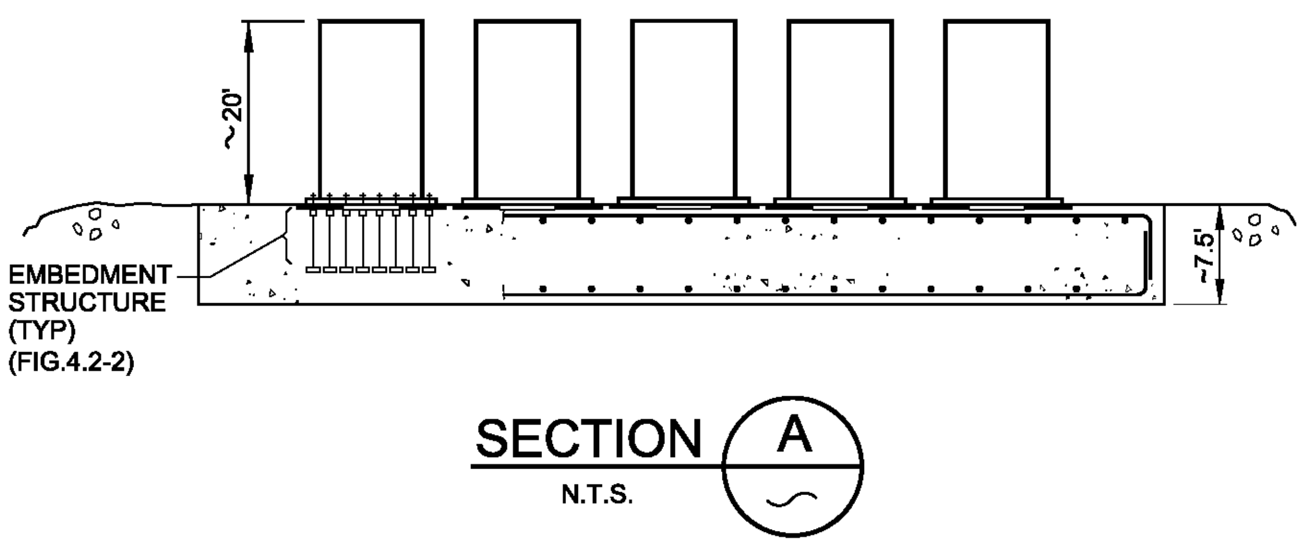
(a) HI-TRAC/MPC short-term operating environment during loading and unloading.



FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 4.1-1 **STORAGE LAYOUT**

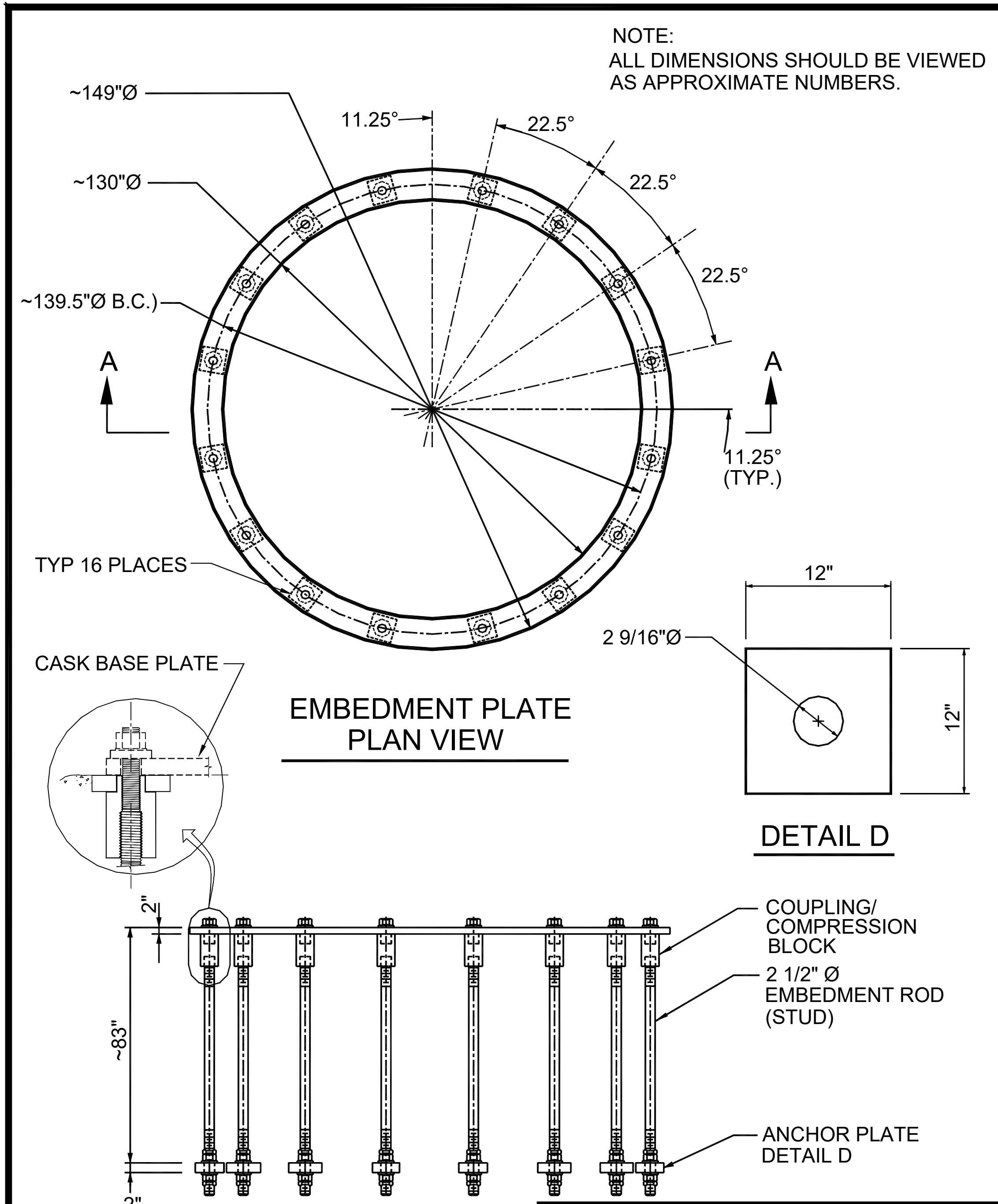
Revision 1 March 2006





FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 4.2-1 TYPICAL CASK STORAGE PAD (SCHEMATIC)

ONPG2ND

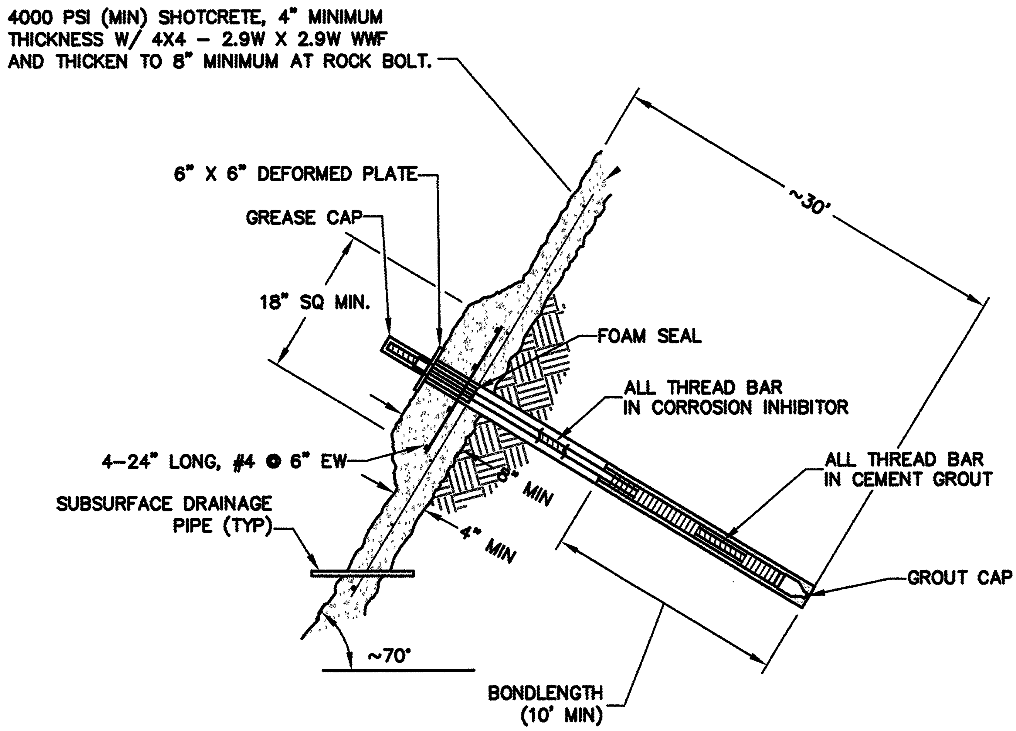


FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-2
EMBEDMENT ANCHORAGE STRUCTURE

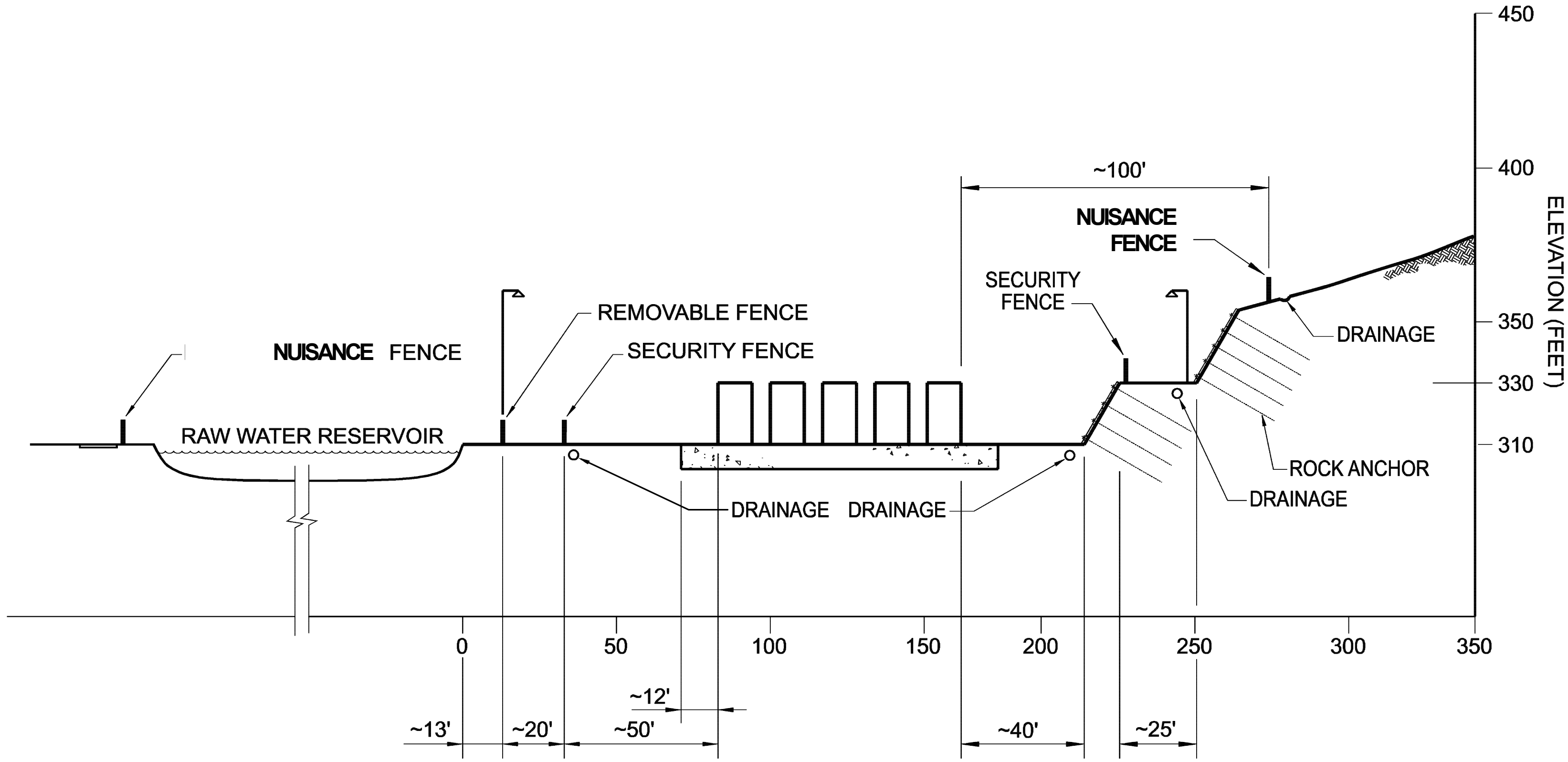
0NPG2ND



SHOTCRETE LAYER AND ROCK ANCHOR DETAIL
NTS

NOTE: ROCK ANCHOR TO BE CONTCH SYSTEMS CLASS 1 PROTECTED BAR ANCHOR, 1" DIA. GRADE 120 H (OR OTHER APPROVED EQUIVALENT). LOCK-OFF AT 50 KIPS.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 4.2-3 TYPICAL SHOTCRETE AND ROCK ANCHOR DETAIL



CROSS SECTION THROUGH ISFSI
LOOKING EAST

SECTION A FIG.4.1-1

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 4.2-5 ISFSI SITE X-SECTION VIEW

Revision 6 March 2016

LICENSING DRAWING PACKAGE COVER SHEET

CLIENT PG&E

PROJECT NO. 1073	P.O. NO. 3100813919
DRAWING PACKAGE I.D. 4461	TOTAL SHEETS 14

LICENSE DRAWING PACKAGE CONTENTS:

SHEET	DESCRIPTION
1	COVER SHEET
2	ASSEMBLY
3	ASSEMBLY
4	INNER SHELL & BASE INLET WELDMENT
5	INNER SHELL & BASE INLET DETAILS
6	OVERPACK BODY ASSEMBLY
7	OVERPACK BODY DETAILS
8	CLOSURE LID ASSEMBLY
9	CLOSURE LID WELDMENT
10	CLOSURE LID DETAILS
11	PEDESTAL ASSEMBLY WELDMENT
12	PEDESTAL ASSEMBLY DETAILS
13	GAMMA SHIELDING ASSEMBLY WELDMENT
14	ANCHORING DETAILS & ASSEMBLY

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REV	AFFECTED DRAWING SHEET NUMBERS	SUMMARY OF CHANGES/ AFFECTED ECOs	PREPARED BY	APPROVAL DATE	VIR# †
15	ALL SHEETS	ECO #1073-121 REV 1	RAS	SEE BELOW	SEE BELOW

† THE VALIDATION IDENTIFICATION RECORD (VIR) NUMBER IS A COMPUTER GENERATED RANDOM NUMBER WHICH CONFIRMS THAT ALL APPROPRIATE REVIEWS OF THIS DRAWING ARE DOCUMENTED IN COMPANY'S NETWORK.

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 1 OF 14

Revision 8 March 2020

NOTES:

1. THE EQUIPMENT DOCUMENTED IN THIS DRAWING PACKAGE HAS BEEN CONFIRMED BY HOLTEC INTERNATIONAL TO COMPLY WITH THE SAFETY ANALYSES DESCRIBED IN THE DIABLO CANYON ISFSI FSAR.
2. DIMENSIONAL TOLERANCES ON THIS DRAWING ARE PROVIDED SOLELY FOR LICENSING PURPOSES TO DEFINE REASONABLE LIMITS ON THE NOMINAL DIMENSIONS USED IN LICENSING WORK. HARDWARE IS FABRICATED IN ACCORDANCE WITH THE FABRICATION DRAWINGS, WHICH HAVE MORE RESTRICTIVE TOLERANCES, TO ENSURE COMPONENT FIT-UP. DO NOT USE WORST-CASE TOLERANCE STACK-UP FROM THIS DRAWING TO DETERMINE COMPONENT FIT-UP.
3. THE REVISION LEVEL OF EACH INDIVIDUAL SHEET IN THE PACKAGE IS THE SAME AS THE REVISION LEVEL OF THIS COVER SHEET. A REVISION TO ANY SHEET(S) IN THIS PACKAGE REQUIRES UPDATING OF REVISION NUMBERS OF ALL SHEETS TO THE NEXT REVISION NUMBER.
4. APPLICABLE CODES AND STANDARDS ARE DELINEATED IN SECTIONS 3.3.1 & 3.4 OF THE FSAR.
5. ALL WELDS REQUIRE VISUAL EXAMINATION. ADDITIONAL NDE INSPECTIONS ARE NOTED ON THE DRAWING IF REQUIRED. NDE TECHNIQUES AND ACCEPTANCE CRITERIA ARE PROVIDED IN THE HOLTEC FSAR.
6. UNLESS OTHEWISE NOTED, FULL PENETRATION WELDS MAY BE MADE FROM EITHER SIDE OF A COMPONENT.
7. THIS COMPONENT IS IMPORTANT-TO-SAFETY, CATEGORY B, BASED ON THE HIGHEST CLASSIFICATION OF ANY SUBCOMPONENT. SUBCOMPONENT CLASSIFICATIONS ARE PROVIDED ON THE FABRICATION DRAWING.
8. ALL WELD SIZES ARE MINIMUMS EXCEPT AS ALLOWED BY APPLICABLE CODES AS CLARIFIED IN THE FSAR. FABRICATORS MAY ADD WELDS WITH HOLTEC APPROVAL.
9. WELDS IDENTIFIED WITH AN # ARE CONSIDERED NON-NF WELDS. WELDS MAY BE MADE USING PREQUALIFIED WELDS IN ACCORDANCE WITH AWS D1.1 OR PER ASME SECTION IX.
10. CHANGES TO THIS DRAWING AFFECT FABRICATION DRAWING 4425.
11. 5" REF. X 5" REF. SQUARE PLATE OR ∅5 3/4" REF. CIRCULAR PLATE CAN BE USED.
    IN ADDITION, ROUND WASHERS MAY BE TRIMMED TO PROVIDE CLEARANCE FOR GUSSET WELDS. MINIMUM WIDTH OF TRIMMED ROUND WASHER IS 5".
12. SHIMS MAY BE INSTALLED BETWEEN HI-STORM BASEPLATE & ISFSI PAD EMBEDMENT RING TO CONTROL GAPS DUE TO OUT OF FLAT CONDITIONS.
13. ONE OF THE VENT DIMENSIONS MAY BE BE LESS THAN THE SPECIFIED MINIMUM IF IT IS DEMONSTRATED THAT THE FLOW AREA OF THE VENT IS GREATER THAN WHAT WOULD EXIST IF BOTH DIMENSIONS WERE AT THEIR MINIMUMS (i.e., 117.9 SQUARE INCHES).
14. DELETED.

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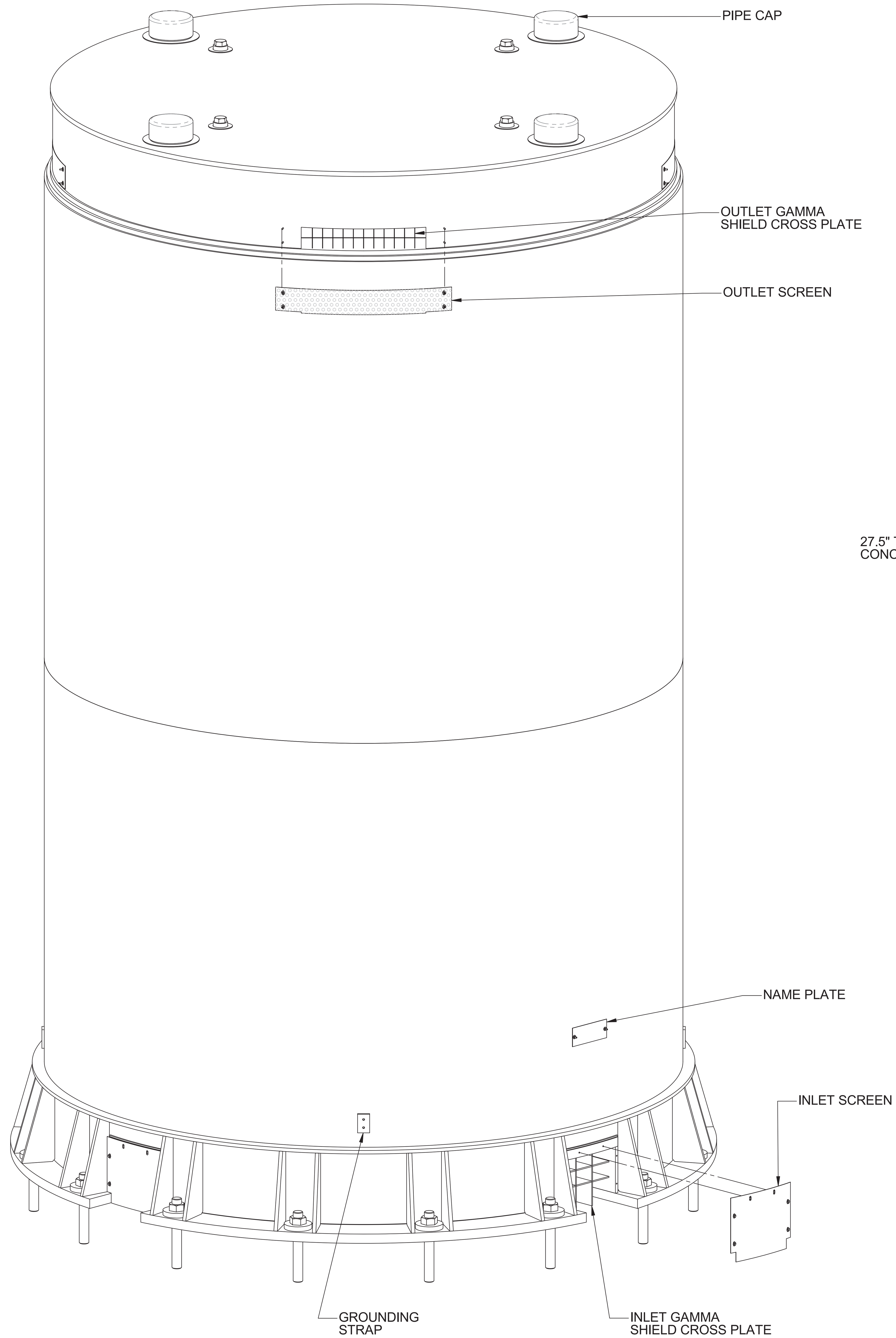
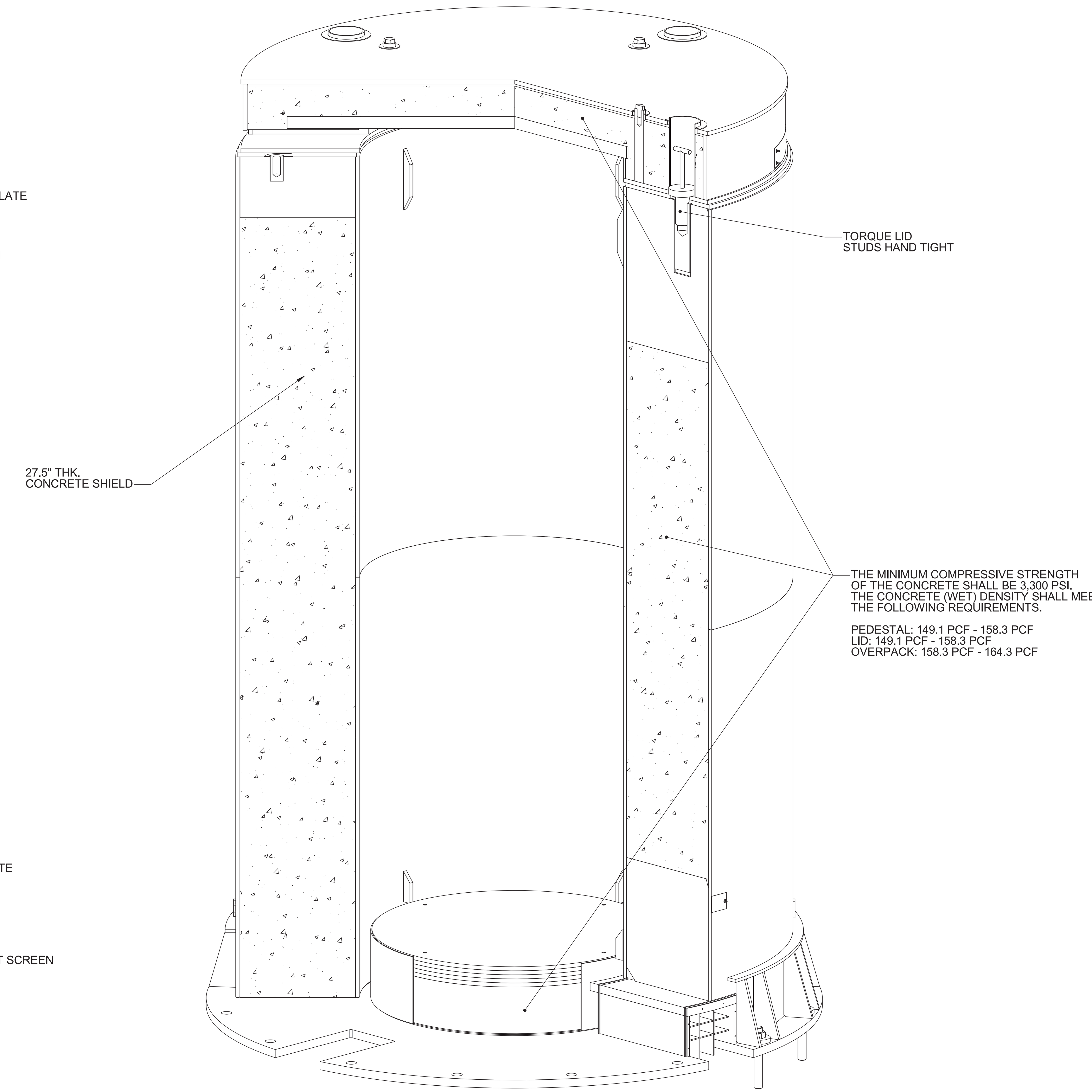
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)

15

CLIENT	PG&E
DESCRIPTION	HI-STORM 100SA ASSEMBLY
PROJECT NO.	1073
P.O. NO.	3100813919
DRAWING NO.	4461
SHEET	1
TOTAL SHEETS	14
FILE PATH:	G:\DRAWINGS\1073\4461





FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 2 OF 14

Revision 8 March 2020

HOLTEC INTERNATIONAL
NUCLEAR POWER DIVISION
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)
15

CLIENT: PG&E

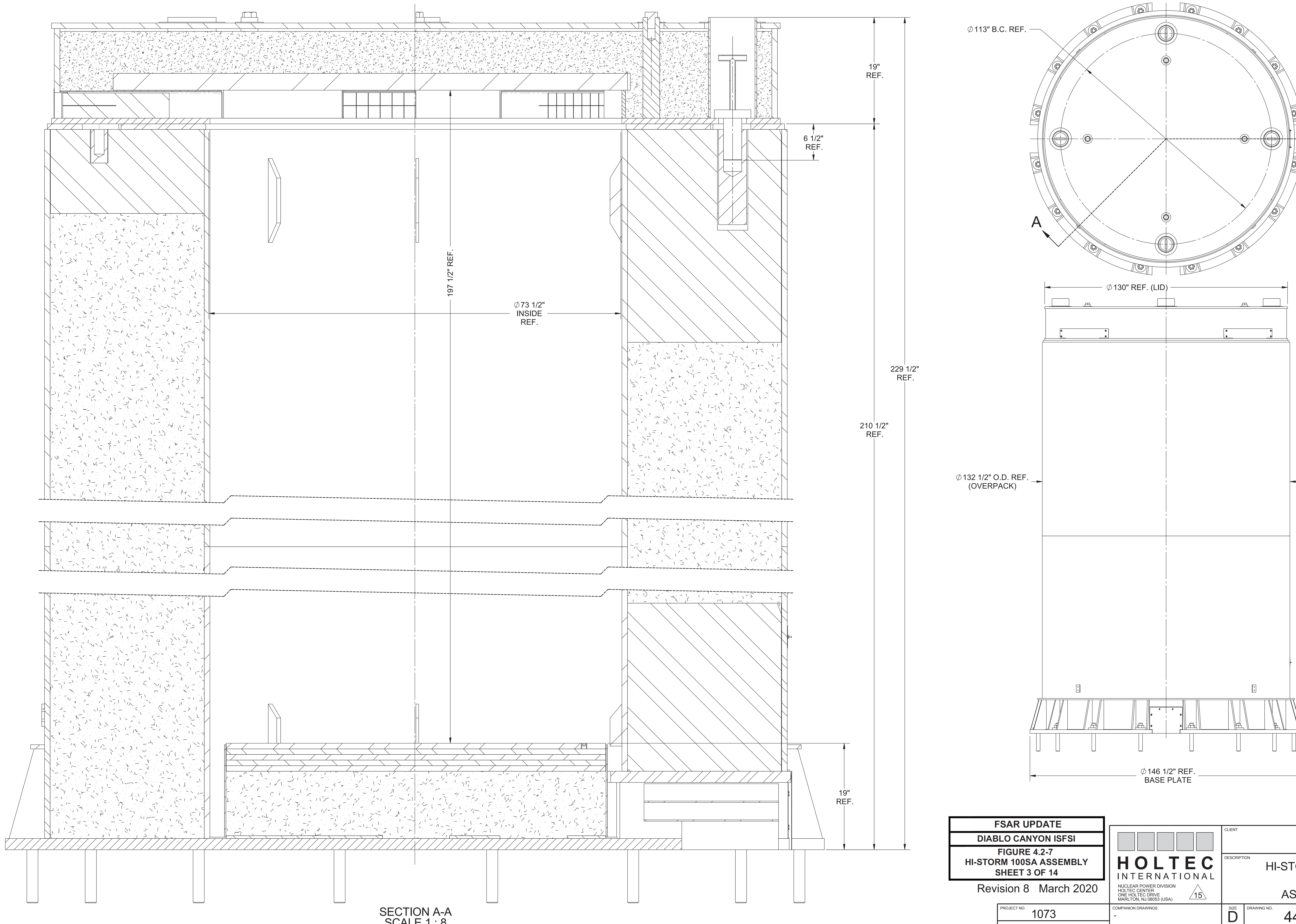
DESCRIPTION: HI-STORM 100SA ASSEMBLY

PROJECT NO. 1073
3100813919

COMPANION DRAWINGS: -

SIZE: D | DRAWING NO. 4461 | SHEET 2 | REV 15

SCALE: NONE | FILE PATH: G:\DRAWINGS\1073\4461



FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 3 OF 14

Revision 8 March 2020

PROJECT NO.	1073
	3100813919

HOLTEC INTERNATIONAL
NUCLEAR POWER DIVISION
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)
15

COMPANION DRAWINGS: -

CLIENT	PG&E
DESCRIPTION	HI-STORM 100SA ASSEMBLY

SIZE	DRAWING NO.	SHEET	REV
D	4461	3	15
SCALE NONE	FILE PATH: G:\DRAWINGS\1073\4461		

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 4 OF 14

Revision 8 March 2020

PROJECT NO.	1073
	3100813919

HOLTEC INTERNATIONAL
NUCLEAR POWER DIVISION
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)
15

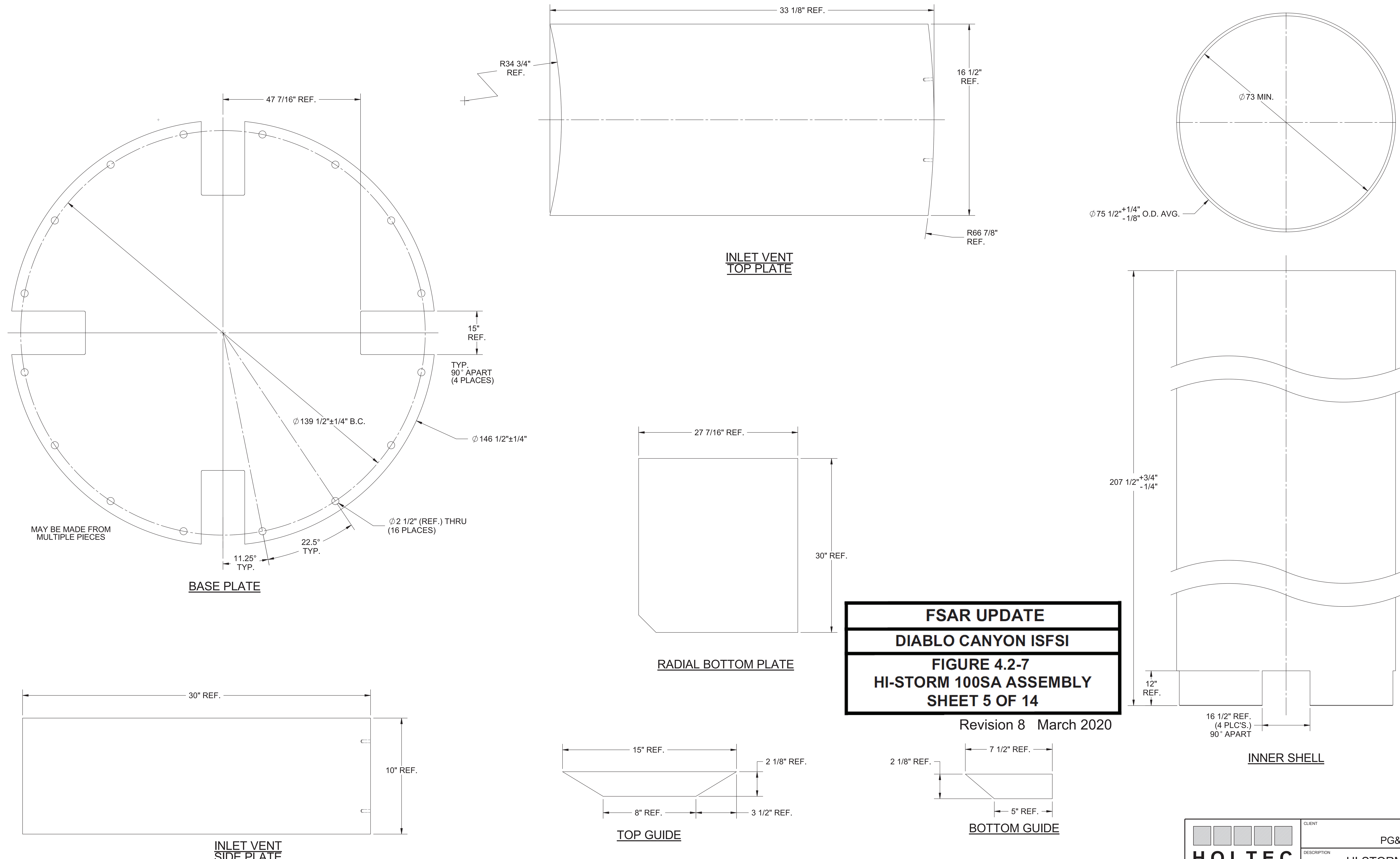
COMPANION DRAWINGS: -

CLIENT: PG&E

DESCRIPTION: HI-STORM 100SA INNER SHELL & BASE INLET WELDMENT

SIZE	DRAWING NO.	SHEET	REV.
D	4461	4	15

SCALE: NONE
FILE PATH: G:\DRAWINGS\1073\4461



FSAR UPDATE

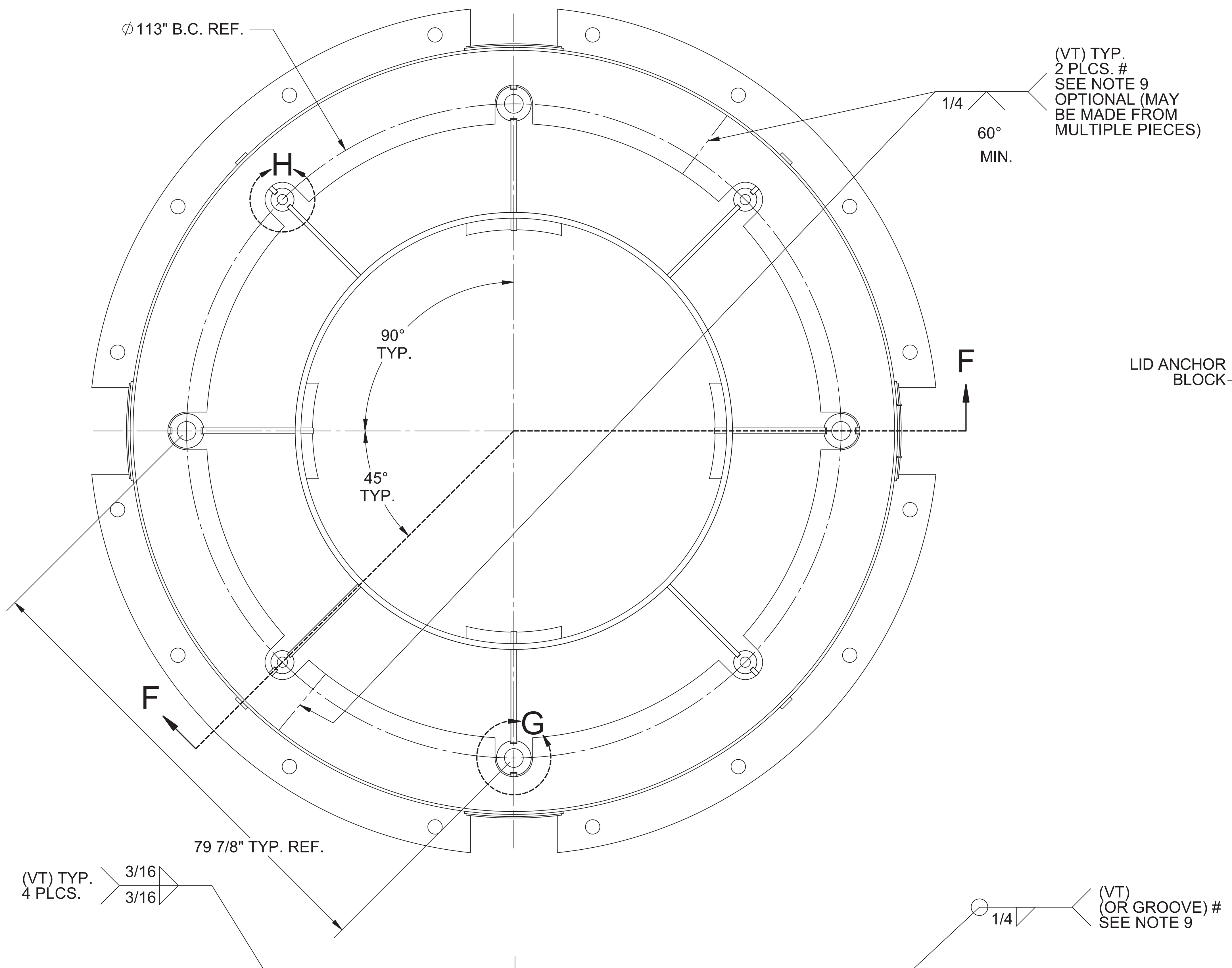
DIABLO CANYON ISFSI

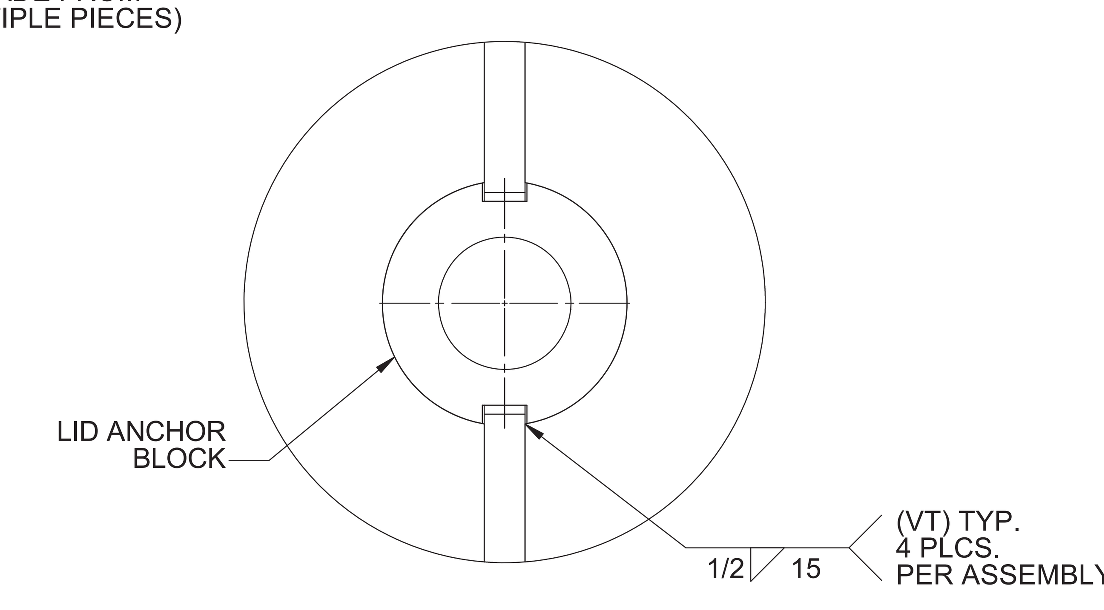
FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 5 OF 14

Revision 8 March 2020

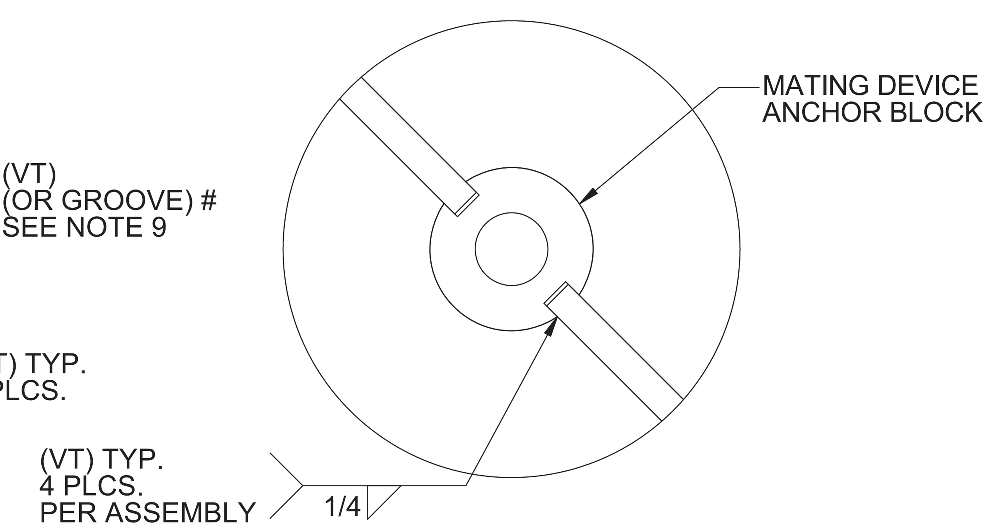
HOLTEC INTERNATIONAL
NUCLEAR POWER DIVISION
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)

CLIENT	PG&E
DESCRIPTION	HI-STORM 100SA INNER SHELL & BASE INLET DETAILS
PROJECT NO.	1073
	3100813919
COMPANION DRAWINGS:	-
SIZE	D
DRAWING NO.	4461
SHEET	5
REV.	15
SCALE	NONE
FILE PATH:	G:\DRAWINGS\1073\4461

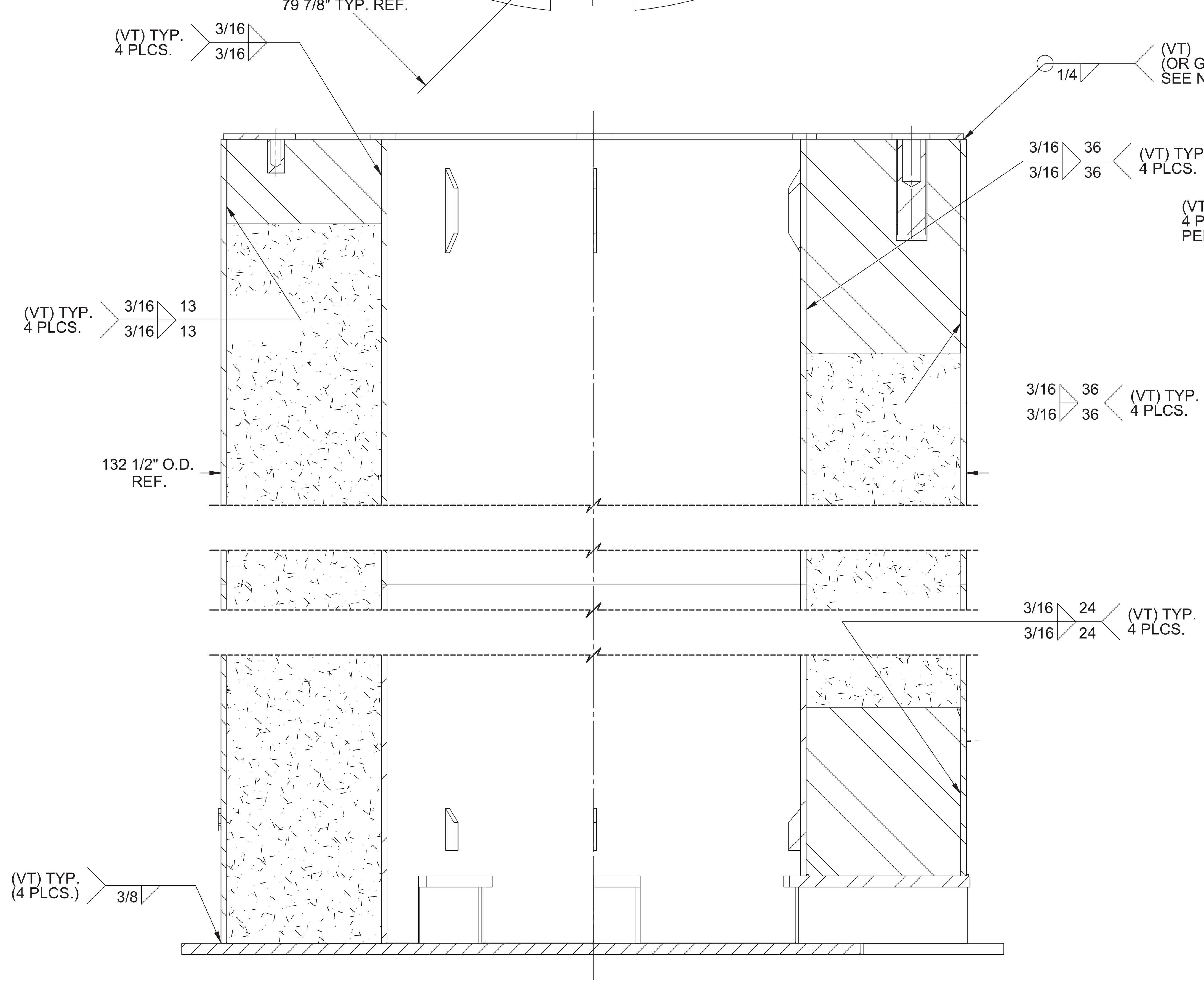




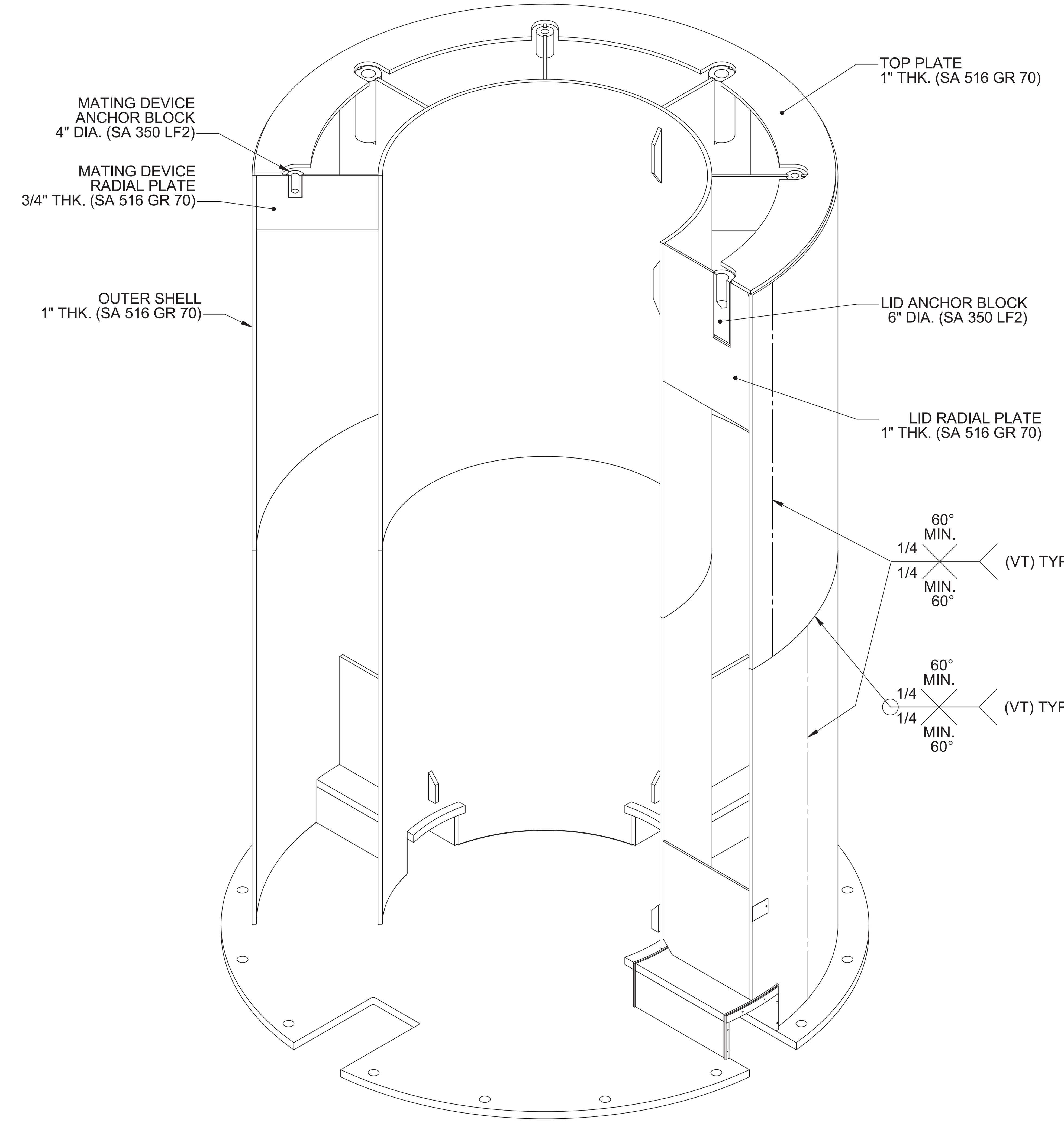
DETAIL G



DETAIL H



SECTION F-F



ISOMETRIC SECTION VIEW

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 6 OF 14

Revision 8 March 2020

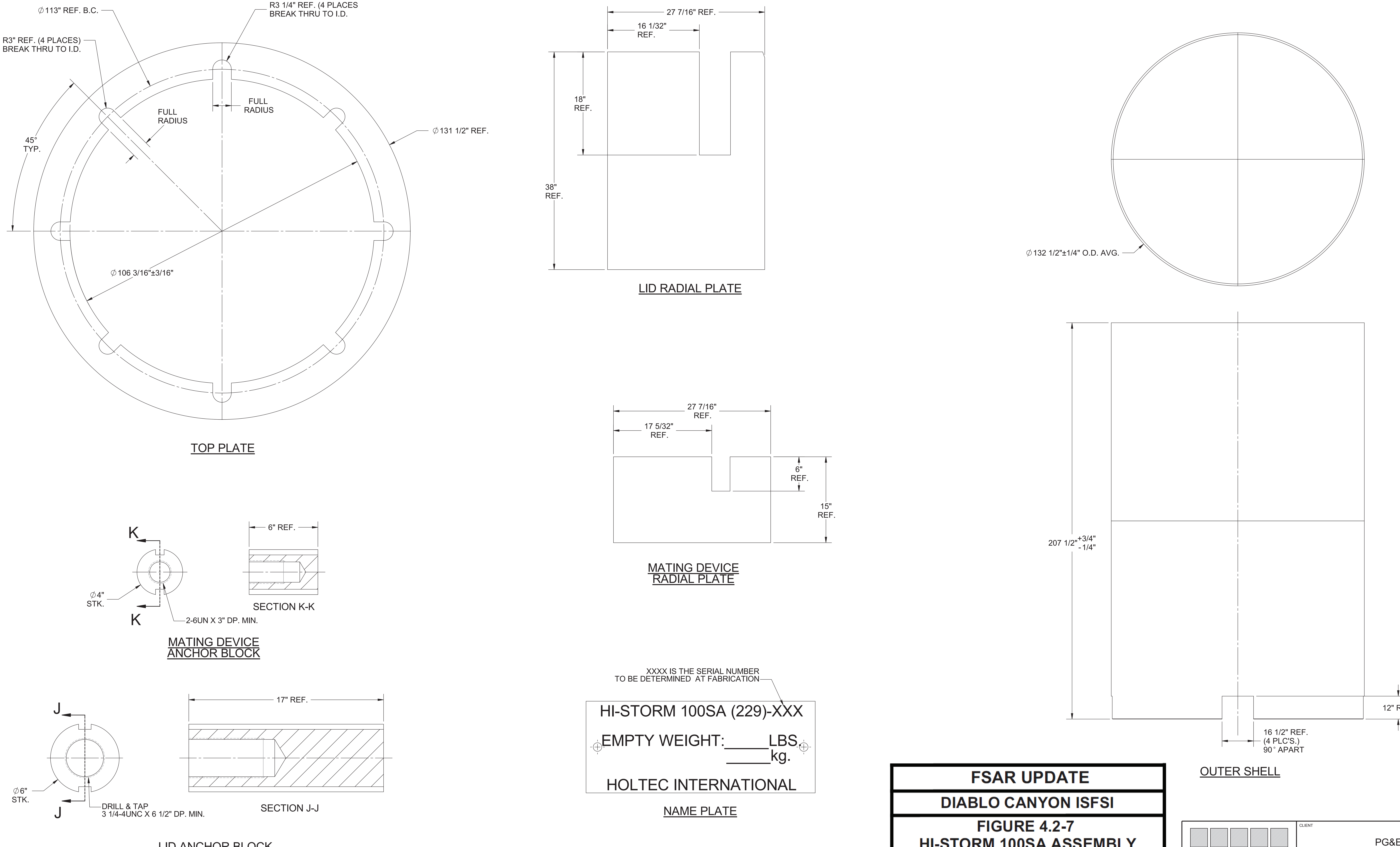
PROJECT NO.	1073
	3100813919

HOLTEC INTERNATIONAL
NUCLEAR POWER DIVISION
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)
15

COMPANION DRAWINGS: -

CLIENT	PG&E
DESCRIPTION	HI-STORM 100SA OVERPACK BODY ASSEMBLY

SIZE	DRAWING NO.	SHEET	REV
D	4461	6	15
SCALE: NONE	FILE PATH: G:\DRAWINGS\1073\4461		



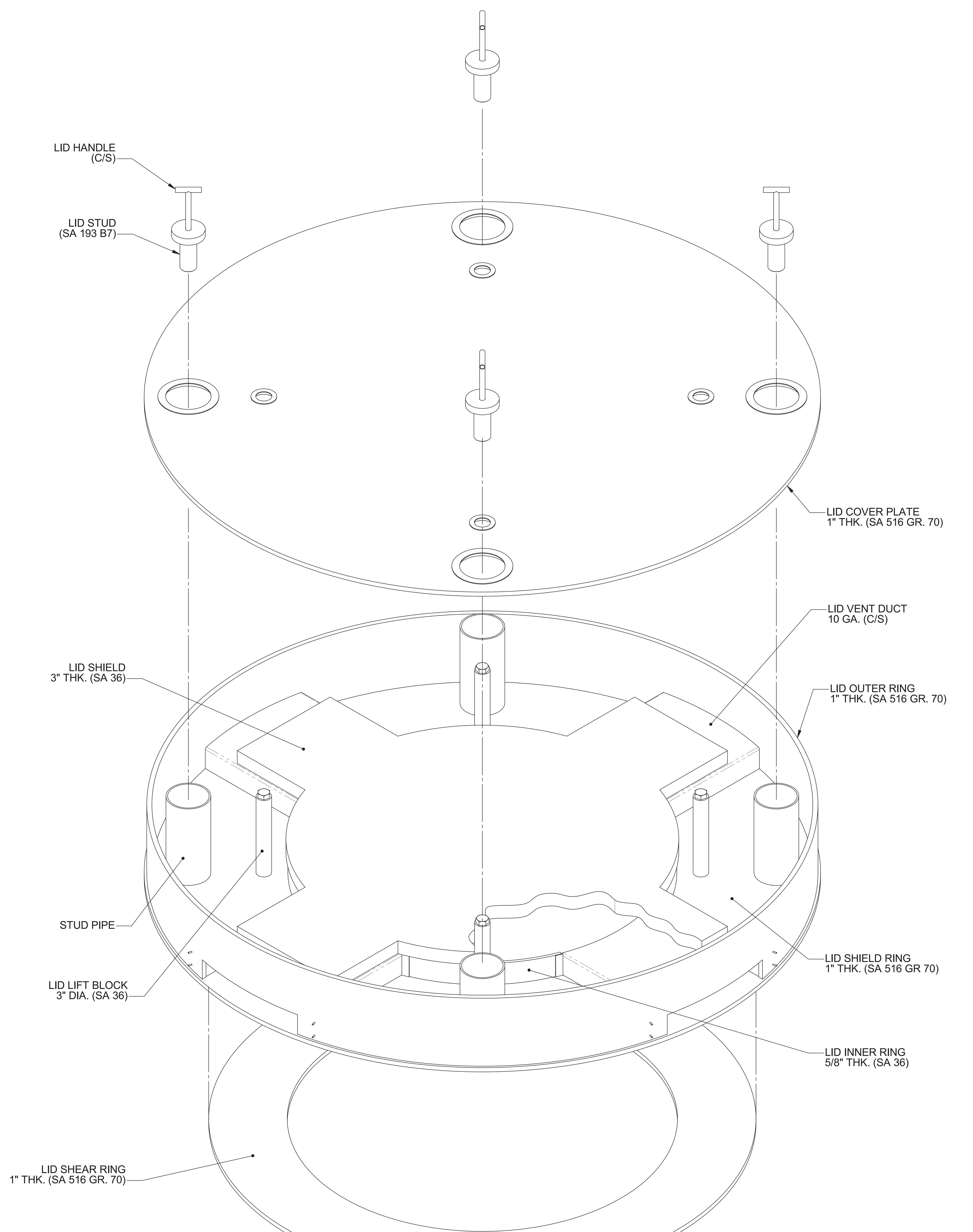
FSAR UPDATE

DIABLO CANYON ISFSI

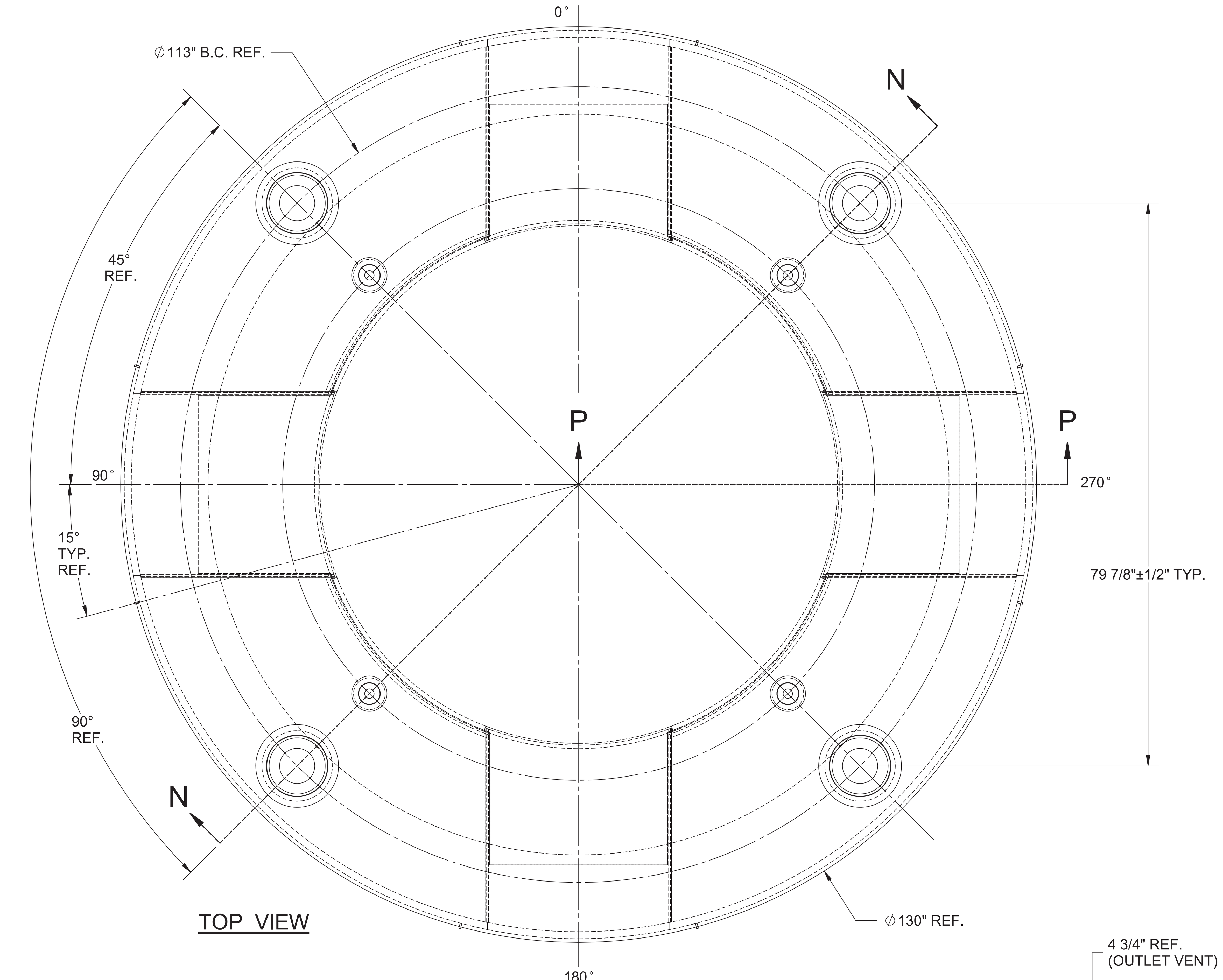
FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 7 OF 14

Revision 8 March 2020

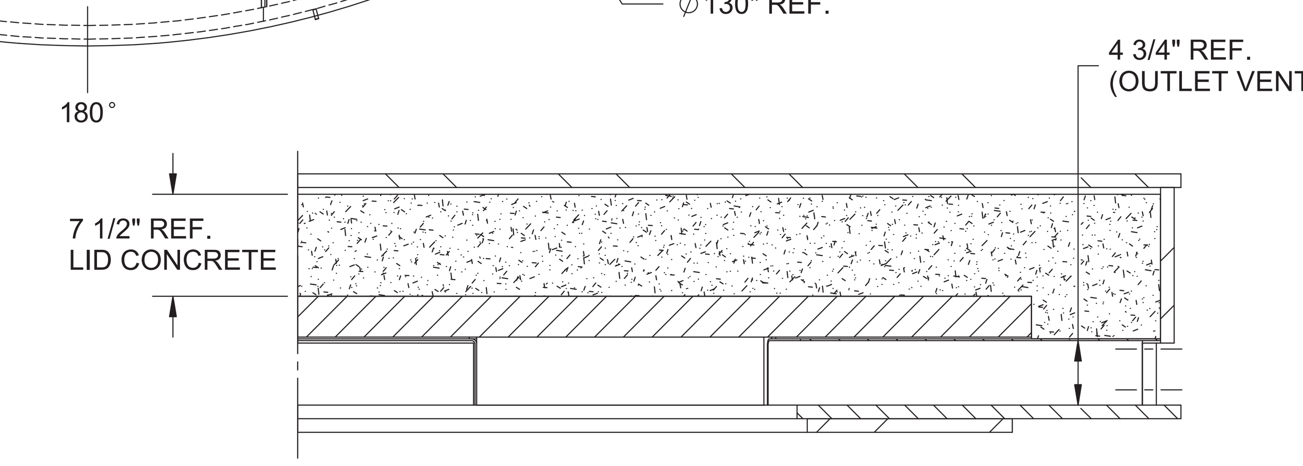
HOLTEC INTERNATIONAL	CLIENT: PG&E
NUCLEAR POWER DIVISION, HOLTEC CENTER, ONE HOLTEC DRIVE, MARLTON, NJ 08053 (USA) 15	DESCRIPTION: HI-STORM 100SA OVERPACK BODY DETAILS
PROJECT NO. 1073 / 3100813919	COMPANION DRAWINGS: -
SIZE: D	DRAWING NO. 4461
SHEET: 7	REV: 15
SCALE: NONE	FILE PATH: G:\DRAWINGS\1073\4461



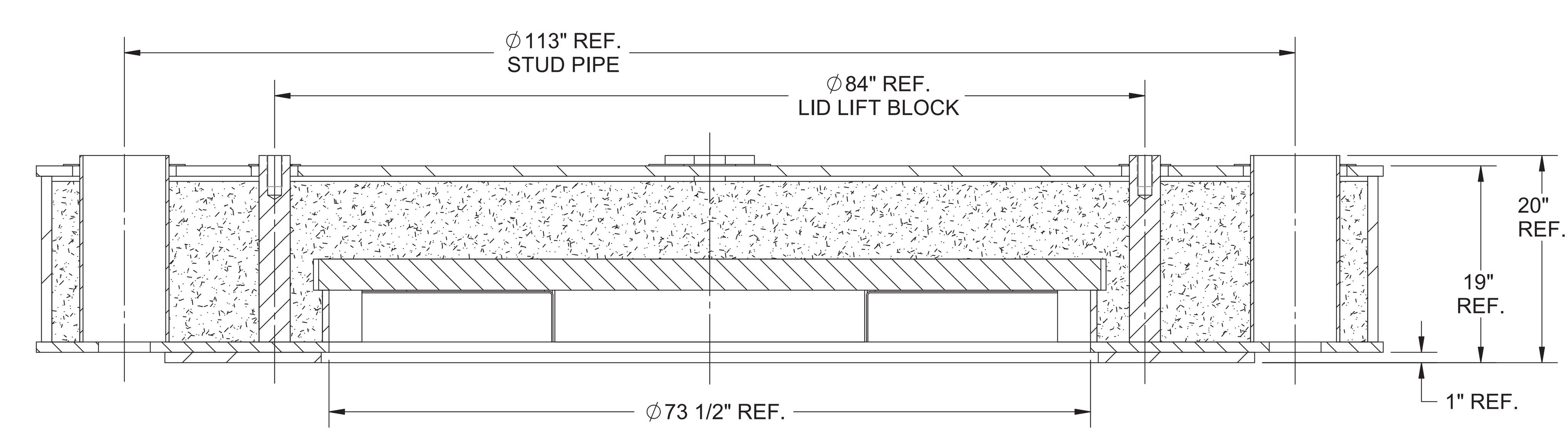
EXPLODED VIEW



TOP VIEW



SECTION P-P



SECTION N-N

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 8 OF 14

Revision 8 March 2020

PROJECT NO.	1073
	3100813919

HOLTEC INTERNATIONAL
NUCLEAR POWER DIVISION
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)
15

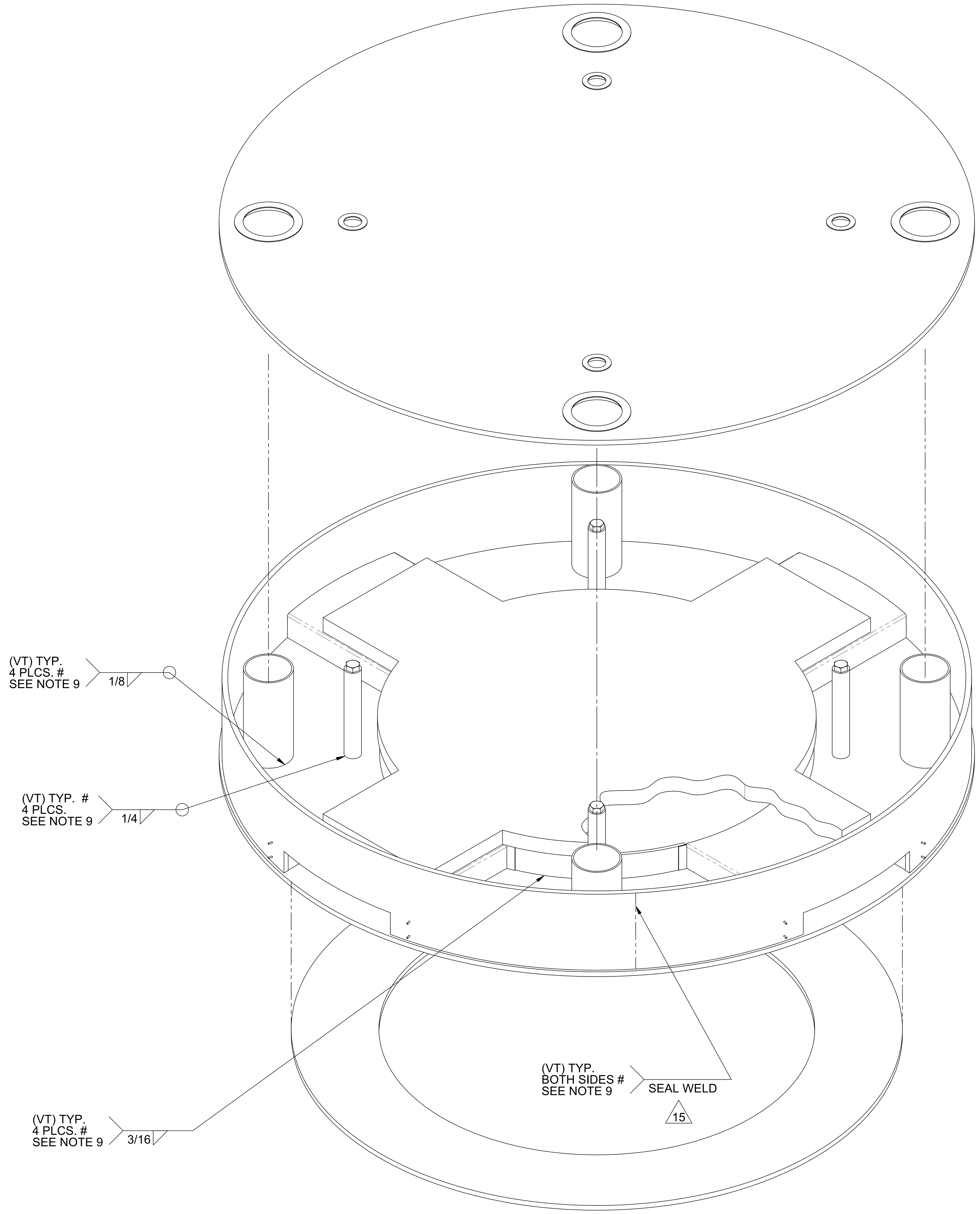
COMPANION DRAWINGS: -

CLIENT: PG&E

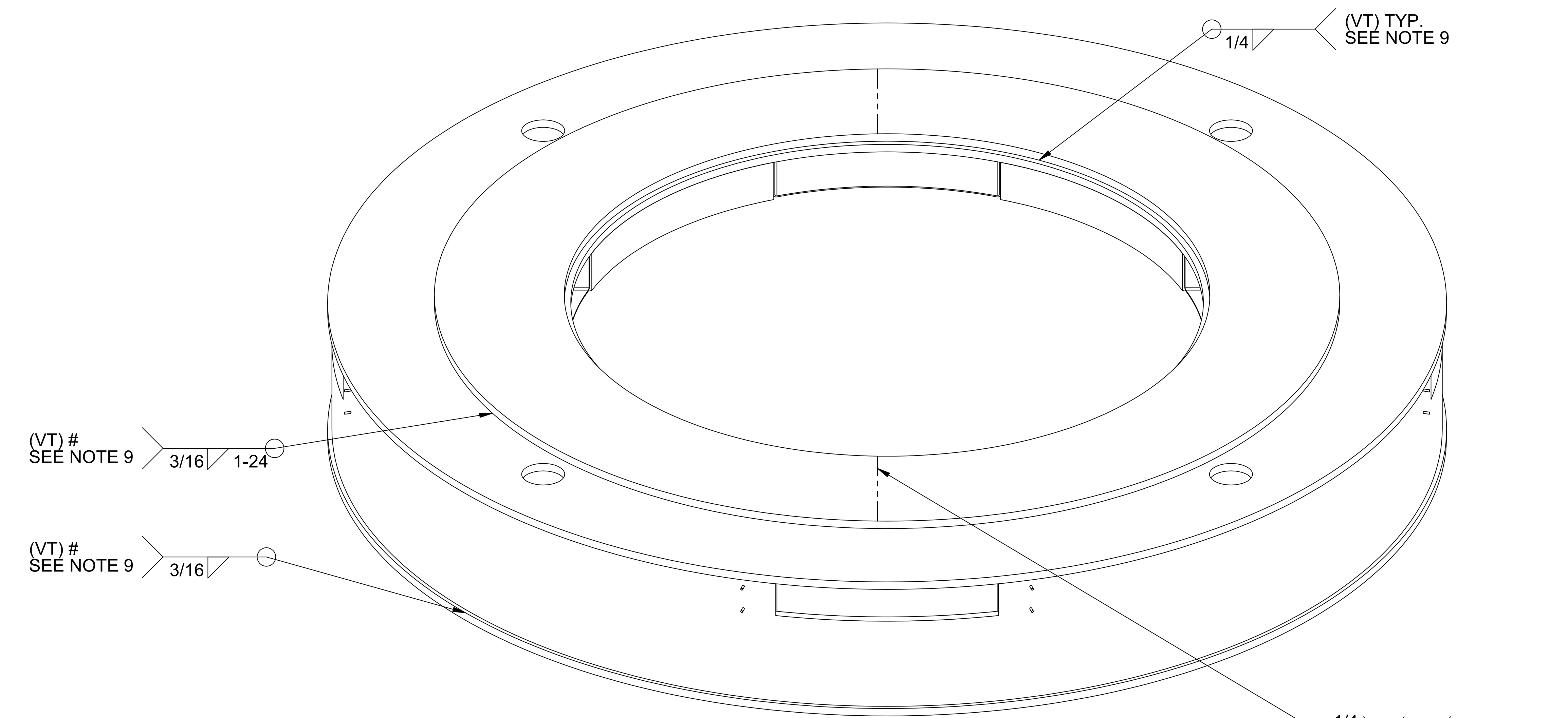
DESCRIPTION: HI-STORM 100SA CLOSURE LID ASSEMBLY

SIZE	DRAWING NO.	SHEET	REV
D	4461	8	15

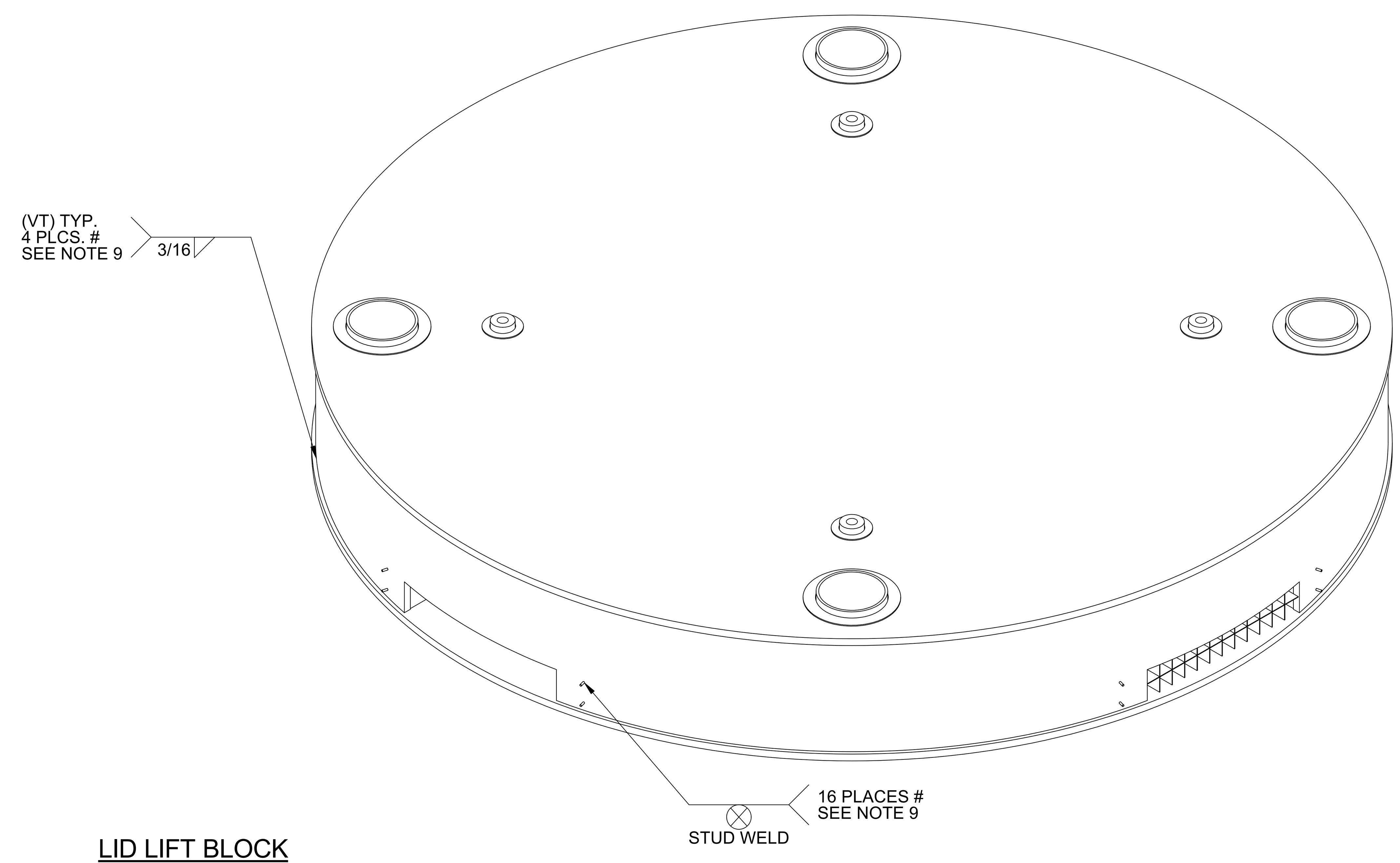
SCALE: NONE
FILE PATH: G:\DRAWINGS\1073\4461



EXPLODED VIEW



ISOMETRIC BOTTOM VIEW



LID LIFT BLOCK

ISOMETRIC ASSEMBLY VIEW

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 4.2-7 HI-STORM 100SA ASSEMBLY SHEET 9 OF 14

Revision 9 December 2021

HOLTEC INTERNATIONAL
NUCLEAR POWER DIVISION
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)
15

CLIENT: PG&E

DESCRIPTION: HI-STORM 100SA CLOSURE LID DETAILS

PROJECT NO. 1073
3100813919

COMPANION DRAWINGS: -

SIZE: D
DRAWING NO. 4461
SHEET 9
REV. 15
SCALE: NONE
FILE PATH: G:\DRAWINGS\1024\4461

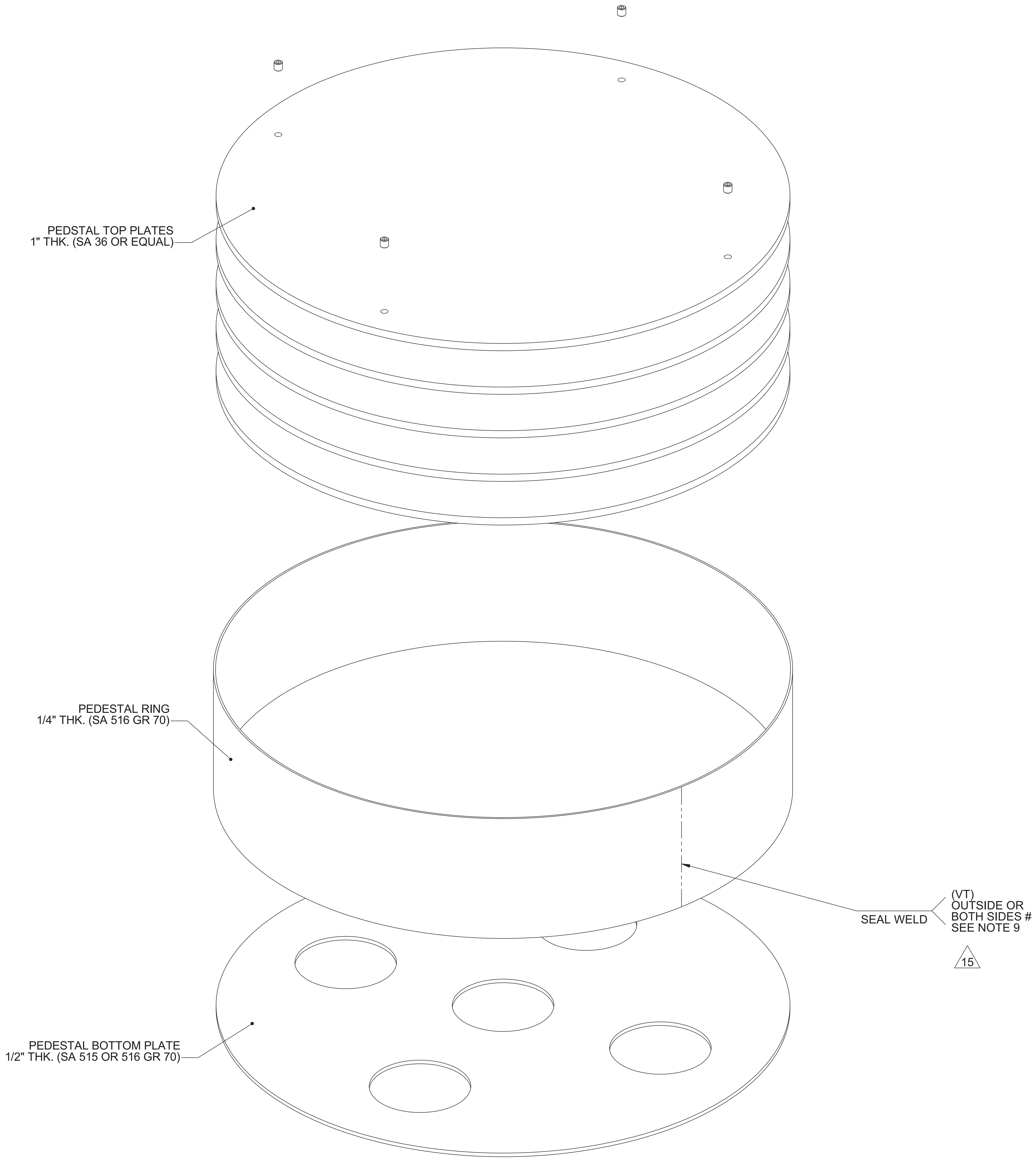
FSAR UPDATE

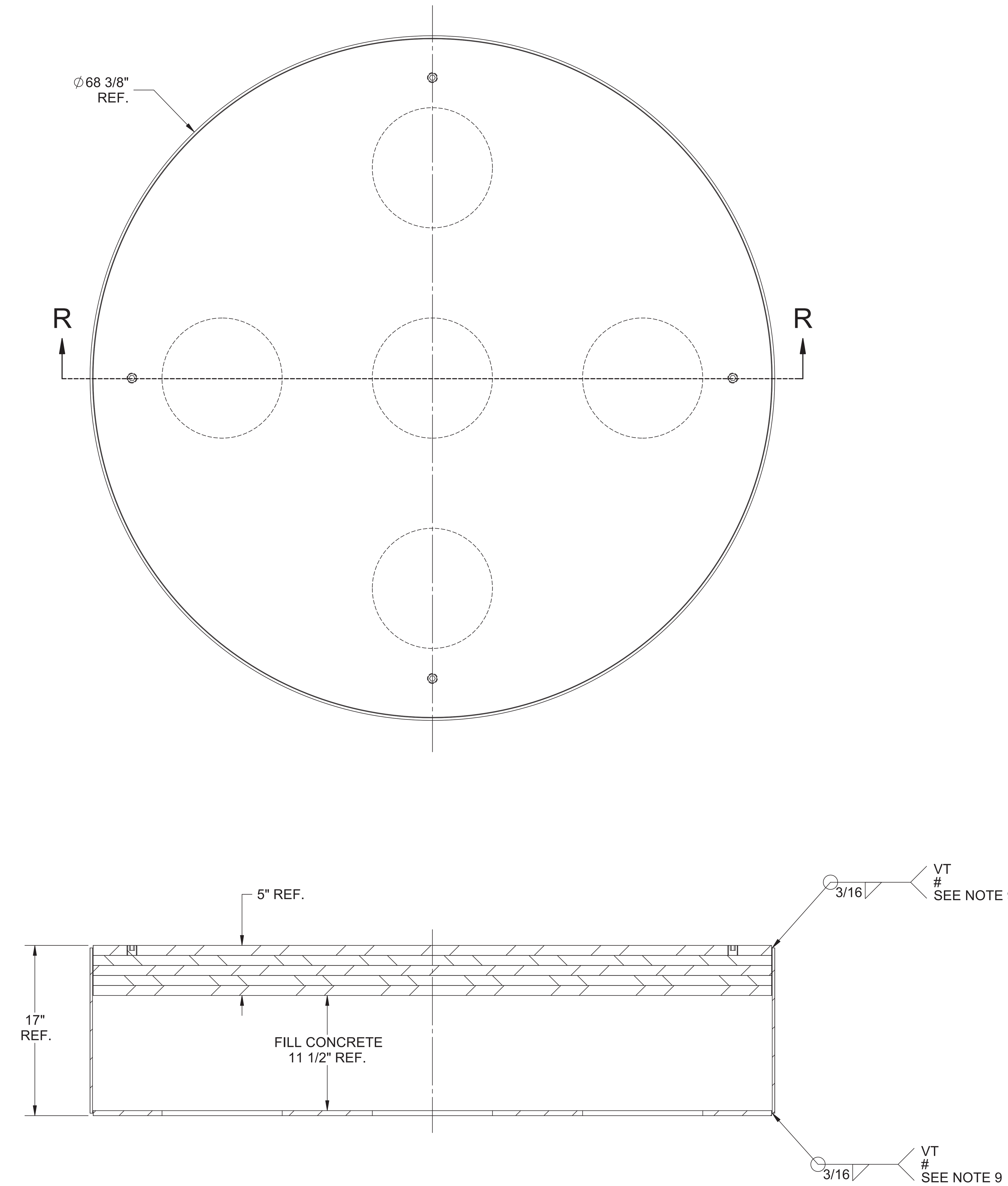
DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 10 OF 14

Revision 8 March 2020

HOLTEC INTERNATIONAL	CLIENT: PG&E
NUCLEAR POWER DIVISION, HOLTEC CENTER, ONE HOLTEC DRIVE, MARLTON, NJ 08053 (USA)	DESCRIPTION: HI-STORM 100SA CLOSURE LID DETAILS
PROJECT NO. 1073 / 3100813919	COMPANION DRAWINGS: -
SIZE: D	DRAWING NO.: 4461
SHEET: 10	REV: 15
SCALE: NONE	FILE PATH: G:\DRAWINGS\1024\4461





SECTION R-R

FSAR UPDATE

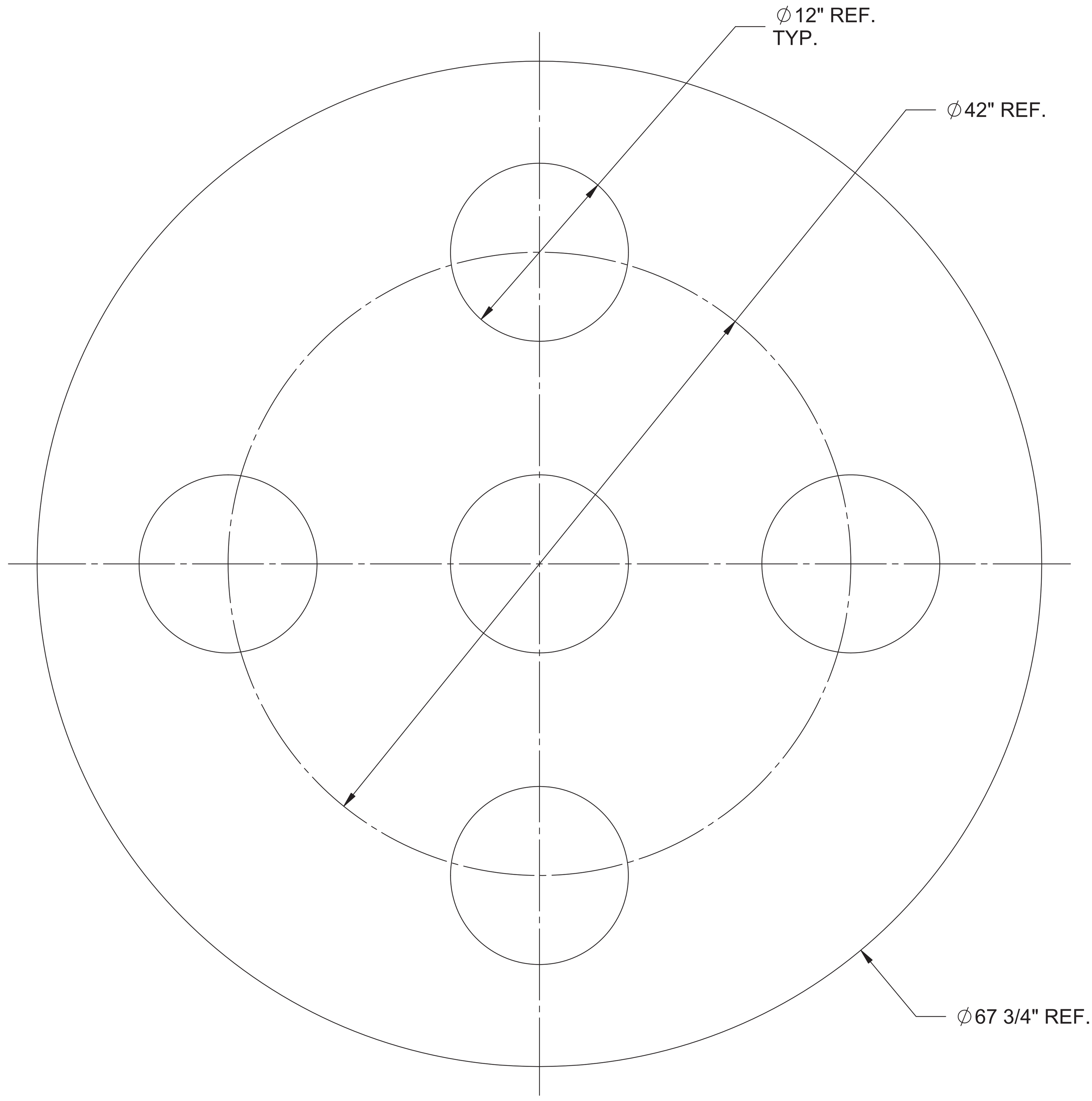
DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 11 OF 14

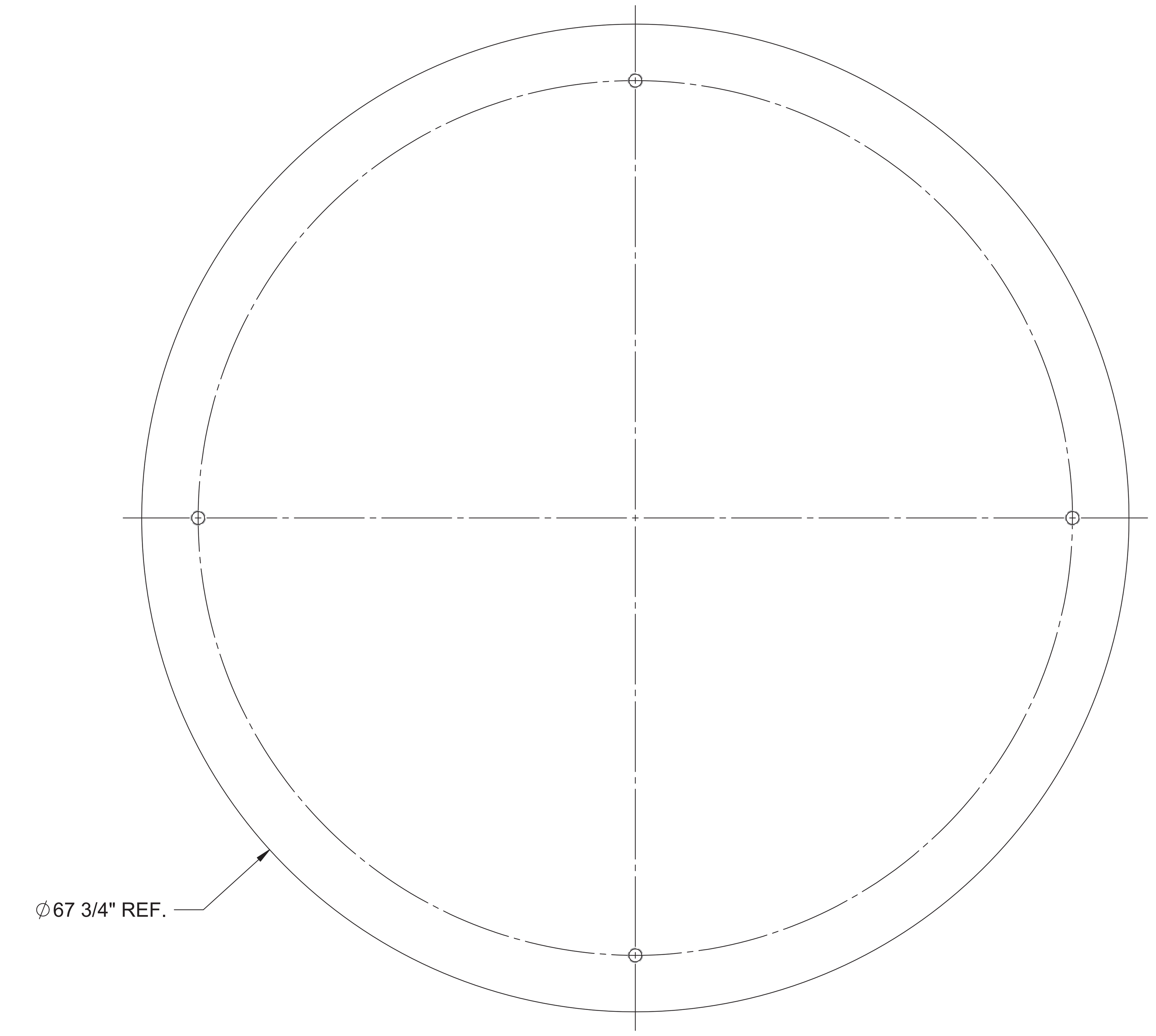
Revision 8 March 2020

HOLTEC INTERNATIONAL
NUCLEAR POWER DIVISION
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)
15

CLIENT	PG&E
DESCRIPTION	HI-STORM 100SA PEDESTAL ASSEMBLY WELDMENT
PROJECT NO.	1073
	3100813919
COMPANION DRAWINGS:	-
SIZE	D
DRAWING NO.	4461
SHEET	11
REV	15
SCALE	NONE
FILE PATH:	G:\DRAWINGS\1073\4461



PEDESTAL BASE PLATE



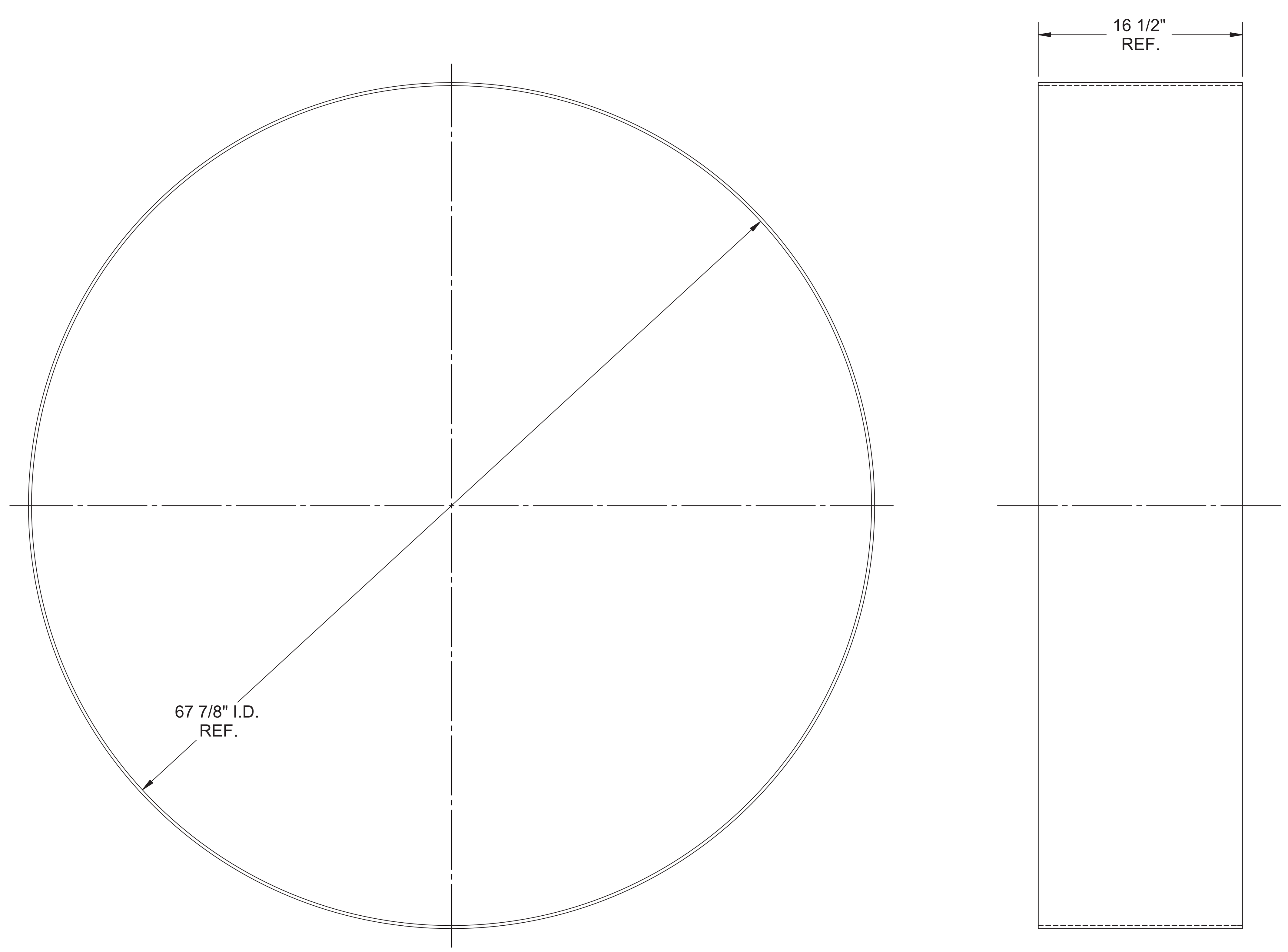
PEDESTAL TOP PLATE

FSAR UPDATE

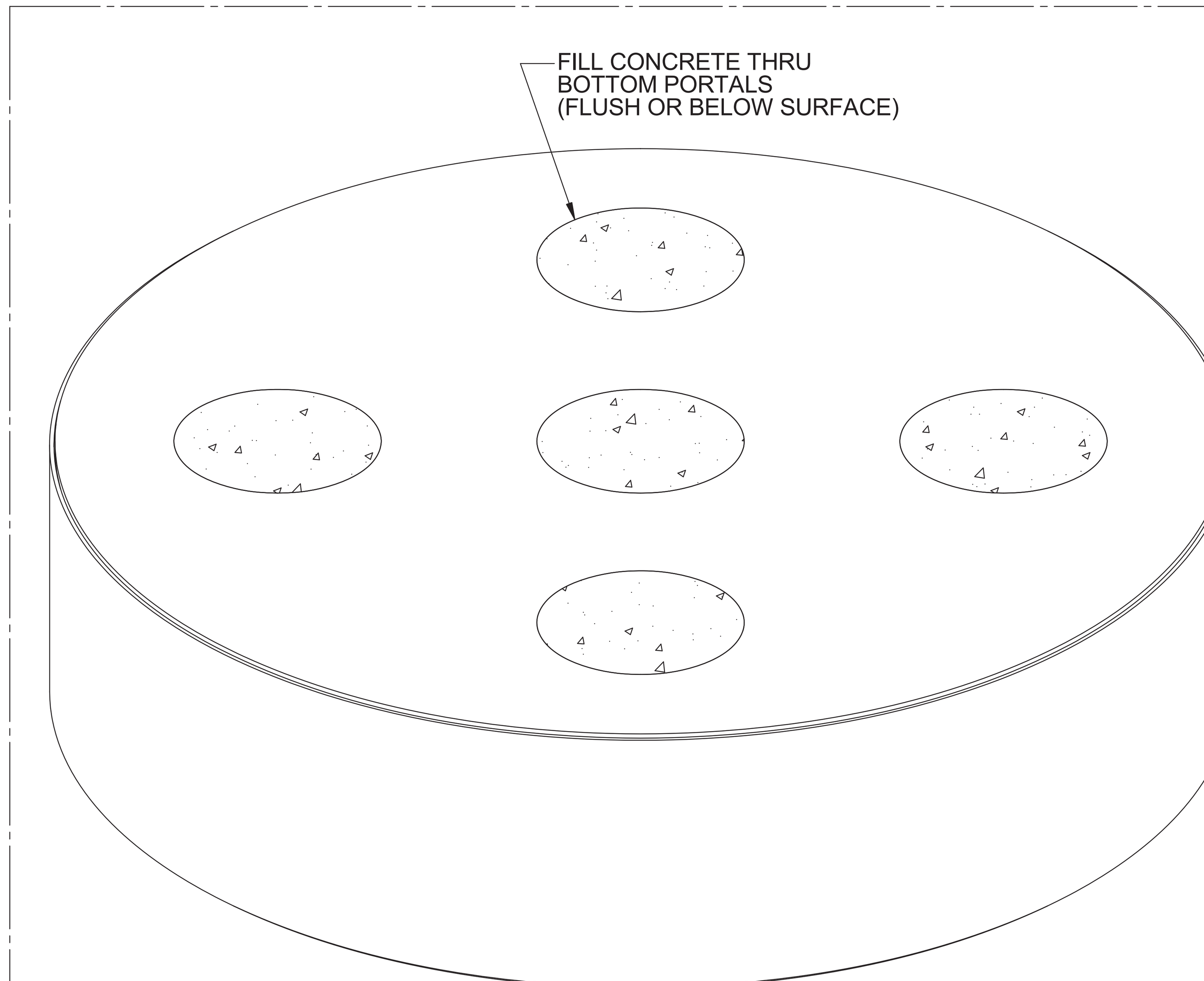
DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 12 OF 14

Revision 8 March 2020

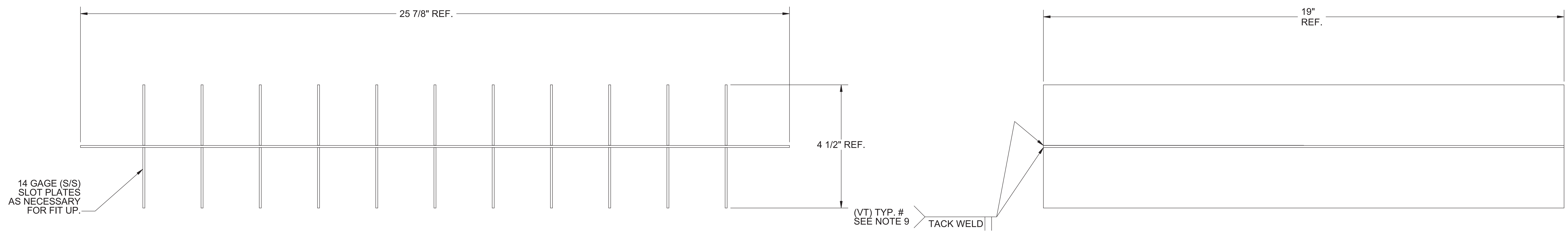


PEDESTAL RING

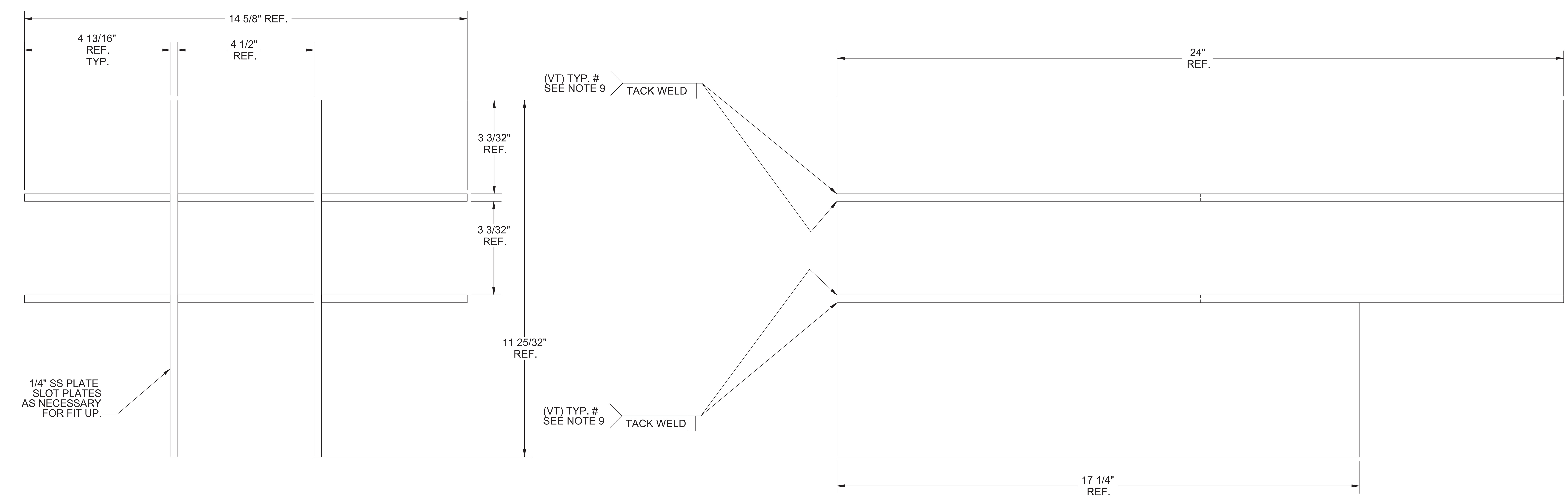


BOTTOM ISOMETRIC VIEW

HOLTEC INTERNATIONAL NUCLEAR POWER DIVISION HOLTEC CENTER ONE HOLTEC DRIVE MARLTON, NJ 08053 (USA) 15	CLIENT: PG&E DESCRIPTION: HI-STORM 100SA PEDESTAL ASSEMBLY DETAILS
PROJECT NO. 1073 3100813919	COMPANION DRAWINGS: -
SIZE: D	DRAWING NO. 4461
SHEET: 12	REV: 15
SCALE: NONE	FILE PATH: G:\DRAWINGS\1073\4461



UPPER GAMMA SHIELD CROSS ASSEMBLY



LOWER GAMMA SHIELD CROSS ASSEMBLY

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 13 OF 14

Revision 8 March 2020

HOLTEC INTERNATIONAL
NUCLEAR POWER DIVISION
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)
15

CLIENT	PG&E
DESCRIPTION	HI-STORM 100SA GAMMA SHIELDING ASSEMBLY WELDMENT
PROJECT NO.	1073
	3100813919
COMPANION DRAWINGS:	-
SIZE	D
DRAWING NO.	4461
SHEET	13
REV	15
SCALE	NONE
FILE PATH:	G:\DRAWINGS\1073\4461

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-7
HI-STORM 100SA ASSEMBLY
SHEET 14 OF 14

Revision 8 March 2020

HOLTEC INTERNATIONAL
NUCLEAR POWER DIVISION
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053 (USA)

CLIENT: PG&E
DESCRIPTION: HI-STORM 100SA ANCHORING DETAILS & ASSEMBLY

PROJECT NO. 1073
3100813919
COMPANION DRAWINGS: -
SIZE D | DRAWING NO. 4461 | SHEET 14 | REV 15
SCALE NONE | FILE PATH G:\DRAWINGS\1073\4461

LICENSING DRAWING PACKAGE COVER SHEET

CLIENT	PG&E		
PROJECT NO.	1073	P.O. NO.	3100813919
DRAWING PACKAGE I.D.	4460	TOTAL SHEETS	10

LICENSING DRAWING PACKAGE CONTENTS:

SHEET	DESCRIPTION
1	COVER SHEET
2	ASSEMBLY AND SECTION VIEW
3	ASSEMBLY DIMENSIONS
4	POOL LID ASSEMBLY
5	INNER / OUTER SHELL ASSEMBLY
6	TOP FLANGE ASSEMBLY
7	TRUNNION AND SHELL ASSEMBLY
8	WATER COMPARTMENT DETAILS
9	WATER JACKET ASSEMBLY
10	TOP LID ASSEMBLY

REVISION LOG

IT IS MANDATORY AT EACH REVISION TO COMPLETE THE REVIEW & APPROVAL LOG STORED IN HOLTEC'S DIRECTORY N:\PDOXWIN\WORKING\DBAL BY ALL RELEVANT TECHNICAL DISCIPLINES. PM AND QA PERSONNEL. EACH ATTACHED DRAWING SHEET CONTAINS ANNOTATED TRIANGLES INDICATING THE REVISION TO THE DRAWING.

REV	AFFECTED DRAWING SHEET NUMBERS	SUMMARY OF CHANGES/ AFFECTED ECOs	PREPARED BY	APPROVAL DATE	VIR# †
0	INITIAL ISSUE		JJB	04/11/05	70201
1	SHT'S 2,3,5,6 & 9	ECO- 1073-4 REV. 1	JJB	06/17/05	49190
2	SHT'S 7 & 9	ECO- 1073-10 REV. 0	JJB	06/29/05	76633
3	SHT'S 4 & 9	ECO- 1073-13 REV. 0	JJB	07/05/05	25164
4	SHT'S 2, 6, & 10	ECO- 1073-18, REV. 0	SLC	08/25/05	85762

† THE VALIDATION IDENTIFICATION RECORD (VIR) NUMBER IS A COMPUTER GENERATED RANDOM NUMBER WHICH CONFIRMS THAT ALL APPROPRIATE REVIEWS OF THIS DRAWING ARE DOCUMENTED IN COMPANY'S NETWORK.

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-8
1 of 10
HI-TRAC 125D Transfer Cask Assembly

Revision 2 June 2008

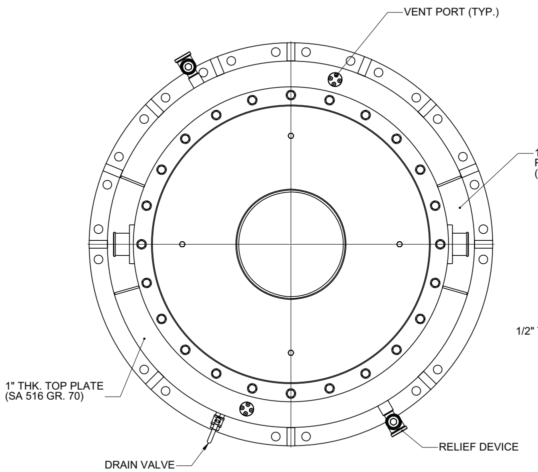
NOTES:

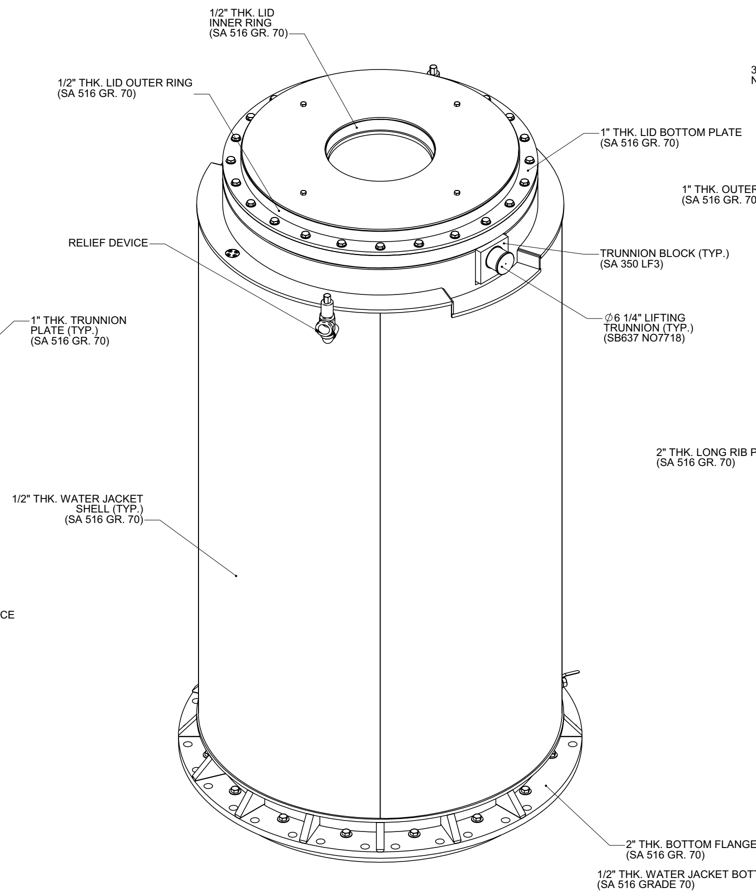
1. THE EQUIPMENT DOCUMENTED IN THIS DRAWING PACKAGE HAS BEEN CONFIRMED BY HOLTEC INTERNATIONAL TO COMPLY WITH THE SAFETY ANALYSES DESCRIBED IN THE DIABLO CANYON ISFSI FSAR.
2. DIMENSIONAL TOLERANCES ON THIS DRAWING ARE PROVIDED SOLELY FOR LICENSING PURPOSES TO DEFINE REASONABLE LIMITS ON THE NOMINAL DIMENSIONS USED IN LICENSING WORK. HARDWARE IS FABRICATED IN ACCORDANCE WITH THE FABRICATION DRAWINGS, WHICH HAVE MORE RESTRICTIVE TOLERANCES, TO ENSURE COMPONENT FIT-UP. DO NOT USE WORST-CASE TOLERANCE STACK-UP FROM THIS DRAWING TO DETERMINE COMPONENT FIT-UP.
3. THE REVISION LEVEL OF EACH INDIVIDUAL SHEET IN THE PACKAGE IS THE SAME AS THE REVISION LEVEL OF THIS COVER SHEET. A REVISION TO ANY SHEET(S) IN THIS PACKAGE REQUIRES UPDATING OF REVISION NUMBERS OF ALL SHEETS TO THE NEXT REVISION NUMBER.
4. APPLICABLE CODES AND STANDARDS ARE DESCRIBED IN D.C.P.P. ISFSI FSAR.
5. ALL WELDS REQUIRE VISUAL EXAMINATION. ADDITIONAL NDE INSPECTIONS ARE NOTED ON THE DRAWING.
6. UNLESS OTHEWISE NOTED, FULL PENETRATION WELDS MAY BE MADE FROM EITHER SIDE OF A COMPONENT.
7. THIS COMPONENT IS IMPORTANT-TO-SAFETY, CATEGORY A, BASED ON THE HIGHEST CLASSIFICATION OF ANY SUBCOMPONENT. SUBCOMPONENT CLASSIFICATIONS ARE PROVIDED ON THE FABRICATION DRAWING.
8. ALL WELD SIZES ARE MINIMUMS, FABRICATOR MAY ADD WELDS WITH HOLTEC APPROVAL.
9. CHANGES TO THIS DRAWING AFFECT FABRICATION DRAWING 4426.

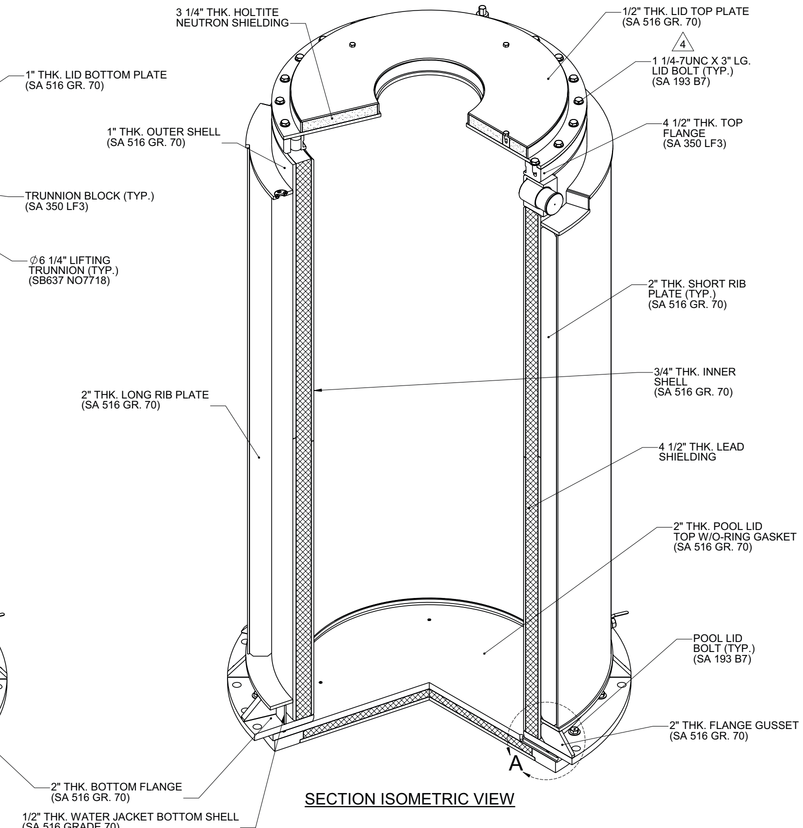
ISOMETRIC VIEW OF HI-TRAC 125D TRANSFER CASK

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

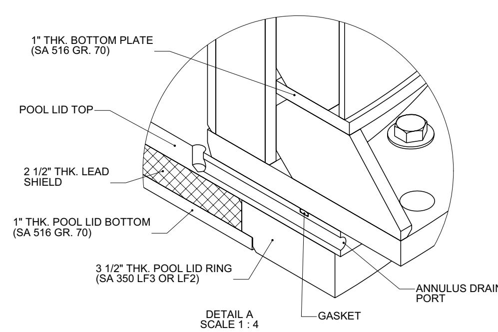
CLIENT	PG&E
DESCRIPTION	125 TON HI-TRAC 125D ASSEMBLY
PROJECT NO.	1073
P.O. NO.	3100813919
DRAWING NO.	4460
SHEET	1
TOTAL SHEETS	10
FILE PATH	G:\DRAWINGS\1073\4460







SECTION ISOMETRIC VIEW



DETAIL A
SCALE 1 : 4

FSAR UPDATE

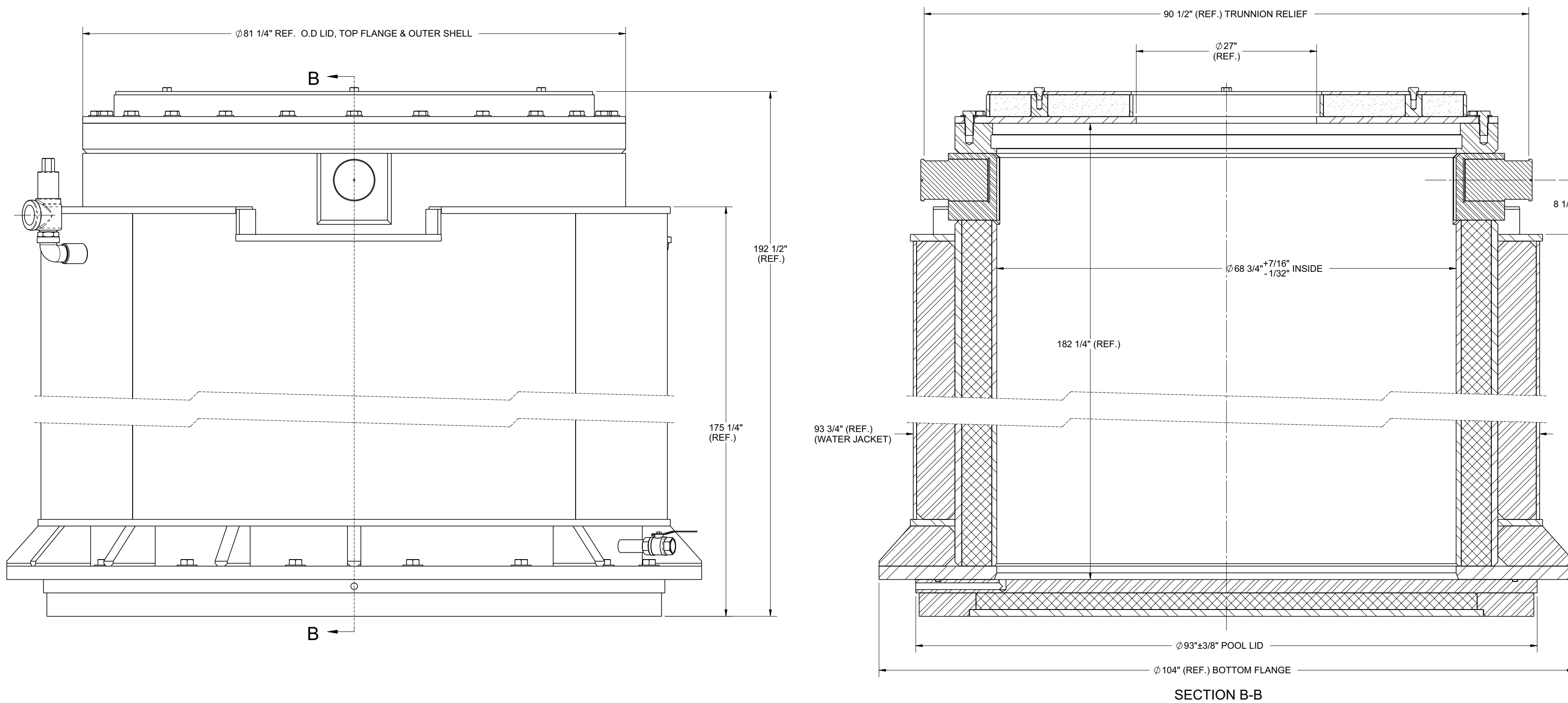
DIABLO CANYON ISFSI

FIGURE 4.2-8
2 of 10
HI-TRAC 125D Transfer Cask Assembly

Revision 2 June 2008

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

CLIENT	PG&E
DESCRIPTION	125 TON HI-TRAC 125D ASSEMBLY
PROJECT NO.	1073
	3100813919
COMPANION DRAWINGS:	-
SIZE	D
DRAWING NO.	4460
SHEET	2
REV	4
SCALE	NONE
FILE PATH	G:\DRAWINGS\1073\4460



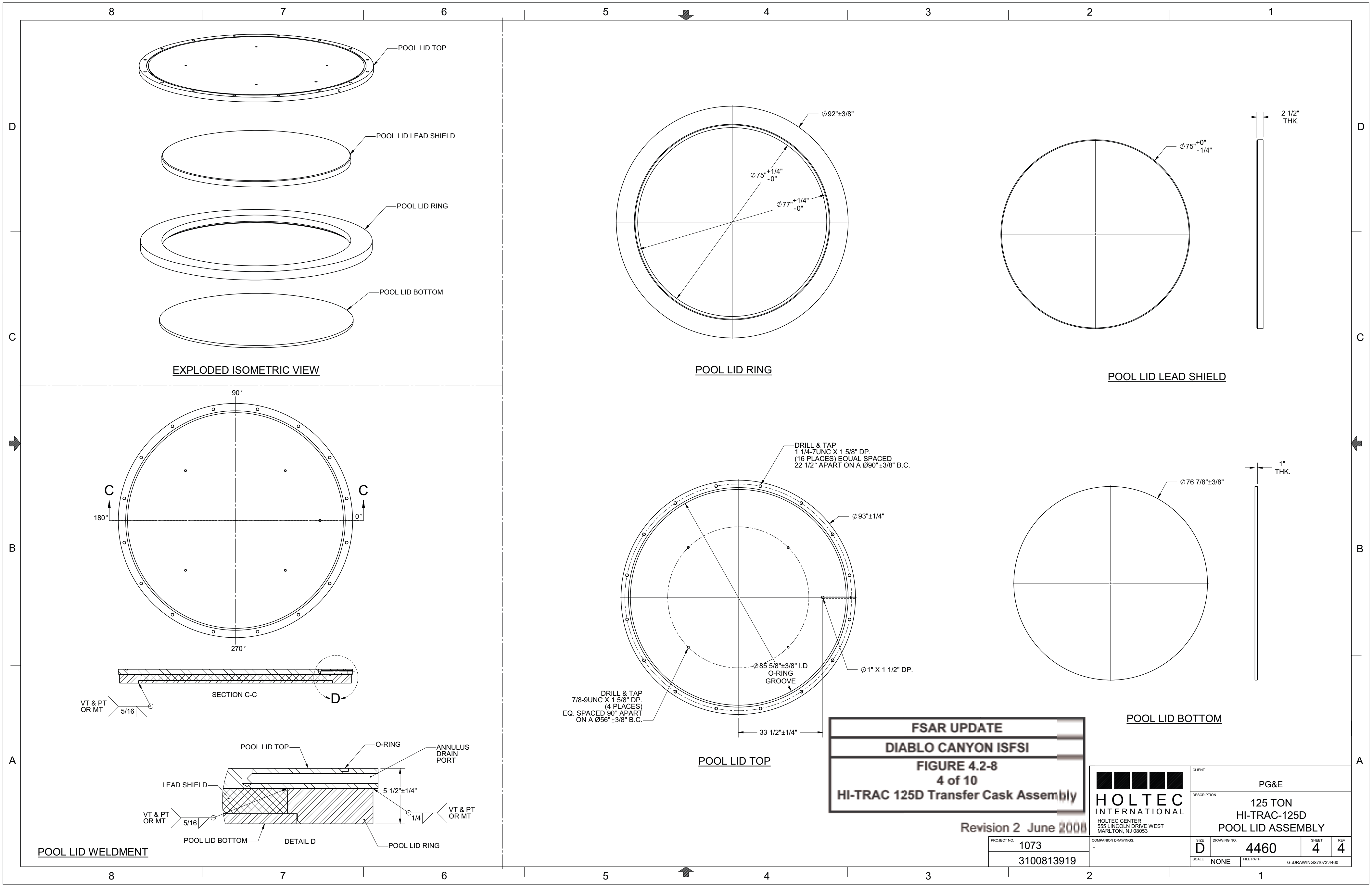
FSAR UPDATE

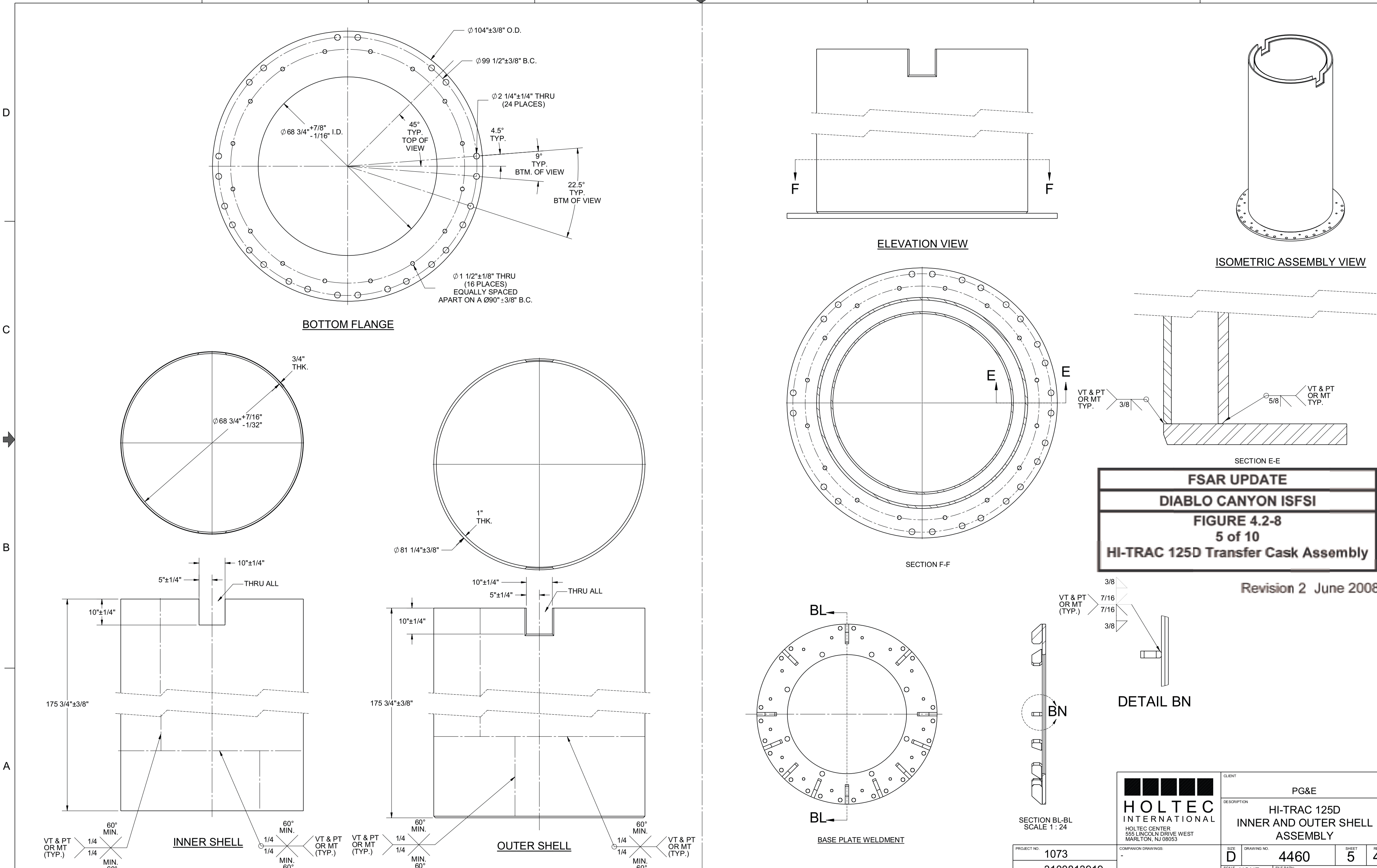
DIABLO CANYON ISFSI

FIGURE 4.2-8
3 of 10
HI-TRAC 125D Transfer Cask Assembly

Revision 2 June 2008

HOLTEC INTERNATIONAL, HOLTEC CENTER, 555 LINCOLN DRIVE WEST, MARLTON, NJ 08053	CLIENT	PG&E
	DESCRIPTION	125 TON HI-TRAC 125D ASSEMBLY
PROJECT NO. 1073 / 3100813919	COMPANION DRAWINGS: -	SIZE D, DRAWING NO. 4460, SHEET 3, REV 4
		SCALE NONE, FILE PATH G:\DRAWINGS\1073\4460


POOL LID TOP
POOL LID LEAD SHIELD
POOL LID RING
POOL LID BOTTOM
EXPLODED ISOMETRIC VIEW
Ø92"±3/8"
Ø75" +1/4" -0"
Ø77" +1/4" -0"
POOL LID RING
Ø75" +0" -1/4"
2 1/2" THK.
POOL LID LEAD SHIELD
90°
180°
0°
270°
C
C
SECTION C-C
D
VT & PT OR MT
5/16
POOL LID TOP
O-RING
ANNULUS DRAIN PORT
LEAD SHIELD
5 1/2"±1/4"
VT & PT OR MT
5/16
1/4
VT & PT OR MT
POOL LID BOTTOM
DETAIL D
POOL LID RING
POOL LID WELDMENT
DRILL & TAP 1 1/4-7UNC X 1 5/8" DP. (16 PLACES) EQUAL SPACED 22 1/2° APART ON A Ø90"±3/8" B.C.
Ø93"±1/4"
Ø85 5/8"±3/8" I.D O-RING GROOVE
Ø1" X 1 1/2" DP.
DRILL & TAP 7/8-9UNC X 1 5/8" DP. (4 PLACES) EQ. SPACED 90° APART ON A Ø56" ±3/8" B.C.
33 1/2"±1/4"
POOL LID TOP
Ø76 7/8"±3/8"
1" THK.
POOL LID BOTTOM
FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 4.2-8
4 of 10
HI-TRAC 125D Transfer Cask Assembly
Revision 2 June 2008
HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053
CLIENT
PG&E
DESCRIPTION
125 TON
HI-TRAC-125D
POOL LID ASSEMBLY
PROJECT NO.
1073
3100813919
COMPANION DRAWINGS:
-
SIZE
D
DRAWING NO.
4460
SHEET
4
REV
4
SCALE
NONE
FILE PATH
G:\DRAWINGS\1073\4460



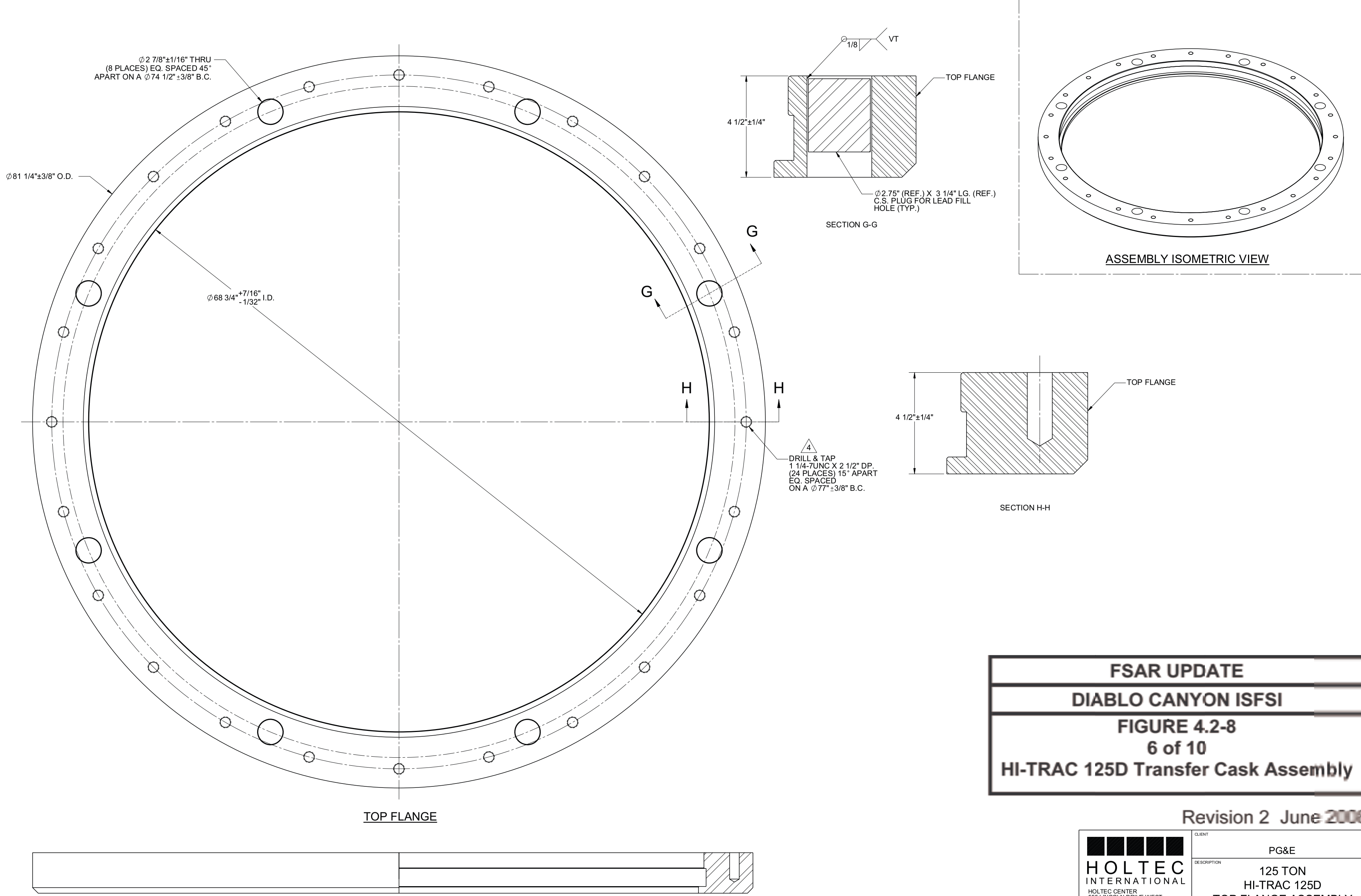
FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-8
5 of 10
HI-TRAC 125D Transfer Cask Assembly

Revision 2 June 2008

HOLTEC INTERNATIONAL	HOLTEC CENTER, 555 LINCOLN DRIVE WEST, MARLTON, NJ 08053
CLIENT	PG&E
DESCRIPTION	HI-TRAC 125D INNER AND OUTER SHELL ASSEMBLY
PROJECT NO.	1073
	3100813919
COMPANION DRAWINGS	-
SIZE	D
DRAWING NO.	4460
SHEET	5
REV.	4
SCALE	NONE
FILE PATH	G:\DRAWINGS\1073\4460



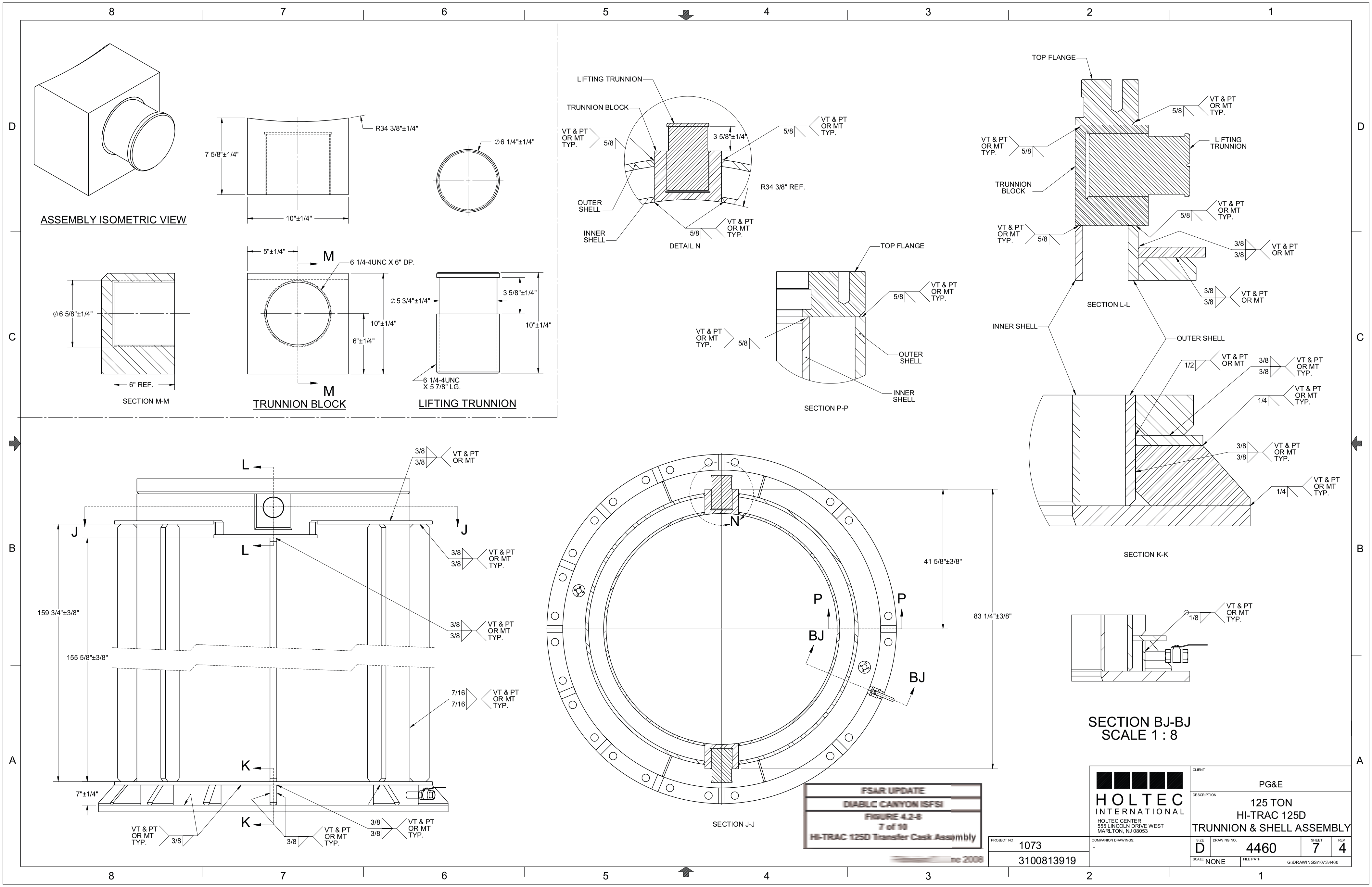
FSAR UPDATE

DIABLO CANYON ISFSI

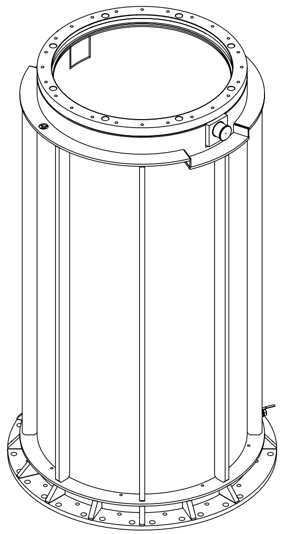
FIGURE 4.2-8
6 of 10
HI-TRAC 125D Transfer Cask Assembly

Revision 2 June 2008

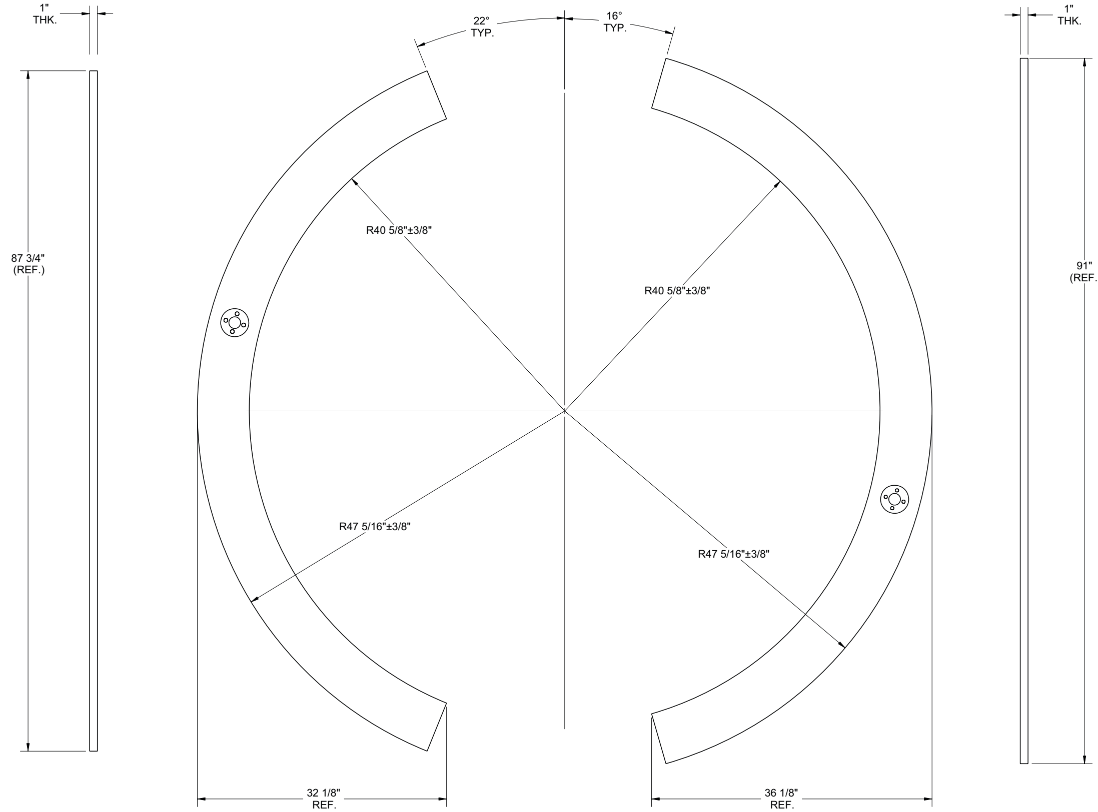
HOLTEC INTERNATIONAL	CLIENT: PG&E
HOLTEC CENTER 555 LINCOLN DRIVE WEST MARLTON, NJ 08053	DESCRIPTION: 125 TON HI-TRAC 125D TOP FLANGE ASSEMBLY
PROJECT NO. 1073 / 3100813919	COMPANION DRAWINGS: -
SIZE D	DRAWING NO. 4460
SHEET 6	REV. 4
SCALE NONE	FILE PATH: G:\DRAWINGS\1073\4460



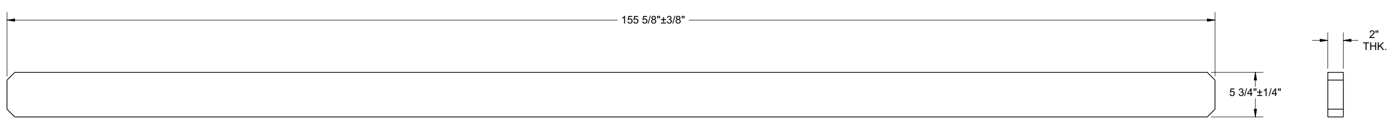
ASSEMBLY ISOMETRIC VIEW
R34 3/8"±1/4"
7 5/8"±1/4"
10"±1/4"
Ø6 1/4"±1/4"
Ø6 5/8"±1/4"
6" REF.
SECTION M-M
5"±1/4"
M
6 1/4-4UNC X 6" DP.
6"±1/4"
10"±1/4"
TRUNNION BLOCK
Ø5 3/4"±1/4"
3 5/8"±1/4"
6 1/4-4UNC X 5 7/8" LG.
LIFTING TRUNNION
LIFTING TRUNNION
TRUNNION BLOCK
VT & PT OR MT TYP.
5/8
3 5/8"±1/4"
R34 3/8" REF.
OUTER SHELL
INNER SHELL
DETAIL N
TOP FLANGE
SECTION P-P
SECTION L-L
SECTION K-K
VT & PT OR MT
1/2
1/4
3/8
7/16
1/8
L
J
K
N
P
BJ
159 3/4"±3/8"
155 5/8"±3/8"
7"±1/4"
41 5/8"±3/8"
83 1/4"±3/8"
SECTION J-J
SECTION BJ-BJ
SCALE 1 : 8
FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 4.2-8
7 of 10
HI-TRAC 125D Transfer Cask Assembly
ne 2008
HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053
CLIENT PG&E
DESCRIPTION 125 TON HI-TRAC 125D TRUNNION & SHELL ASSEMBLY
PROJECT NO. 1073
3100813919
COMPANION DRAWINGS: -
SIZE D
DRAWING NO. 4460
SHEET 7
REV 4
SCALE NONE
FILE PATH: G:\DRAWINGS\1073\4460



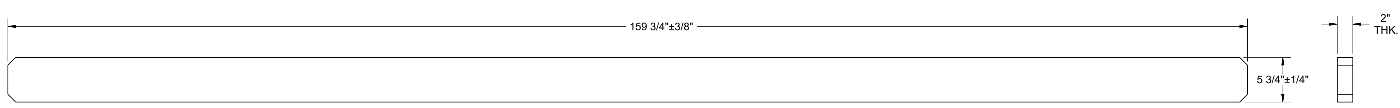
ASSEMBLY ISOMETRIC VIEW



WATER JACKET TOP PLATE



SHORT RIB PLATE



LONG RIB PLATE

FSAR UPDATE

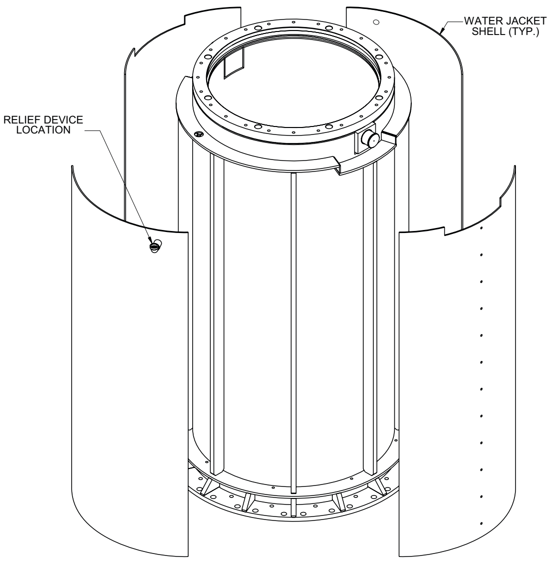
DIABLO CANYON ISFSI

FIGURE 4.2-8
8 of 10
HI-TRAC 125D Transfer Cask Assembly

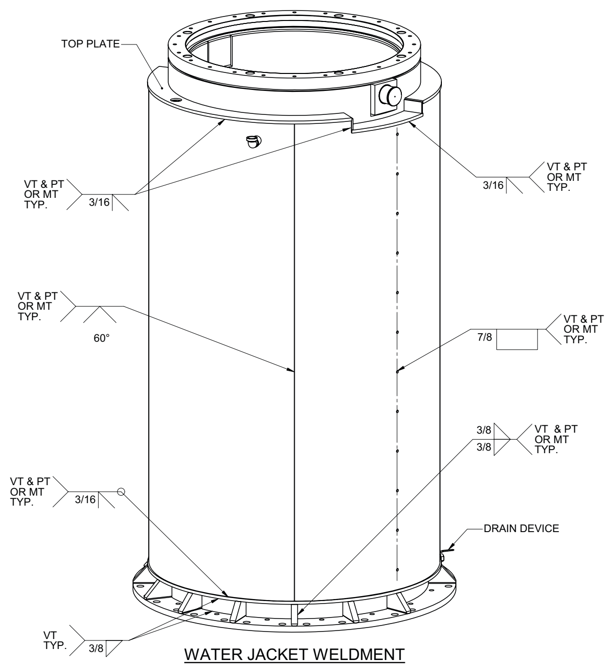
Revision 2 June 2008

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

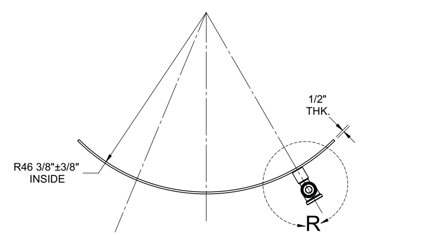
CLIENT	PG&E
DESCRIPTION	HI-TRAC 125D WATER COMPARTMENT DETAILS
PROJECT NO.	1073
	N/A
COMPANION DRAWINGS:	-
SIZE	D
DRAWING NO.	4460
SHEET	8
REV	4
SCALE	NONE
FILE PATH	G:\DRAWINGS\1073\4460

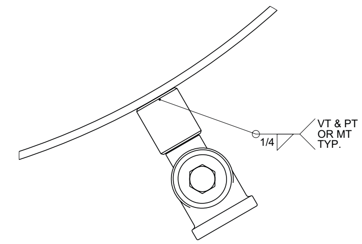


ISOMETRIC ASSEMBLY VIEW

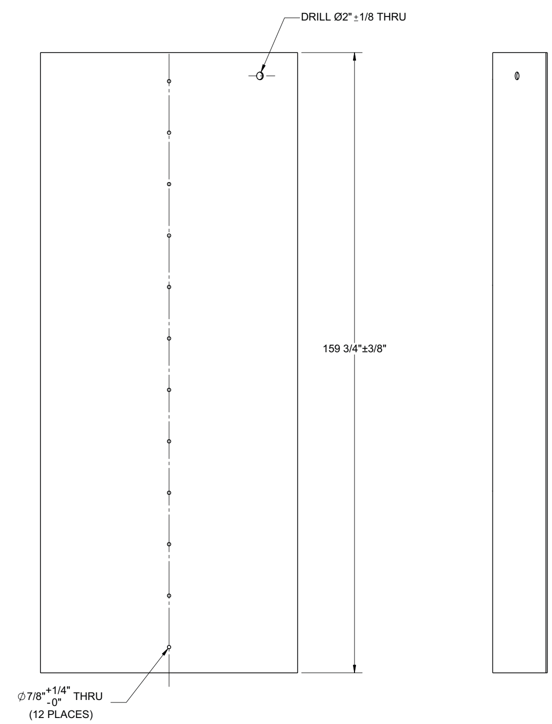


WATER JACKET WELDMENT





DETAIL R
RELIEF DEVICE
(ACTUAL CONFIGURATION MAY VARY)



WATER JACKET SHELL

FSAR UPDATE

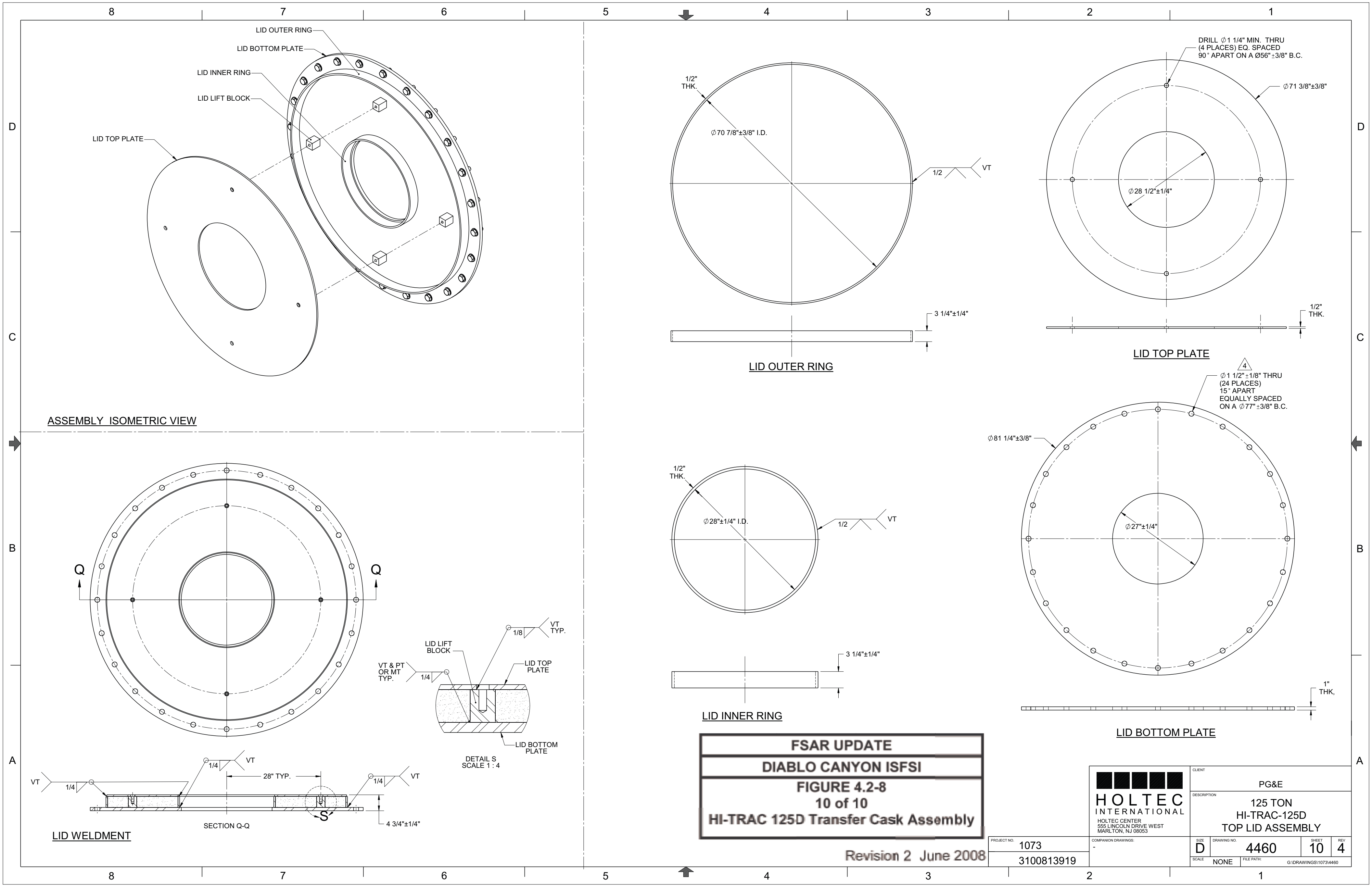
DIABLO CANYON ISFSI

FIGURE 4.2-8
9 of 10
HI-TRAC 125D Transfer Cask Assembly

Revision 2 June 2008

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

CLIENT	PG&E
DESCRIPTION	125 TON HI-TRAC-125D WATER JACKET ASSEMBLY
PROJECT NO.	1073 3100813919
COMPANION DRAWINGS:	-
SIZE	D
DRAWING NO.	4460
SHEET	9
REV	4
SCALE	NONE
FILE PATH	G:\DRAWINGS\1073\4460



FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-8
10 of 10
HI-TRAC 125D Transfer Cask Assembly

Revision 2 June 2008

LICENSING DRAWING PACKAGE COVER SHEET

CLIENT PG&E

PROJECT NO. 1073 | P.O. NO. 3100813919

DRAWING PACKAGE I.D. 4459 | TOTAL SHEETS 7 15

LICENSING DRAWING PACKAGE CONTENTS:

SHEET	DESCRIPTION
1	COVER SHEET
2	MPC ENCLOSURE VESSEL GENERAL ARRANGEMENT
3	MPC ENCLOSURE VESSEL ELEVATION DETAILS
4	MPC ENCLOSURE VESSEL LID DETAILS
5	MPC FUEL SPACER DETAILS
6	MPC FUEL BASKET SUPPORTS
7	OPTIONAL MPC LOWER FUEL SPACER DETAILS (15)

REVISION LOG

IT IS MANDATORY AT EACH REVISION TO COMPLETE THE REVIEW & APPROVAL LOG STORED IN HOLTEC'S DIRECTORY N:\PDOXWIN\WORKING\DBAL BY ALL RELEVANT TECHNICAL DISCIPLINES. PM AND QA PERSONNEL. EACH ATTACHED DRAWING SHEET CONTAINS ANNOTATED TRIANGLES INDICATING THE REVISION TO THE DRAWING.

REV	AFFECTED DRAWING SHEET NUMBERS	SUMMARY OF CHANGES/ AFFECTED ECOs	PREPARED BY	APPROVAL DATE	VIR# †
11	SHEETS 2, 3, & 6	ECO 1073-112 & ECO 1073-113	MDB	03/26/2013	76584
12	SHEET 2	ECO 1073-115, REV 0	MS	01/02/2015	11100
13	ALL SHEETS	ECO 1073-116, REV 0	SLC	06/11/2015	92122
14	1, 2, 3 & 6	ECO 1073-122, REV 0	SMS	12/23/2015	94160
15	1, 2, & 7	ECO 1073-123, REV 0	DCB	SEE BELOW	SEE BELOW

† THE VALIDATION IDENTIFICATION RECORD (VIR) NUMBER IS A COMPUTER GENERATED RANDOM NUMBER WHICH CONFIRMS THAT ALL APPROPRIATE REVIEWS OF THIS DRAWING ARE DOCUMENTED IN COMPANY'S NETWORK.

GENERAL NOTES CONTINUED:

11. THIS COMPONENT IS CLASSIFIED AS ITS-A BASED ON THE HIGHEST CLASSIFICATION OF ANY SUBCOMPONENT. SUBCOMPONENT CLASSIFICATIONS ARE PROVIDED IN SAR TABLE 1.3.3 (TRANSPORTATION) AND ARE DISCUSSED IN DC ISFSI FSAR SECTION 4.5 (STORAGE).
12. NOT USED.
13. LOCAL GRINDING OF THE MPC SHELL SHALL NOT RESULT IN GREATER THAN 10% LOSS IN BASE METAL THICKNESS OVER ANY AREA WHICH EXCEEDS 4 INCHES IN THE LONGITUDINAL OR CIRCUMFERENTIAL DIRECTION. FOR THE WELD AREA AND ADJACENT BASE METAL, AN ADDITIONAL ALLOWANCE OF 0.02" LOSS IN NOMINAL BASE METAL THICKNESS IS ALLOWED OVER THE ENTIRE LENGTH OF THE WELD. THE SUM OF ALL AREAS OF LOCAL METAL LOSS SHALL NOT EXCEED 10% OF THE OVERALL INSIDE SURFACE AREA OF THE MPC SHELL. FINAL THICKNESSES IN LOCAL AREAS OF GRINDING SHALL BE CONFIRMED BY UT EXAMINATION, AS APPROPRIATE.
14. FOR NON-CODE WELDS, THE PROVISIONS OF EITHER ASME IX OR AWS MAY BE FOLLOWED.
15. TOLERANCES FOR THICKNESS OF ASME CODE MPC ENCLOSURE VESSEL MATERIAL ARE SPECIFIED IN ASME SECTION II.
16. DELETED.
17. DIMENSIONS NOTED AS NOMINAL ("NOM.") IN THIS DRAWING ARE FOR INFORMATION ONLY, IN ORDER TO INDICATE THE GENERAL SIZE OF THE COMPONENT OR PART.

FSAR UPDATE

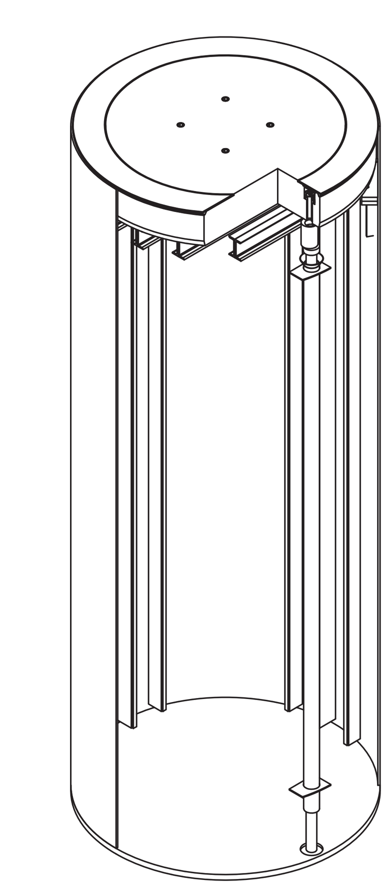
DIABLO CANYON ISFSI

FIGURE 4.2-13
MPC-32 ASSEMBLY
SHEET 1 OF 7

Revision 8 March 2020

GENERAL NOTES:

1. DELETED.
2. DIMENSIONAL TOLERANCES ON THIS DRAWING ARE PROVIDED TO ENSURE THAT THE EQUIPMENT DESIGN IS CONSISTENT WITH THE SUPPORTING ANALYSES. HARDWARE IS FABRICATED IN ACCORDANCE WITH THE DESIGN DRAWINGS, WHICH MAY HAVE MORE RESTRICTIVE TOLERANCES, TO ENSURE COMPONENT FIT-UP. DO NOT USE WORST-CASE TOLERANCE STACK-UP FROM THIS DRAWING TO DETERMINE COMPONENT FIT-UP.
3. THE REVISION LEVEL OF EACH INDIVIDUAL SHEET IN THIS PACKAGE IS THE SAME AS THE REVISION LEVEL OF THIS COVER SHEET. A REVISION TO ANY SHEET(S) IN THIS PACKAGE REQUIRES UPDATING OF REVISION NUMBERS OF ALL SHEETS TO THE NEXT REVISION NUMBER.
4. THE ASME BOILER AND PRESSURE VESSEL CODE (ASME CODE), 1995 EDITION WITH ADDENDA THROUGH 1997 IS THE GOVERNING CODE FOR THE MPC ENCLOSURE VESSEL, WITH CERTAIN APPROVED ALTERNATIVES AS LISTED IN SAR TABLE 1.3.2 (HI-STAR 100 TRANSPORTATION) AND DC ISFSI FSAR TABLE 3.4-6. THE MPC ENCLOSURE VESSEL IS CONSTRUCTED IN ACCORDANCE WITH ASME SECTION III, SUBSECTION NB. THE MPC BASKET SUPPORTS ARE CONSTRUCTED IN ACCORDANCE WITH ASME SECTION III, SUBSECTION NG. NEW OR REVISED ASME CODE ALTERNATIVES REQUIRE PRIOR NRC APPROVAL BEFORE IMPLEMENTATION.
5. ALL MPC ENCLOSURE VESSEL STRUCTURAL MATERIALS ARE "ALLOY X" UNLESS OTHERWISE NOTED. ALLOY X IS ANY OF THE FOLLOWING STAINLESS STEEL TYPES: 316, 316LN, 304, AND 304LN. ALLOY X MATERIAL MUST COMPLY WITH ASME SECTION II, PART A. WELD MATERIAL COMPLIES WITH ASME SECTION II, PART C. MPC ENCLOSURE VESSEL WALL (I.E. CYLINDER SHELL) WILL BE FABRICATED OF PIECES MADE FROM THE SAME TYPE OF STAINLESS STEEL.
6. ALL WELDS REQUIRE VISUAL EXAMINATION (VT). ADDITIONAL NDE INSPECTIONS ARE NOTED ON THE DRAWING AS REQUIRED. NDE TECHNIQUES AND ACCEPTANCE CRITERIA ARE GOVERNED BY ASME SECTIONS V AND III, RESPECTIVELY, AS CLARIFIED IN THE APPLICABLE HOLTEC SAFETY ANALYSIS REPORT.
7. UNLESS OTHERWISE NOTED, FULL PENETRATION WELDS MAY BE MADE FROM EITHER SIDE OF A COMPONENT.
8. DELETED.
9. NOT USED.
10. ALL WELD SIZES ARE MINIMUMS. LARGER WELDS ARE PERMITTED. LOCAL AREAS OF UNDERSIZE WELDS ARE ACCEPTABLE WITHIN THE LIMITS SPECIFIED IN THE ASME CODE, AS APPLICABLE.



MPC ENCLOSURE VESSEL ISOMETRIC VIEW

HOLTEC INTERNATIONAL
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053

CLIENT: PG&E

DESCRIPTION: DIABLO CANYON ENCLOSURE VESSEL 15

PROJECT NO. 1073 | DRAWING NO. 4459 | SHEET 1 | TOTAL SHEETS 7

P.O. NO. 3100813919

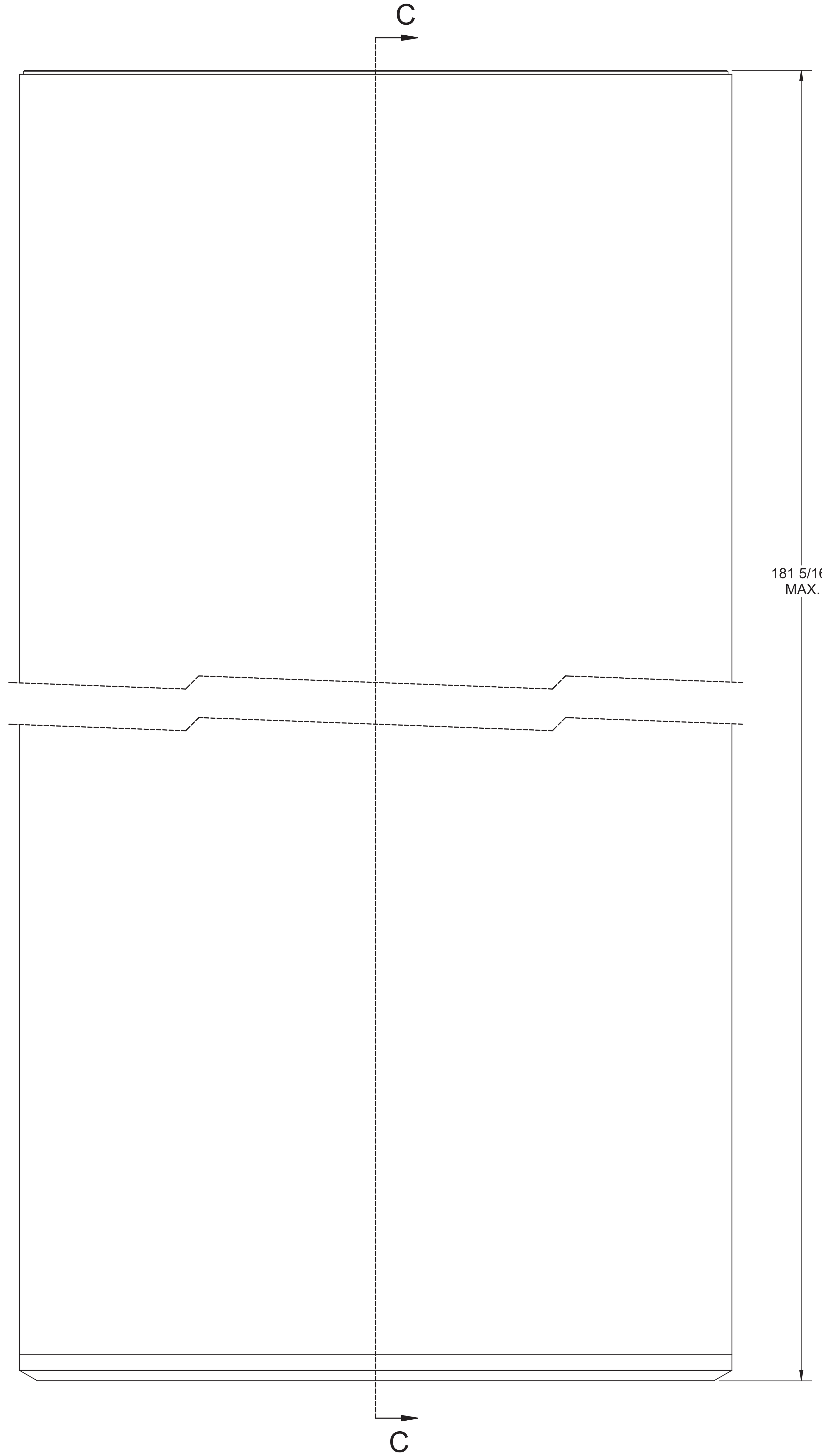
FILE PATH: G:\DRAWINGS\1073\4459

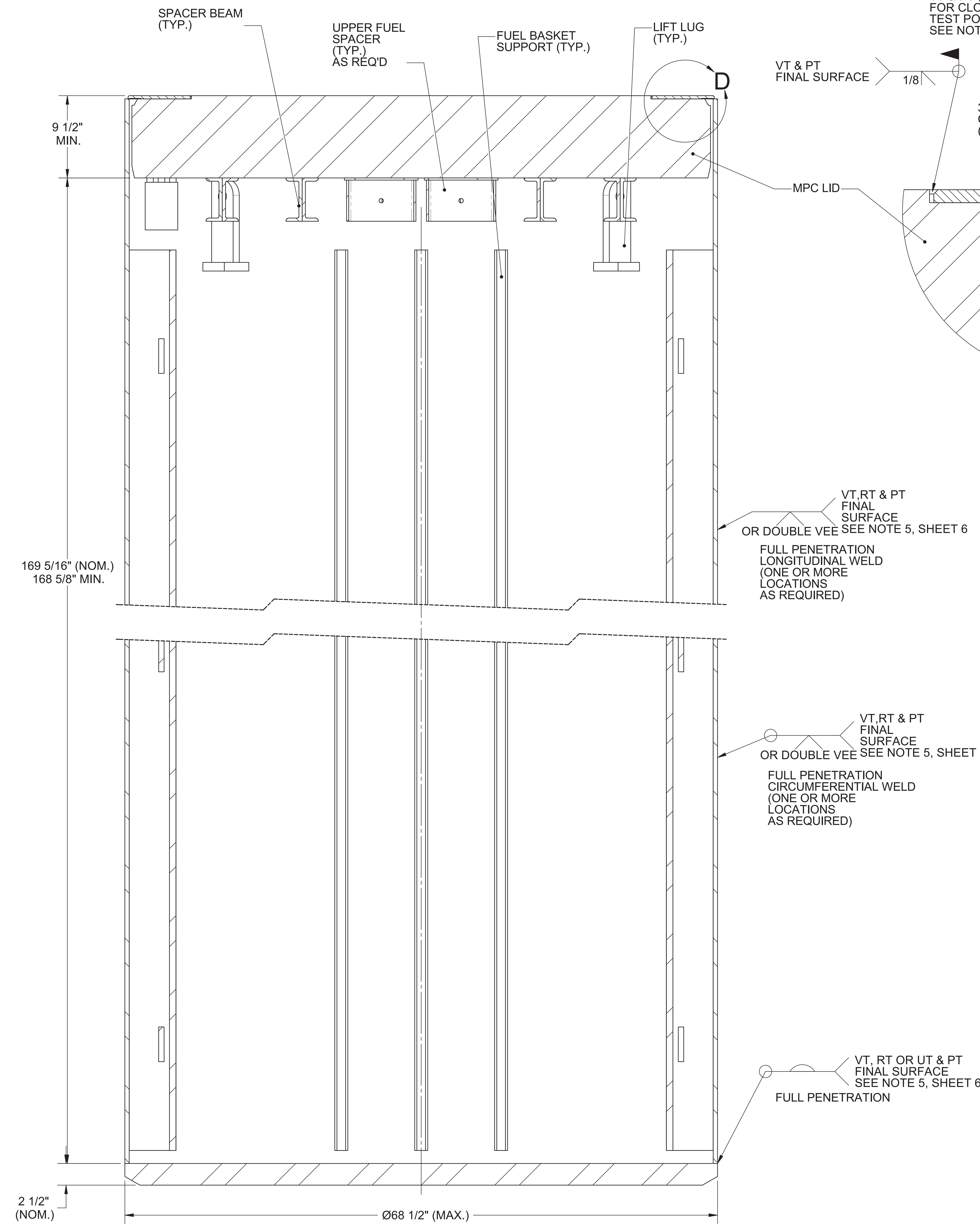
FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 4.2-13 MPC-32 ASSEMBLY SHEET 2 OF 7

Revision 8 March 2020

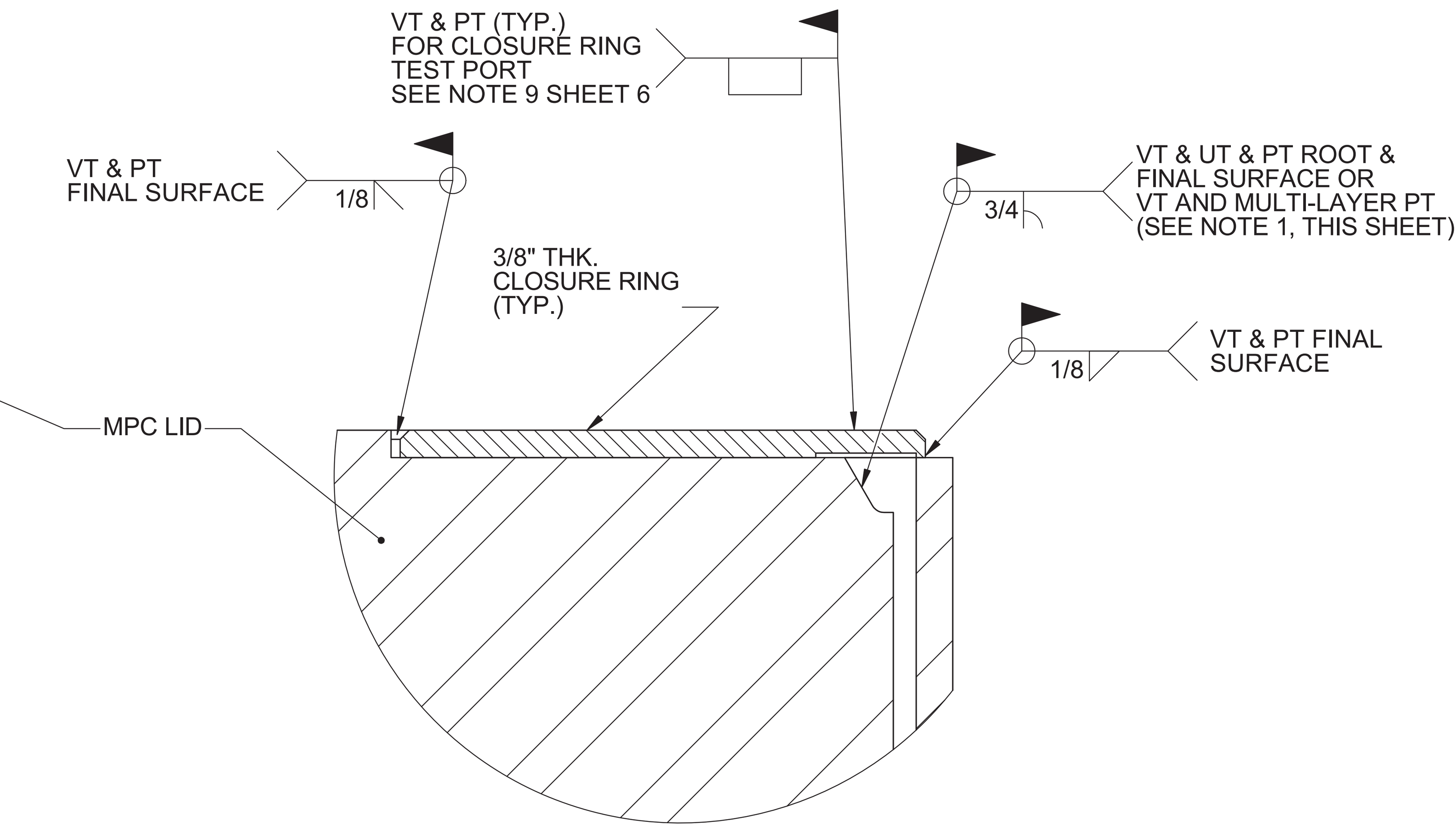
HOLTEC INTERNATIONAL
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053

CLIENT	PG&E
DESCRIPTION	MPC ENCLOSURE VESSEL GENERAL ARRANGEMENT
PROJECT NO.	1073
	3100813919
COMPANION DRAWINGS	4407,4408,4458
SIZE	D
DRAWING NO.	4459
SHEET	2
REV.	15
SCALE	NONE
FILE PATH	G:\DRAWINGS\1073\4459





SECTION C-C



DETAIL D

NOTE:

1. MULTI-LAYER PT SHALL INCLUDE ROOT AND FINAL PASSES AND APPROXIMATELY EACH 3/8" OF WELD DEPTH.

CONTINUED ON SHEET 6.

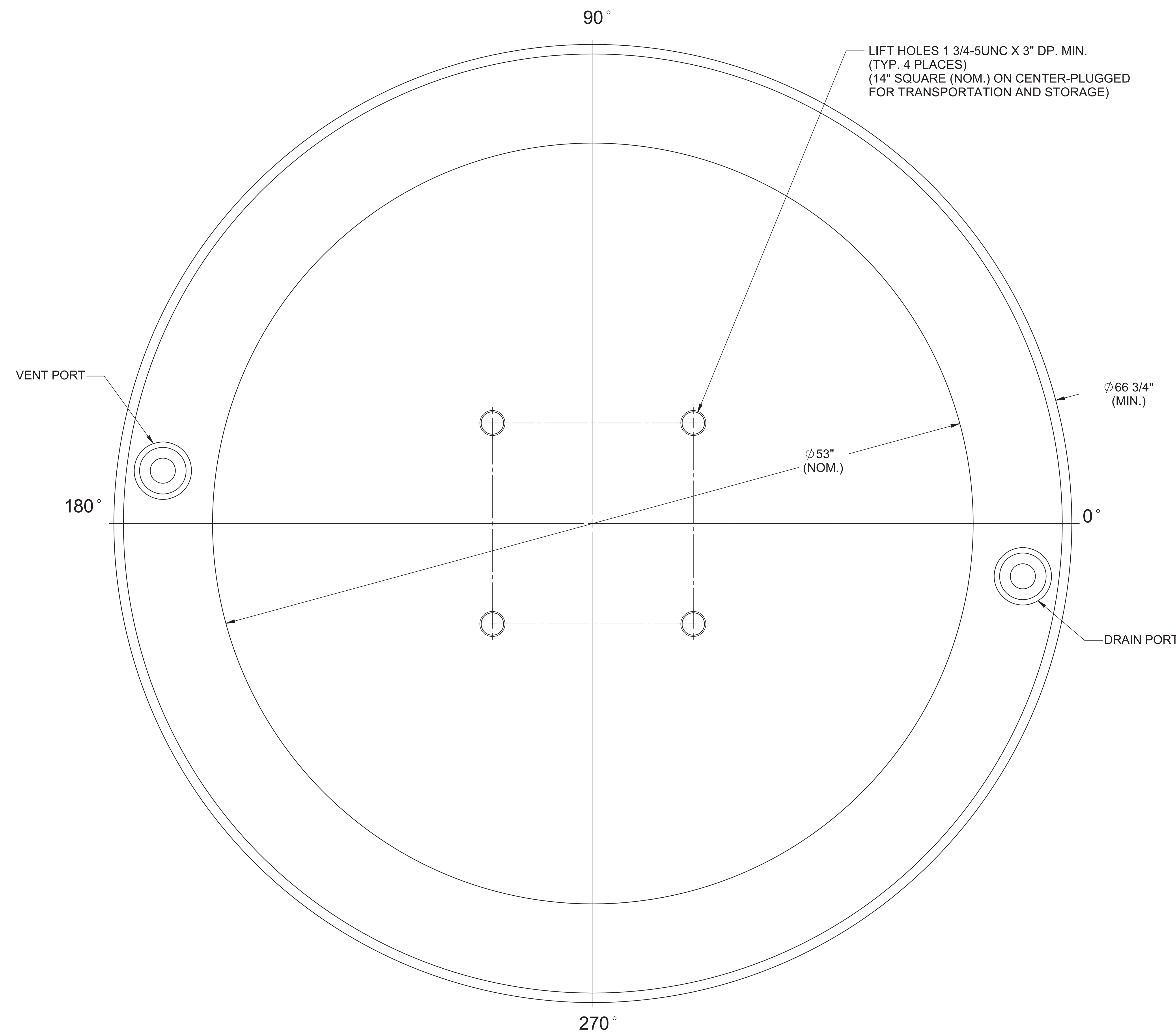
FSAR UPDATE

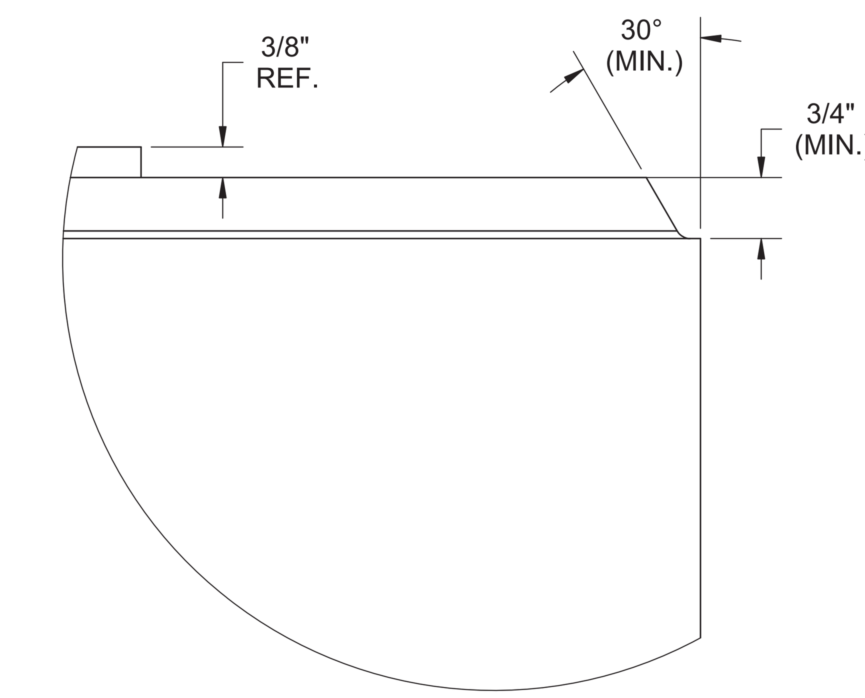
DIABLO CANYON ISFSI

FIGURE 4.2-13
MPC-32 ASSEMBLY
SHEET 3 OF 7

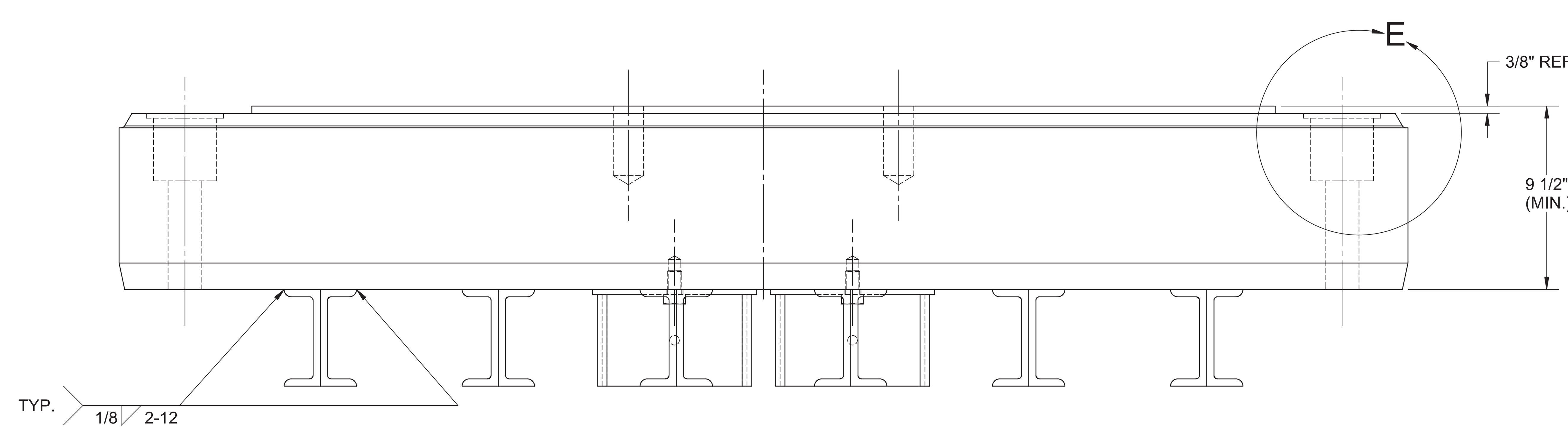
Revision 8 March 2020

HOLTEC INTERNATIONAL HOLTEC CENTER ONE HOLTEC DRIVE MARLTON, NJ 08053	CLIENT: PG&E
	DESCRIPTION: MPC ENCLOSURE ENCLOSURE VESSEL ELEVATION DETAILS
PROJECT NO. 1073 / 3100813919	COMPANION DRAWINGS: 4407,4408,4458
SIZE: D	DRAWING NO. 4459
SHEET: 3	REV: 15
SCALE: NONE	FILE PATH: G:\DRAWINGS\1073\4459





DETAIL E



MPC LID
(SEE NOTE 4, SHEET 6)

FSAR UPDATE

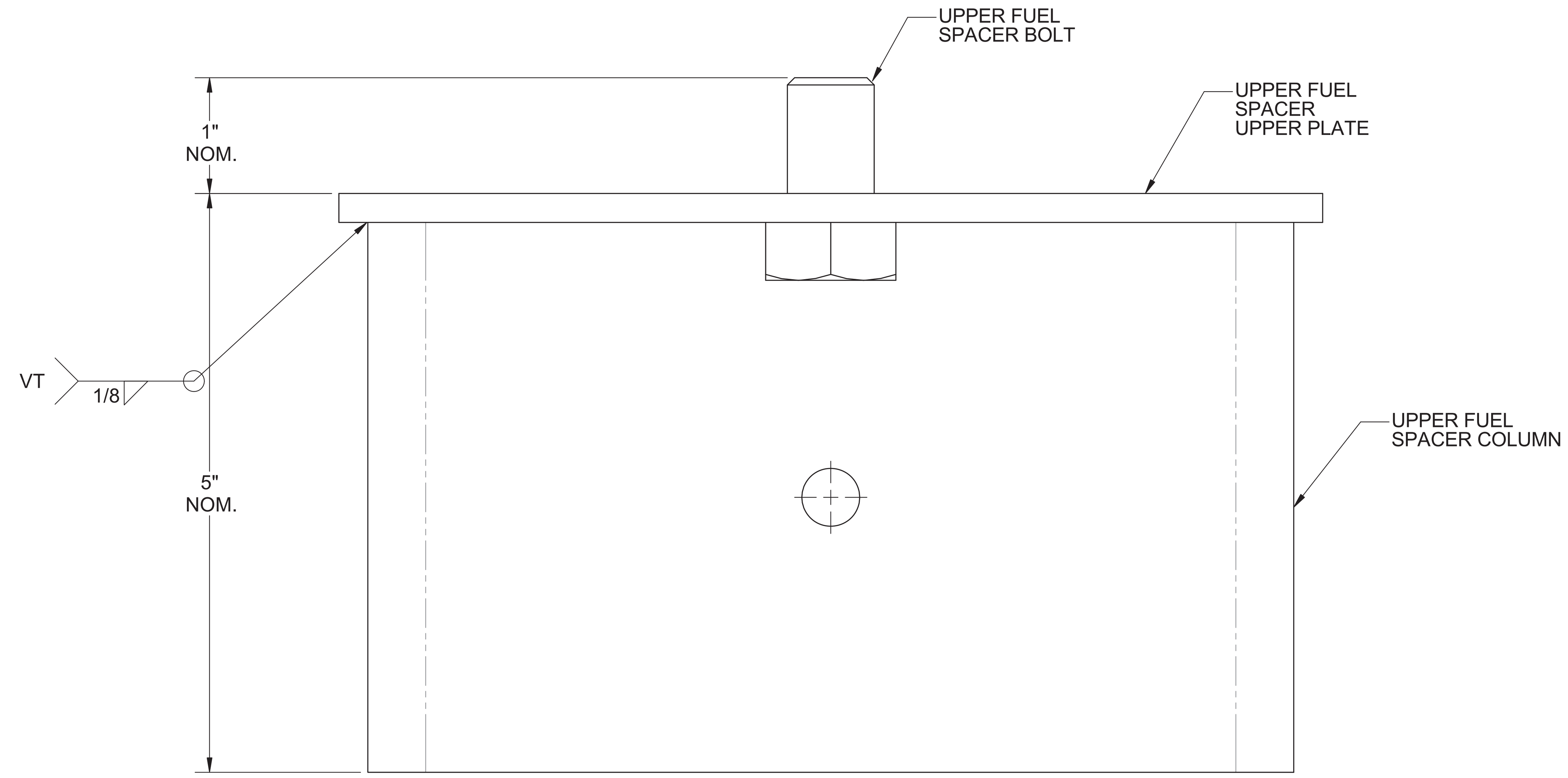
DIABLO CANYON ISFSI

FIGURE 4.2-13
MPC-32 ASSEMBLY
SHEET 4 OF 7

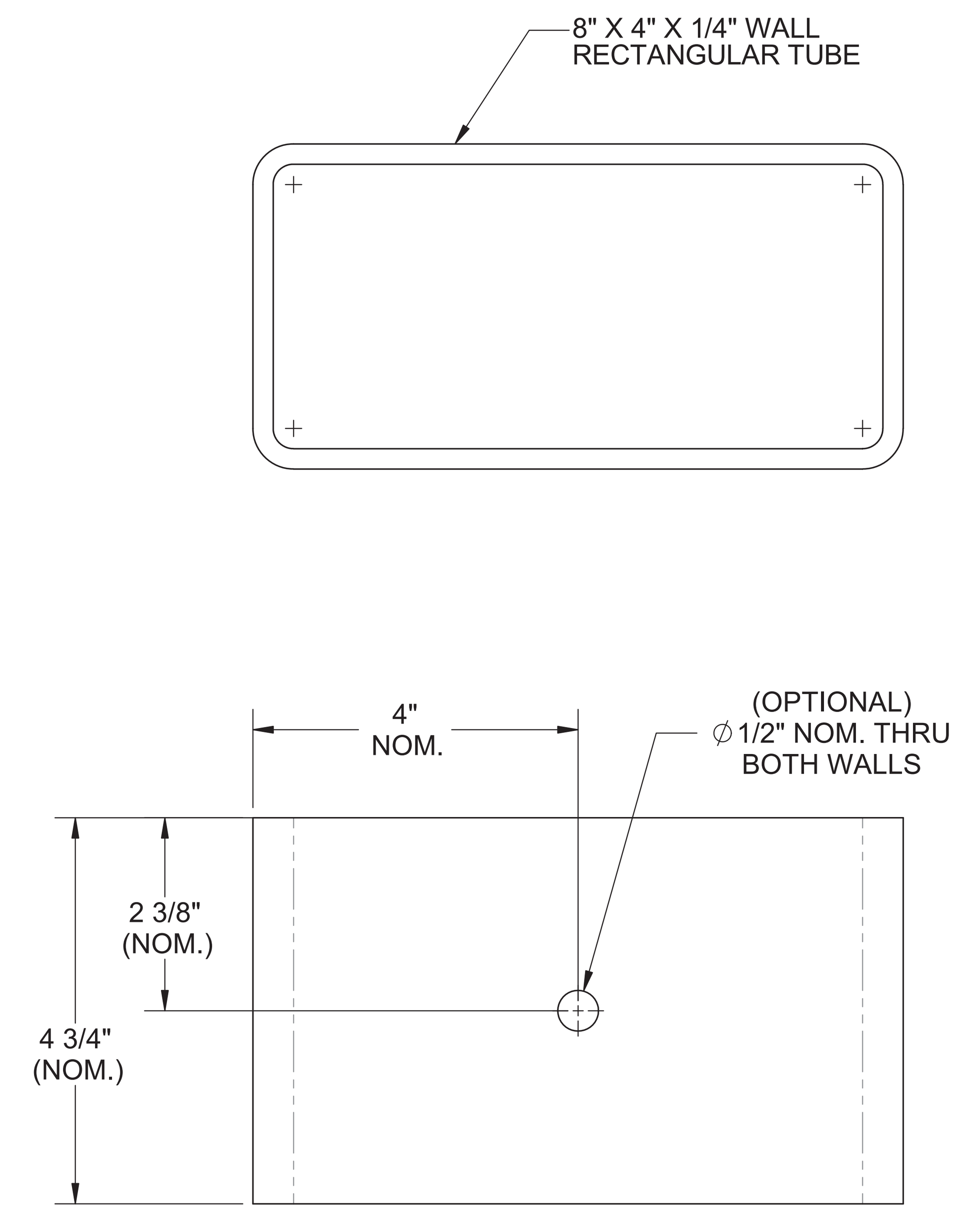
Revision 8 March 2020

HOLTEC INTERNATIONAL
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053

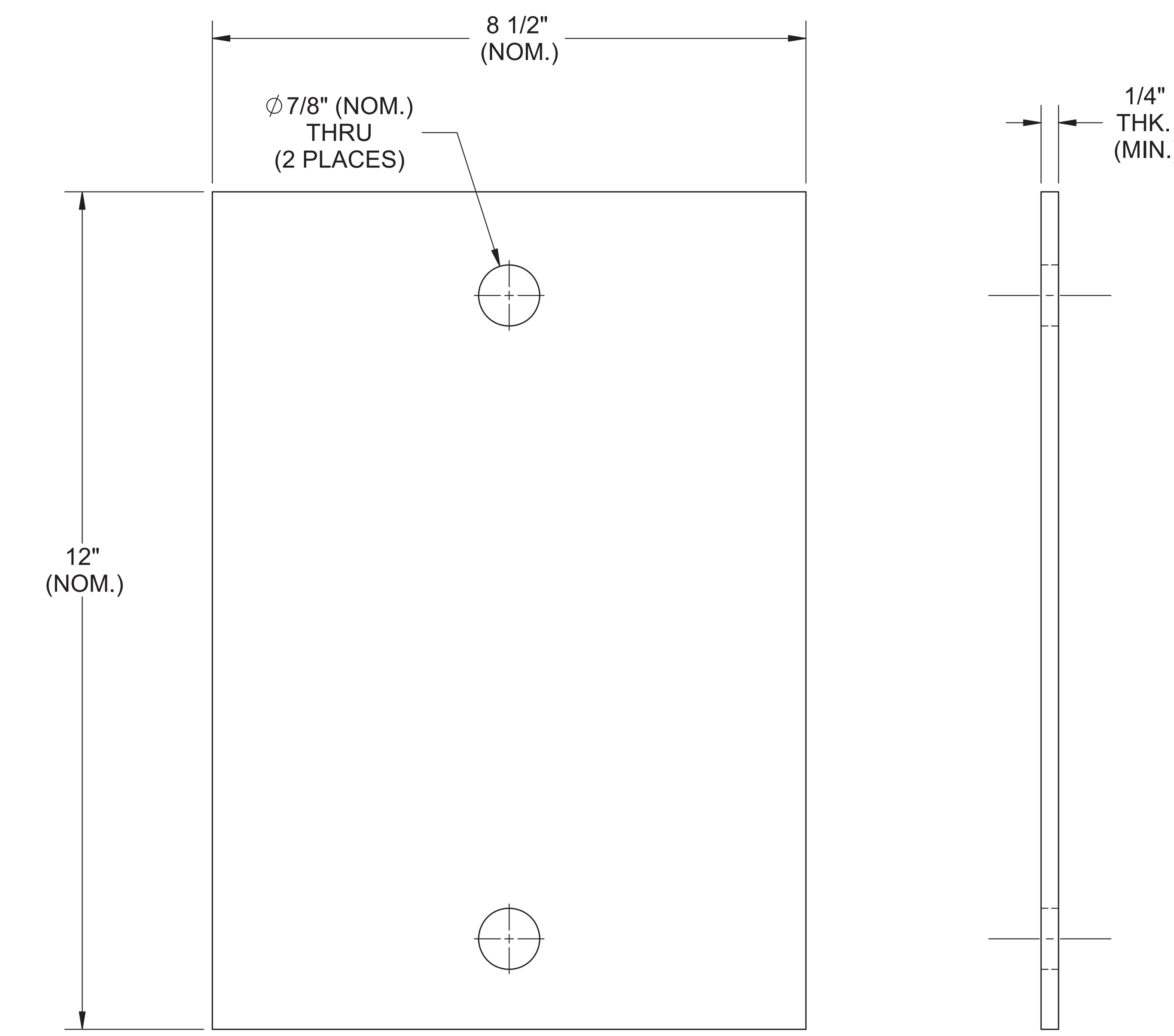
CLIENT	PG&E
DESCRIPTION	MPC ENCLOSURE VESSEL LID DETAILS
PROJECT NO.	1073
	3100813919
COMPANION DRAWINGS:	4407,4408,4458
SIZE	D
DRAWING NO.	4459
SHEET	4
REV	15
SCALE	NONE
FILE PATH:	G:\DRAWINGS\1073\4459



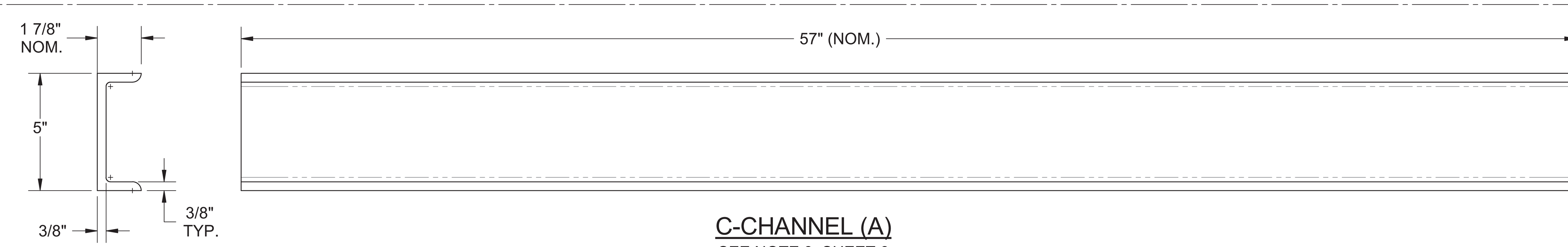
UPPER FUEL SPACER ASSEMBLY



UPPER FUEL SPACER COLUMN

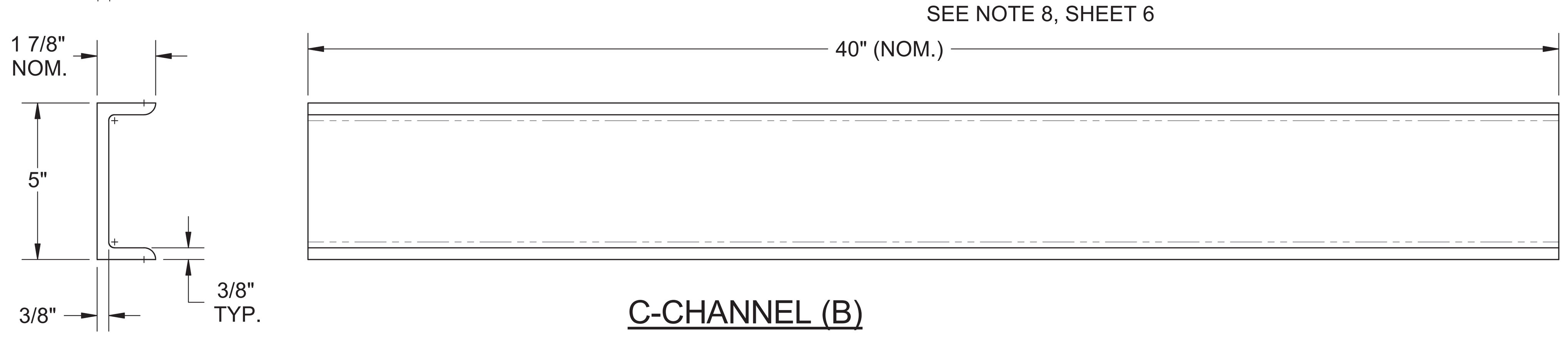


UPPER FUEL SPACER UPPER PLATE



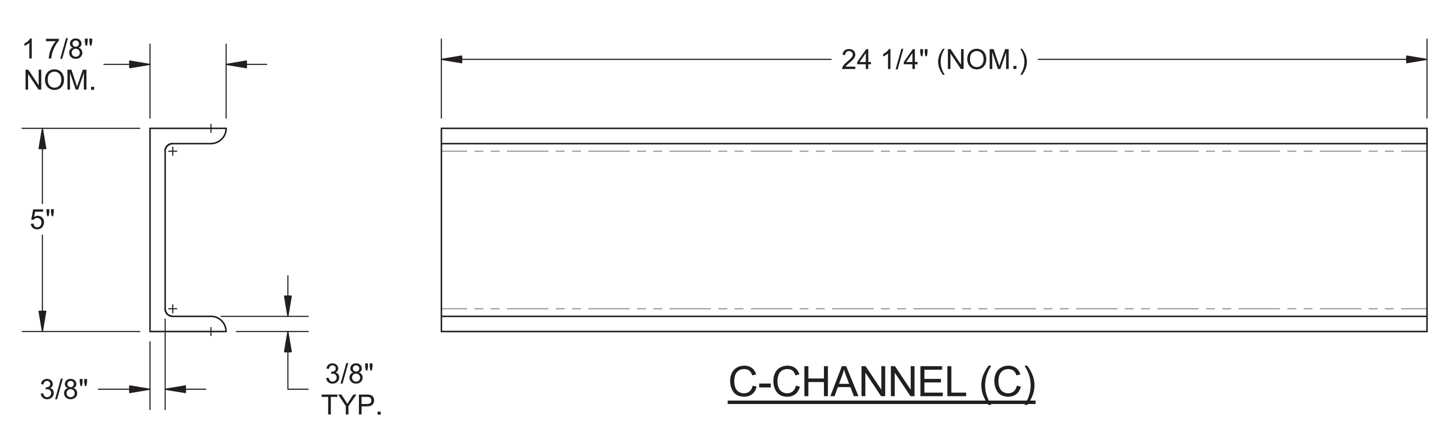
C-CHANNEL (A)

SEE NOTE 8, SHEET 6



C-CHANNEL (B)

SEE NOTE 8, SHEET 6



C-CHANNEL (C)

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-13
MPC-32 ASSEMBLY
SHEET 5 OF 7

Revision 8 March 2020

HOLTEC INTERNATIONAL
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053

CLIENT: PG&E

DESCRIPTION: FUEL SPACER DETAILS

PROJECT NO. 1073
3100813919

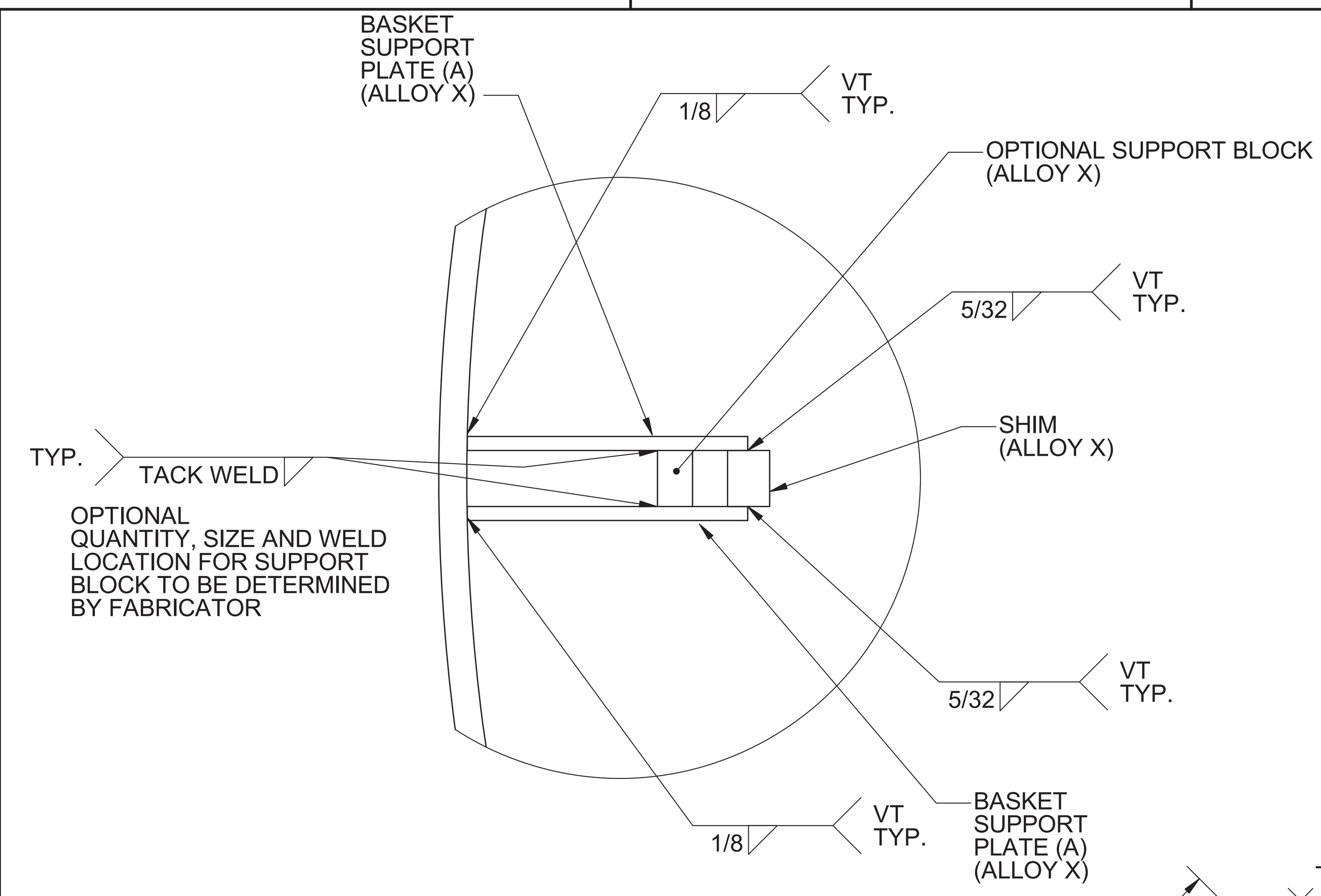
COMPANION DRAWINGS: 4407,4408,4458

SIZE: D | DRAWING NO. 4459 | SHEET 5 | REV 15

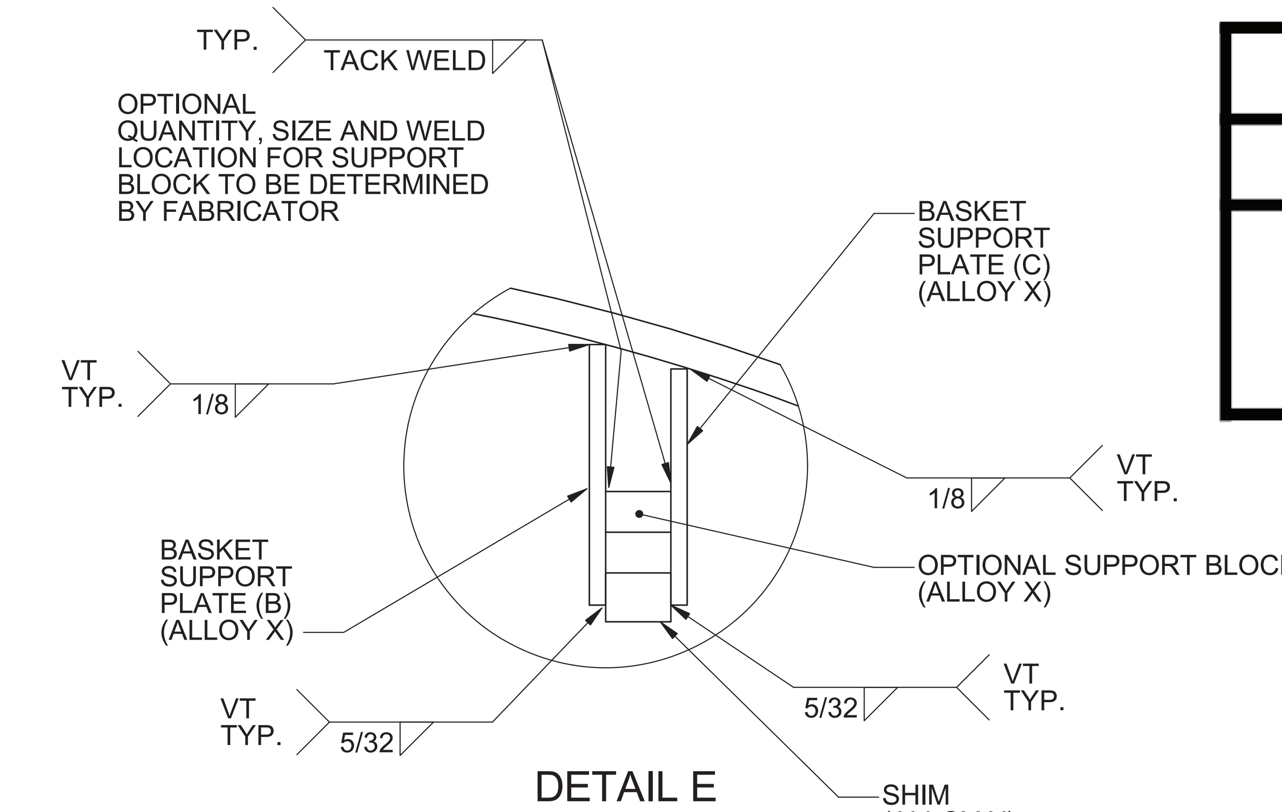
SCALE: NONE | FILE PATH: H:\DRAWINGS\1073\4459

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 4.2-13
MPC-32 ASSEMBLY
SHEET 6 OF 7

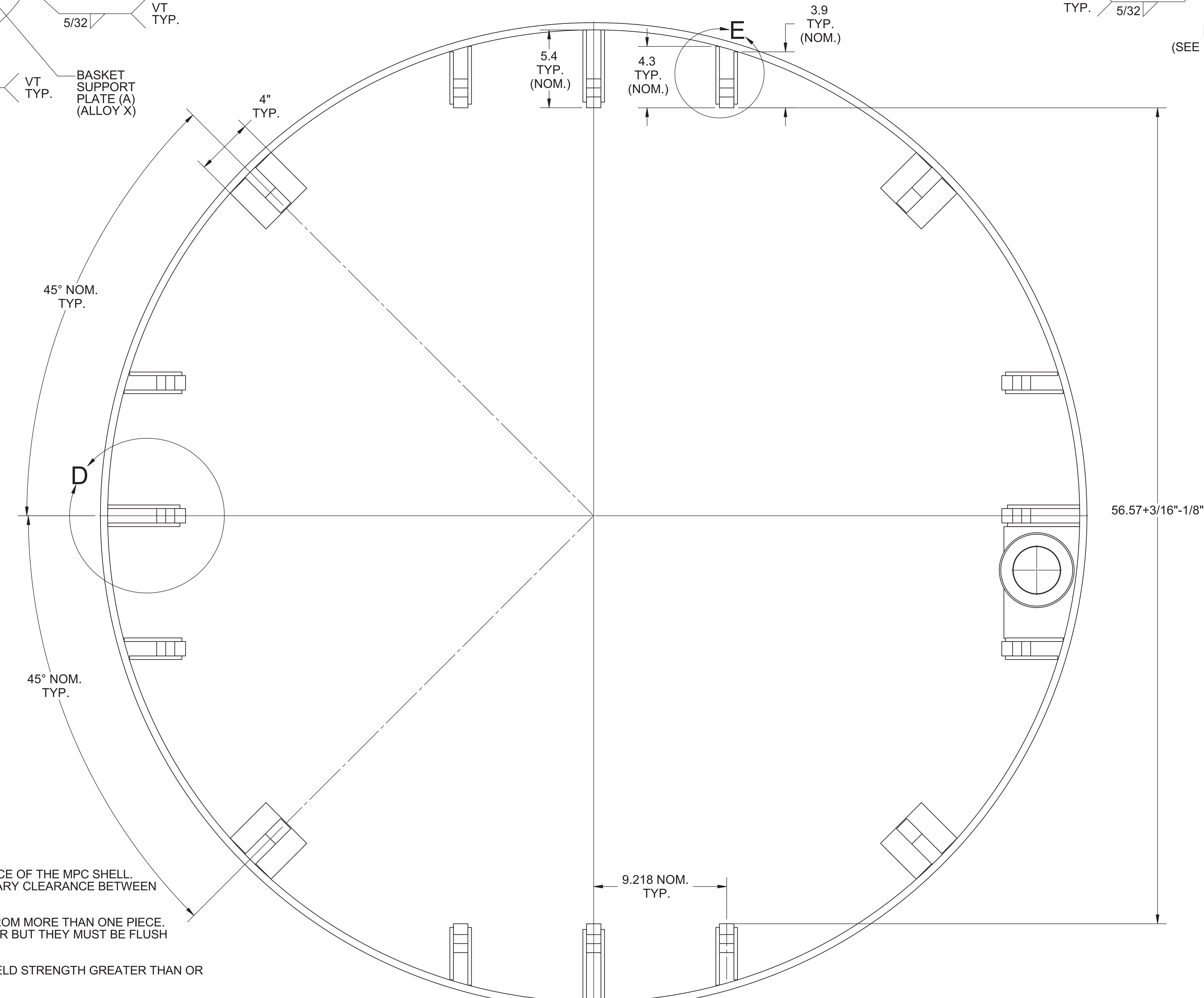
Revision 8 March 2020

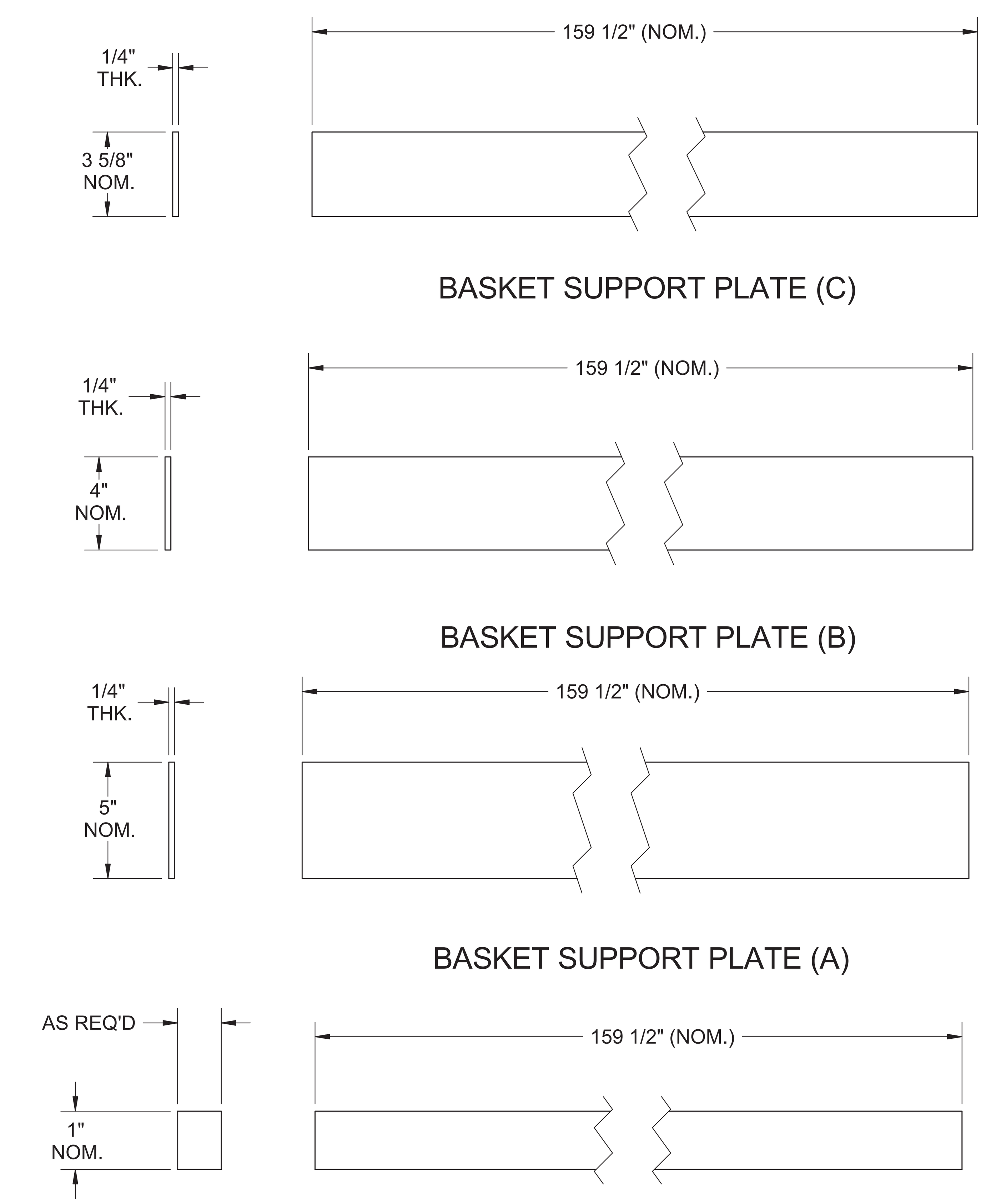


DETAIL D
(SEE NOTES 2 AND 3)



DETAIL E
(SEE NOTES 2 AND 3)

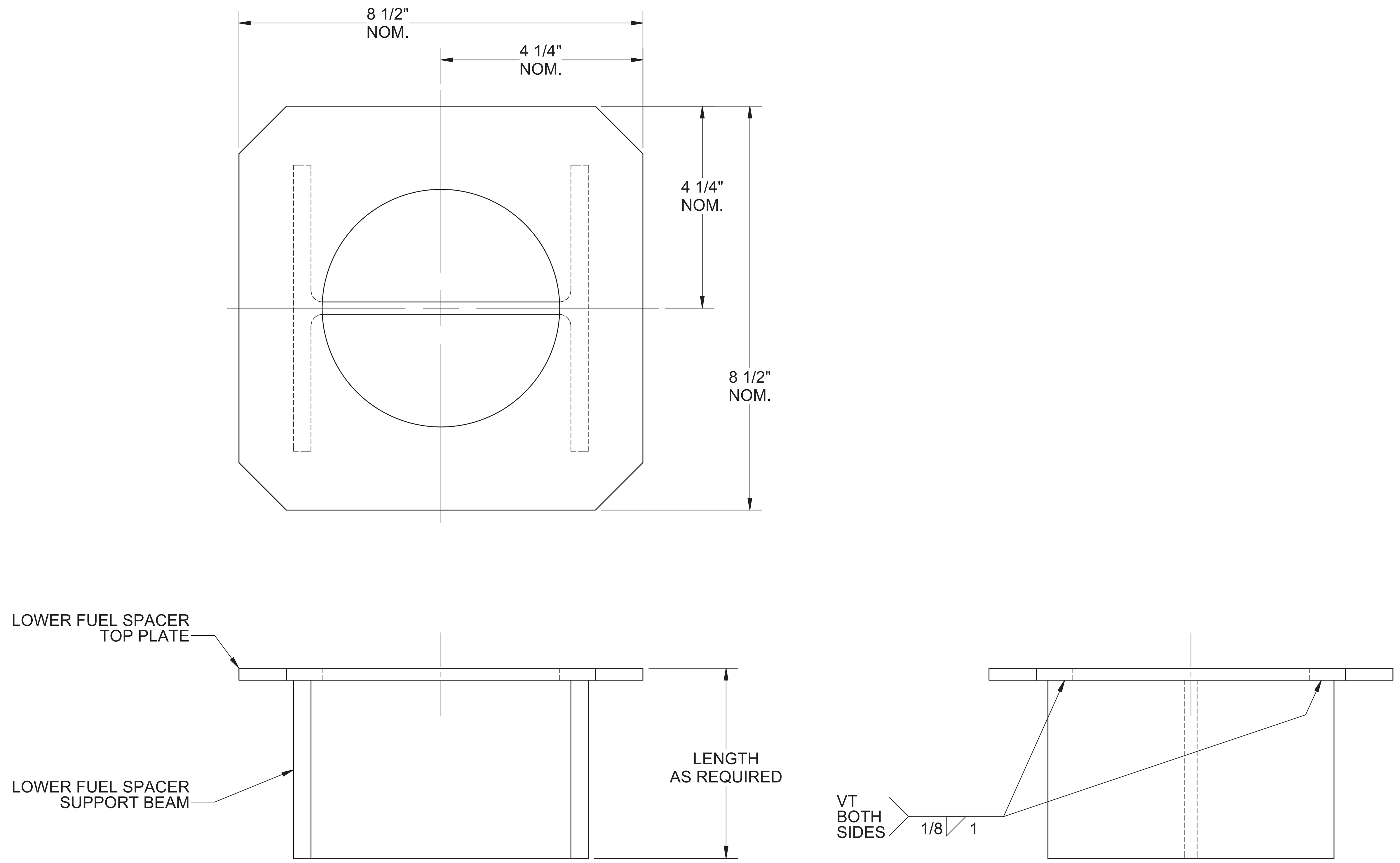




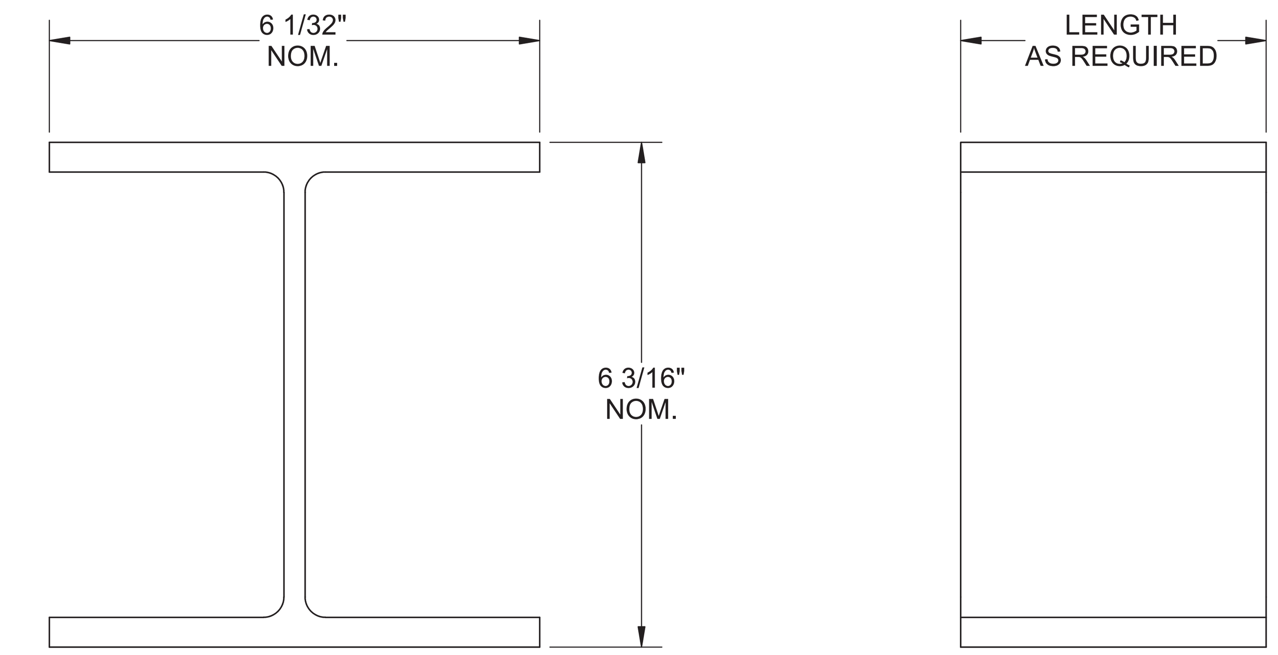
NOTES CONTINUED:

2. THE MPC BASKET SUPPORTS ARE WELDED TO THE INSIDE SURFACE OF THE MPC SHELL. THEY MAY INCLUDE ADJUSTABLE SHIMS TO ENSURE THE NECESSARY CLEARANCE BETWEEN THE BASKET TOUCH POINTS AND THE BASKET SUPPORTS.
3. BASKET SUPPORT PLATES AND SUPPORT SHIMS MAY BE MADE FROM MORE THAN ONE PIECE. THE ENDS OF THE PIECES DO NOT NEED TO BE WELDED TOGETHER BUT THEY MUST BE FLUSH WITH EACH OTHER WHEN INSTALLED.
4. THE ALLOY X MATERIAL USED FOR THE MPC LID SHALL HAVE A YIELD STRENGTH GREATER THAN OR EQUAL TO 33ksi AT ROOM TEMPERATURE.
5. HELIUM LEAK TESTING OF THE MPC SHELL WELDS SHALL BE PERFORMED TO THE LEAKTIGHT CRITERIA OF ANSI N14.5-1997. DIABLO CANYON MPC's S/N 8, 10, 12, 14, 17, 20, 21, 65, 120, 121, 122, 123, 124, 125, 126 AND 127 WERE LEAK TESTED TO 5X10^-6 STD-CC/SEC (HELIUM) PER ANSI N14.5-1997.
6. A MINIMUM GAP OF 0.08 INCHES IS REQUIRED BETWEEN THE BASKET CORNERS OF CELLS 1, 4, 5, 10, 23, 28, 29 AND 32 AND THE MPC SHELL. THIS MAY BE VERIFIED BY CONFIRMATION OF A DIAMETRICAL GAP OF 0.16 INCHES FOR OPPOSING CORNERS (E.G. CELLS 1 AND 32 OR CELLS 5 AND 28).
7. MPC SHELL OUTER DIAMETER SHALL NOT EXCEED 68 1/2" INCLUDING ANY WELD REINFORCEMENT. INFORMATION CHANGED TO ADDRESS CORRECTIVE ACTION(S) PER QI-1232. REFER TO QI-1232 PRIOR TO MAKING ANY CHANGES TO INFORMATION IDENTIFIED WITH THIS NOTE.
8. CHANNELS SHALL NOT EXTEND OVER FUEL STORAGE LOCATIONS USED TO ACCOMMODATE DAMAGED FUEL AND/OR FUEL DEBRIS. CHANNEL LENGTH SHALL BE ADJUSTED AS NECESSARY. DAMAGED FUEL AND/OR FUEL DEBRIS IS NOT CERTIFIED FOR TRANSPORTATION.
9. OPTIONAL CONSTRUCTION FOR THE CLOSURE RING PROVIDES PENETRATIONS TO ALLOW HELIUM LEAKAGE TESTING OF THE MPC LID-TO-SHELL AND VENT/DRAIN PORT COVER PLATE WELDS DURING A SINGLE TEST. STAINLESS STEEL SET SCREWS MAY BE USED IN ADDITION TO PLUG WELDS TO SEAL THE PENETRATIONS. SET SCREWS SHALL BE POSITIONED A MINIMUM OF 1/8" BELOW FLUSH.

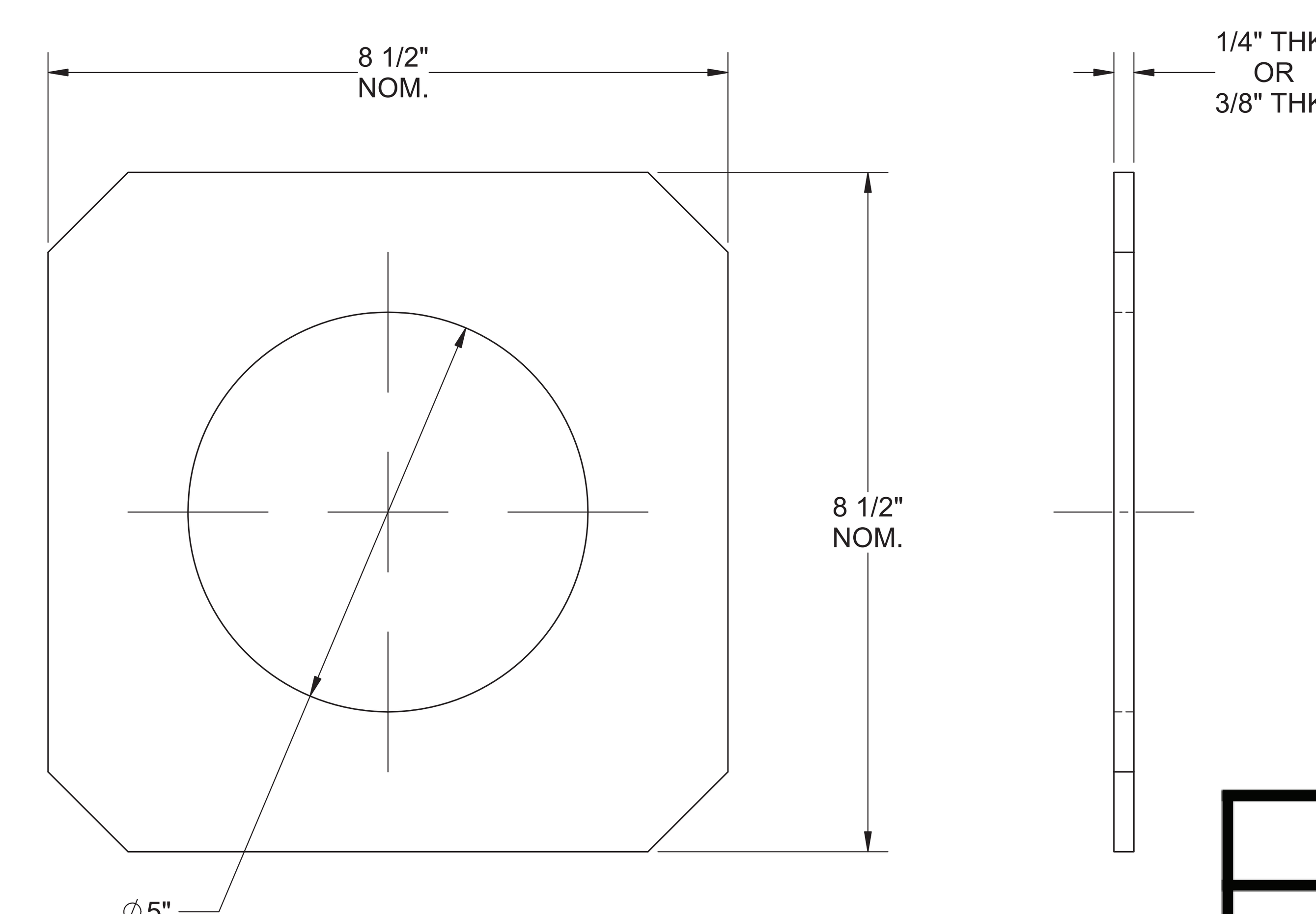
HOLTEC INTERNATIONAL HOLTEC CENTER ONE HOLTEC DRIVE MARLTON, NJ 08053	CLIENT: PG&E
	DESCRIPTION: DIABLO CANYON MPC-32 FUEL BASKET SUPPORTS
PROJECT NO.: 1073	COMPANION DRAWINGS: 4407,4408,4458
3100813919	SIZE: D · DRAWING NO.: 4459 · SHEET: 6 · REV: 15
	SCALE: NONE · FILE PATH: G:\DRAWINGS\1073\4459



OPTIONAL LOWER FUEL SPACER ASSEMBLY



LOWER FUEL SPACER SUPPORT BEAM



LOWER FUEL SPACER TOP PLATE

NOTES (CONTINUED):

10. OPTIONAL LOWER FUEL SPACERS MAY BE USED IN ANY OF THE FOUR INNER MOST FUEL BASKET CELL LOCATIONS. THE LOWER FUEL SPACER MAY NOT BE USED SIMULTANEOUSLY WITH THE UPPER FUEL SPACER IN THE SAME CELL LOCATION.

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-13
MPC-32 ASSEMBLY
SHEET 7 OF 7

Revision 8 March 2020

15

HOLTEC INTERNATIONAL
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053

CLIENT: PG&E

DESCRIPTION: OPTIONAL MPC LOWER FUEL SPACER DETAILS

PROJECT NO. 1073
3100813919

COMPANION DRAWINGS: 4407

SIZE D | DRAWING NO. 4459 | SHEET 7 | REV 15

SCALE NONE | FILE PATH H:\DRAWINGS\1073\4459

CLIENT

PG&E

LICENSING DRAWING PACKAGE COVER SHEET

PROJECT NO. 1073	P.O. NO. 3100813919
DRAWING PACKAGE I.D. 4458	TOTAL SHEETS 3

LICENSING DRAWING PACKAGE CONTENTS:

SHEET	DESCRIPTION
1	COVER SHEET
2	FUEL BASKET ARRANGEMENT
3	FUEL BASKET LAYOUT AND WELD DETAILS

REVISION LOG

IT IS MANDATORY AT EACH REVISION TO COMPLETE THE REVIEW & APPROVAL LOG STORED IN HOLTEC'S DIRECTORY N:\PDOXWIN\WORKING\DBAL BY ALL RELEVANT TECHNICAL DISCIPLINES, PM AND QA PERSONNEL. EACH ATTACHED DRAWING SHEET CONTAINS ANNOTATED TRIANGLES INDICATING THE REVISION TO THE DRAWING.

REV	AFFECTED DRAWING SHEET NUMBERS	SUMMARY OF CHANGES/ AFFECTED ECOs	PREPARED BY	APPROVAL DATE	VIR# †
9	SHEET 3	1073-107, REV. 0	DCB	10/13/11	19935
10	ALL	1073-116, REV. 0	DPS	6/11/2015	90807
11	1 AND 3	1073-122, REV. 0	SMS	SEE BELOW	SEE BELOW

† THE VALIDATION IDENTIFICATION RECORD (VIR) NUMBER IS A COMPUTER GENERATED RANDOM NUMBER WHICH CONFIRMS THAT ALL APPROPRIATE REVIEWS OF THIS DRAWING ARE DOCUMENTED IN COMPANY'S NETWORK.

GENERAL NOTES CONTINUED:

14. NEUTRON ABSORBER PANELS MAY HAVE A REDUCTION IN WIDTH OF UP TO 1/32" OVER A LENGTH OF NO MORE THAN 12" PROVIDED THE AVERAGE WIDTH OF THE PANEL IS NO LESS THAN THE MINIMUM SPECIFIED.

15. NOT USED.

16. DIMENSIONS NOTED AS NOMINAL ("NOM.") IN THE DRAWING ARE FOR INFORMATION ONLY, IN ORDER TO INDICATE THE GENERAL SIZE OF THE COMPONENT OR PART.

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-14
MPC-32 FUEL BASKET ASSEMBLY
SHEET 1 OF 3

Revision 8 March 2020

GENERAL NOTES:

1. DELETED.

2. DIMENSIONAL TOLERANCES ON THIS DRAWING ARE PROVIDED TO ENSURE THAT THE EQUIPMENT DESIGN IS CONSISTENT WITH THE SUPPORTING ANALYSIS. HARDWARE IS FABRICATED IN ACCORDANCE WITH THE DESIGN DRAWINGS, WHICH MAY HAVE MORE RESTRICTIVE TOLERANCES, TO ENSURE COMPONENT FIT-UP. DO NOT USE WORST-CASE TOLERANCE STACK-UP FROM THIS DRAWING TO DETERMINE COMPONENT FIT-UP.

3. THE REVISION LEVEL OF EACH INDIVIDUAL SHEET IN THIS PACKAGE IS THE SAME AS THE REVISION LEVEL OF THIS COVER SHEET. A REVISION TO ANY SHEET(S) IN THIS PACKAGE REQUIRES UPDATING OF REVISION NUMBERS OF ALL SHEETS TO THE NEXT REVISION NUMBER.

4. THE ASME BOILER AND PRESSURE VESSEL CODE (ASME CODE), 1995 EDITION WITH ADDENDA THROUGH 1997, IS THE GOVERNING CODE, WITH CERTAIN APPROVED ALTERNATIVES AS LISTED IN SAR TABLE1.3.2 (HI-STAR 100 TRANSPORTATION) AND DC ISFSI FSAR TABLE 3.4-6. THE MPC FUEL BASKET IS CONSTRUCTED IN ACCORDANCE WITH ASME SECTION III, SUBSECTION NG AS DESCRIBED IN THE SAR AND FSAR. NEW OR REVISED ASME CODE ALTERNATIVES REQUIRE PRIOR NRC APPROVAL BEFORE IMPLEMENTATION.

5. ALL MPC BASKET STRUCTURAL MATERIALS COMPLY WITH THE REQUIREMENTS OF ASME SECTION II, PART A. WELD MATERIAL COMPLIES WITH THE REQUIREMENTS OF ASME SECTION II, PART C.

6. ALL WELDS REQUIRE VISUAL EXAMINATION (VT). ADDITIONAL NDE INSPECTIONS ARE NOTED ON THE DRAWING AS REQUIRED. NDE TECHNIQUES AND ACCEPTANCE CRITERIA ARE PROVIDED IN THE APPLICABLE CODES AS CLARIFIED IN THE APPLICABLE HOLTEC SAFETY ANALYSIS REPORT.

7. FABRICATOR MAY ADD WELDS TO STITCH WELDS AT THEIR DISCRETION.

8. DO NOT MAKE A CONTINUOUS SEAL WELD BETWEEN THE SHEATHING AND THE CELL WALL.

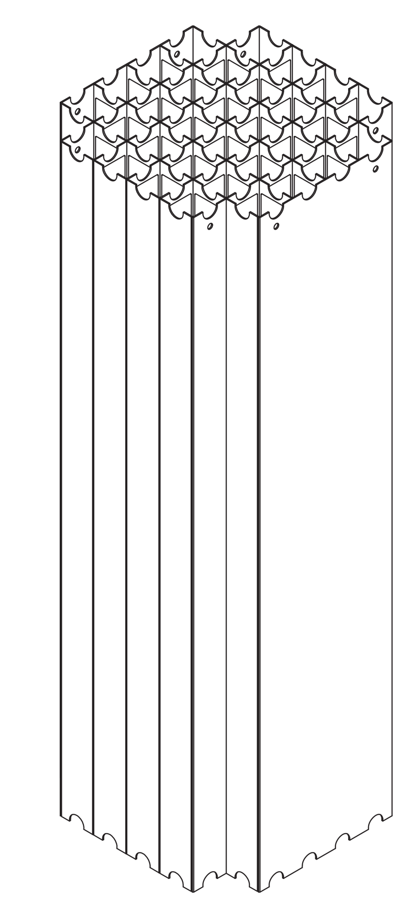
9. ALL STRUCTURAL MATERIALS ARE "ALLOY X" UNLESS OTHERWISE NOTED. ALLOY X IS ANY OF THE FOLLOWING STAINLESS STEEL TYPES: 316, 316 LN, 304, AND 304 LN.

10. BORAL PANELS MAY BE SUBSTITUTED BY METAMIC PANELS AND ARE INTENDED TO HAVE NO SIGNIFICANT FLAWS. HOWEVER, TO ACCOUNT FOR MANUFACTURING DEVIATIONS OCCURING DURING INSTALLATION OF THE PANELS INTO THE MPC FUEL BASKET, BORAL OR METAMIC DAMAGE OF UP TO THE EQUIVALENT OF A 1" DIAMETER HOLE IN EACH PANEL HAS BEEN ANALYZED AND FOUND TO BE ACCEPTABLE.

11. THIS COMPONENT IS CLASSIFIED AS ITS-A BASED ON THE HIGHEST CLASSIFICATION OF ANY SUBCOMPONENT. SUBCOMPONENT CLASSIFICATIONS ARE PROVIDED IN SAR TABLE 1.3.3 (TRANSPORTATION) AND ARE DISCUSSED IN DC ISFSI FSAR SECTION 4.5 (STORAGE).

12. TOLERANCES FOR THICKNESS OF ASME CODE MATERIAL ARE SPECIFIED IN ASME SECTION II.

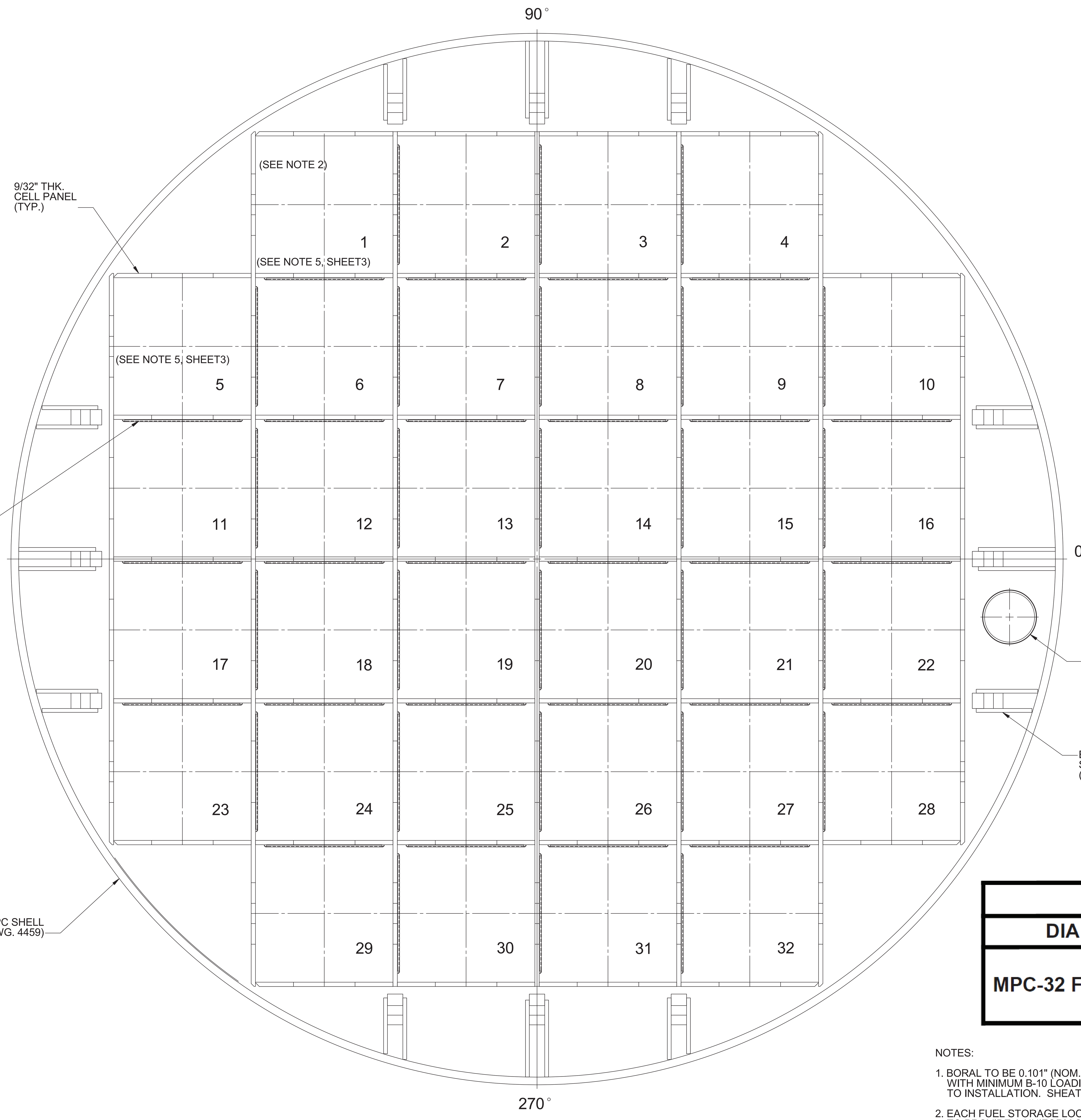
13. ALL WELD SIZES ARE MINIMUMS. LARGER WELDS ARE PERMITTED. LOCAL AREAS OF UNDERSIZE WELDS ARE ACCEPTABLE WITHIN THE LIMITS SPECIFIED IN THE ASME CODE, AS CLARIFIED IN THE DC ISFSI FSAR AND THE HI-STAR 100 TRANSPORT SAR.



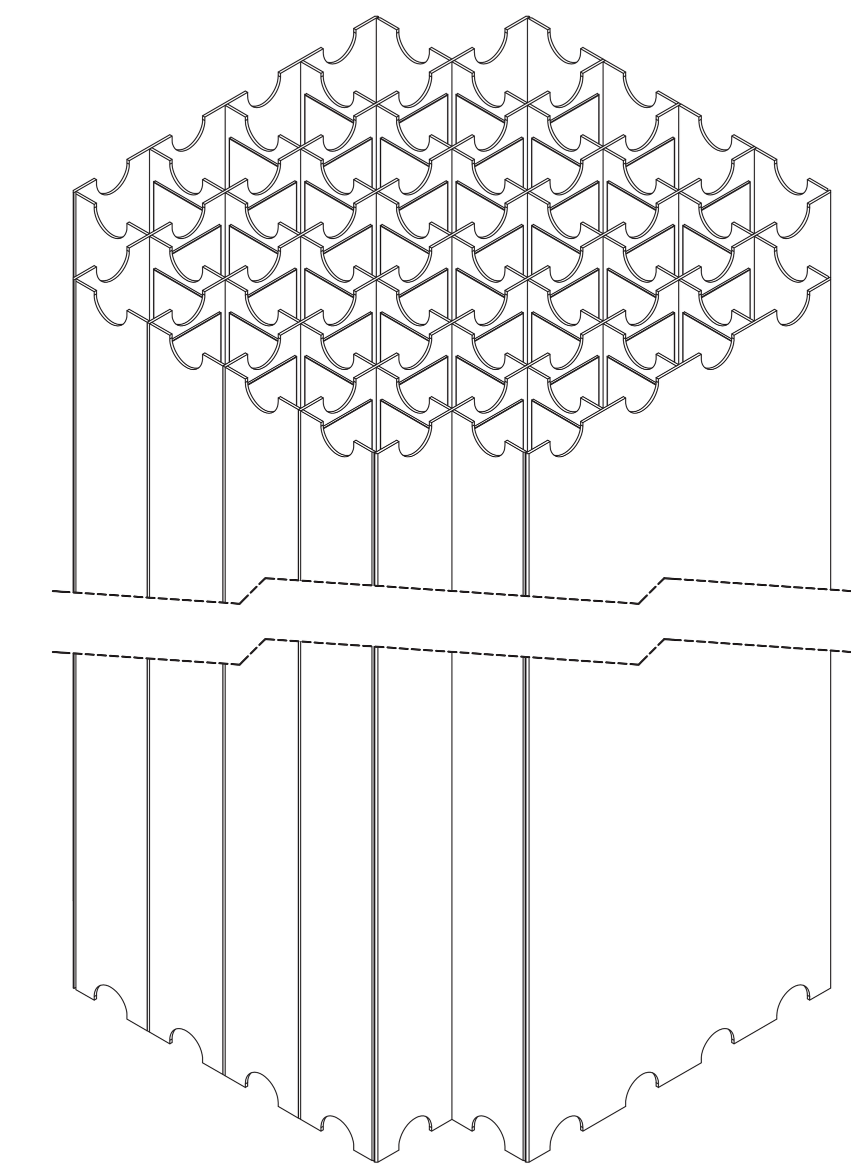
MPC-32 FUEL BASKET
ISOMETRIC VIEW

HOLTEC INTERNATIONAL
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053

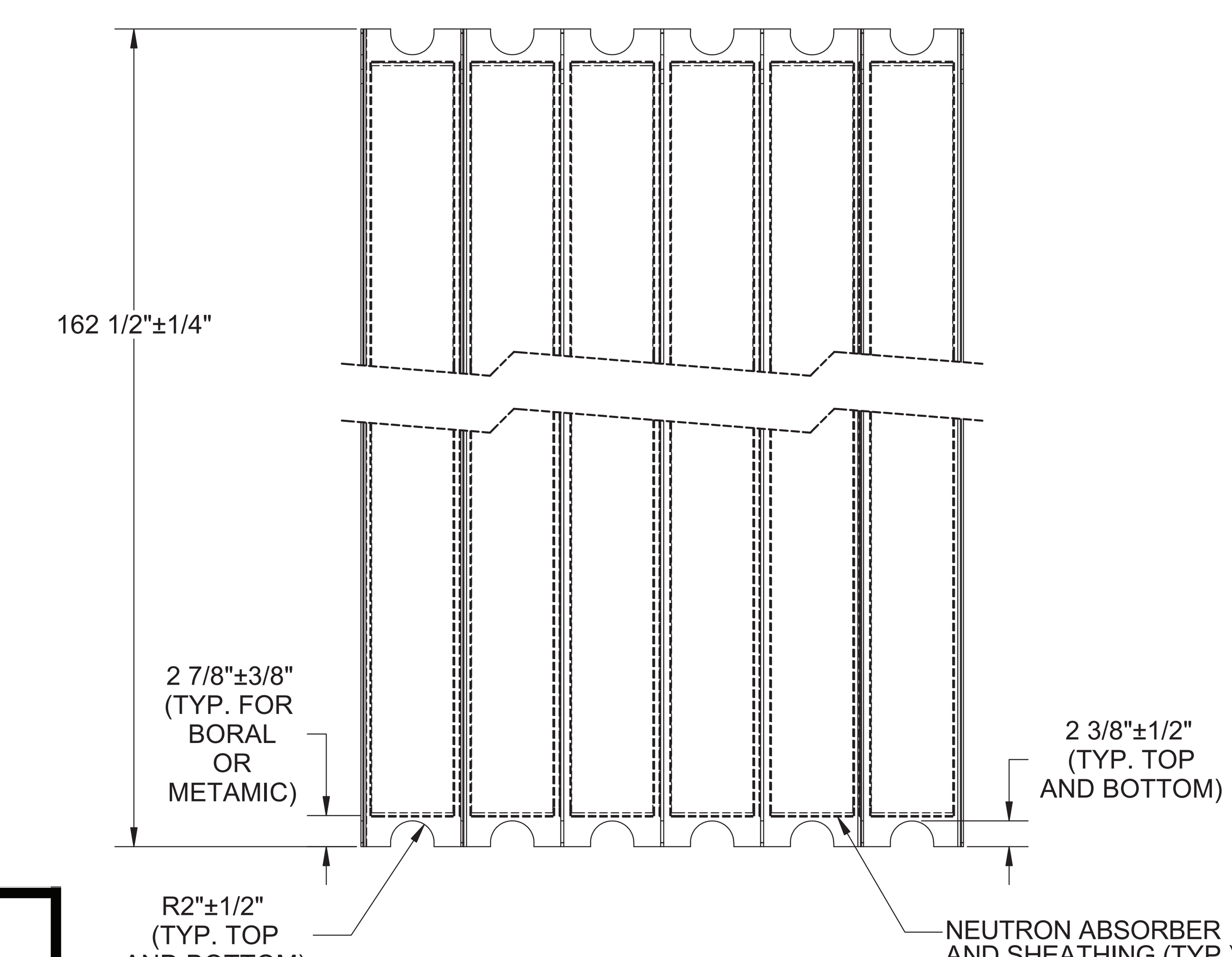
CLIENT	PG&E
DESCRIPTION	DIABLO CANYON MPC-32 FUEL BASKET ASSEMBLY
PROJECT NO.	1073
P.O. NO.	3100813919
DRAWING NO.	4458
SHEET	1
TOTAL SHEETS	3
FILE PATH:	N:\DRAWINGS\1073\4458



CROSS SECTIONAL VIEW OF THE MPC-32



ISOMETRIC ASSEMBLY VIEW



BASKET ELEVATION VIEW

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-14
MPC-32 FUEL BASKET ASSEMBLY
SHEET 2 OF 3

Revision 8 March 2020

NOTES:

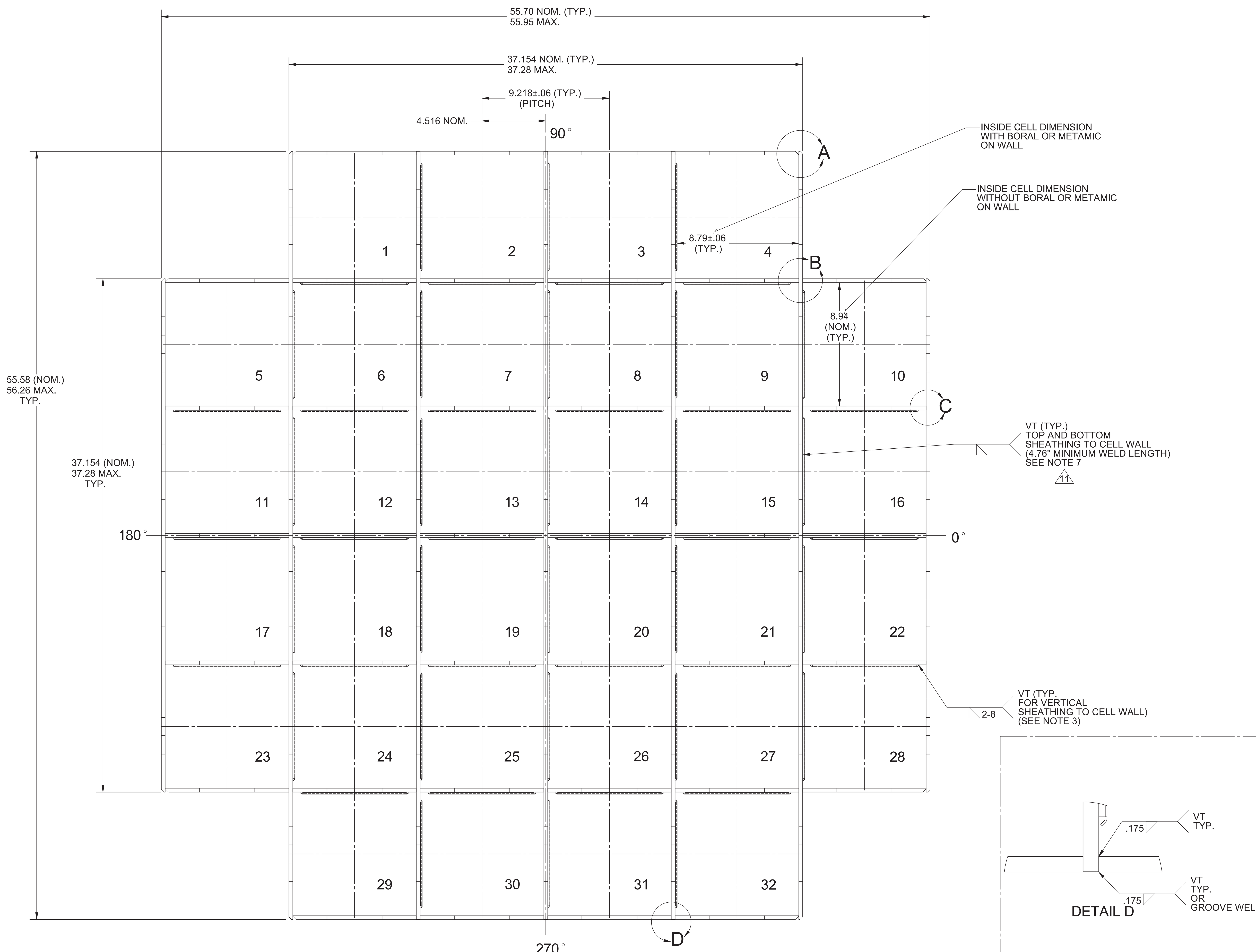
1. BORAL TO BE 0.101" (NOM.)THK. X 7 1/2" WIDE MIN. X 155 7/8" MIN., (156" NOM.) LG. WITH MINIMUM B-10 LOADING OF 0.0372 g/cm². BORAL TO BE PASSIVATED PRIOR TO INSTALLATION. SHEATHING 0.035" THK.

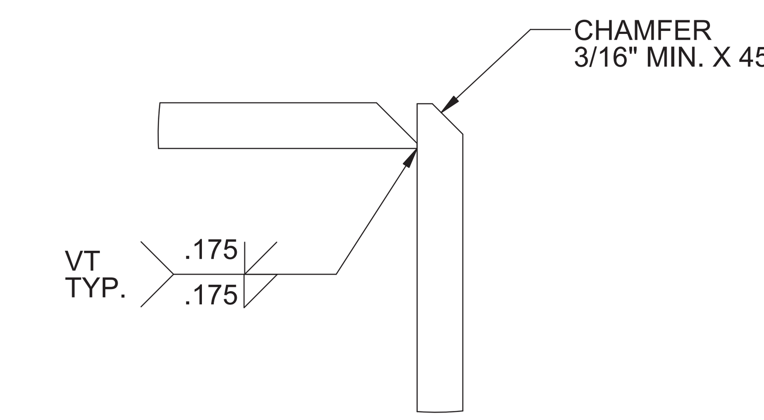
2. EACH FUEL STORAGE LOCATION IS GIVEN A UNIQUE IDENTIFICATION NUMBER TO PROVIDE A UNIFORM SYSTEM FOR RECORDING THE LOCATION OF EACH FUEL ASSEMBLY.

CONTINUED ON SHEET 3.

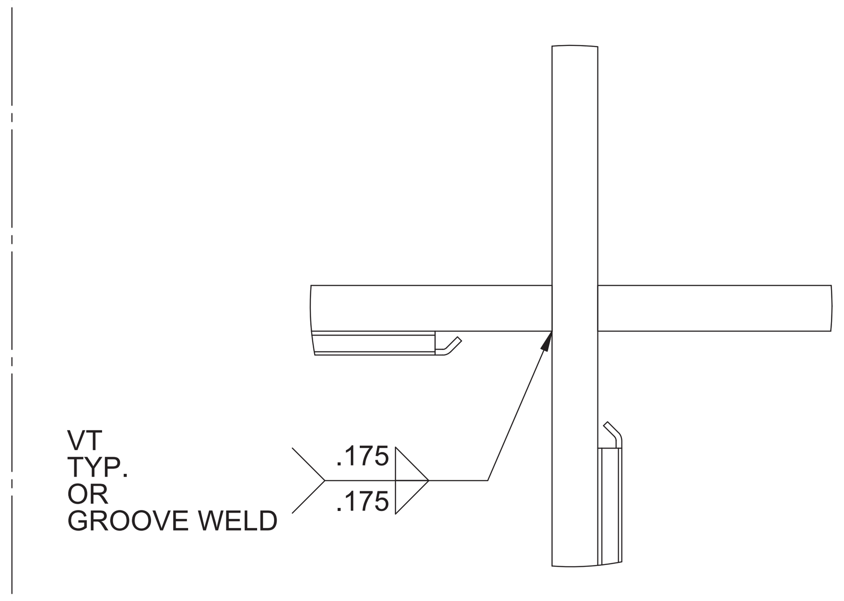
HOLTEC INTERNATIONAL
HOLTEC CENTER
ONE HOLTEC DRIVE
MARLTON, NJ 08053

CLIENT	PG&E
DESCRIPTION	DIABLO CANYON FUEL BASKET ARRANGEMENT
PROJECT NO.	1073 / 3100813919
COMPANION DRAWINGS:	4407,4408,4459
SIZE	D
DRAWING NO.	4458
SHEET	2
REV	11
SCALE	NONE
FILE PATH:	N:\DRAWINGS\1073\4458

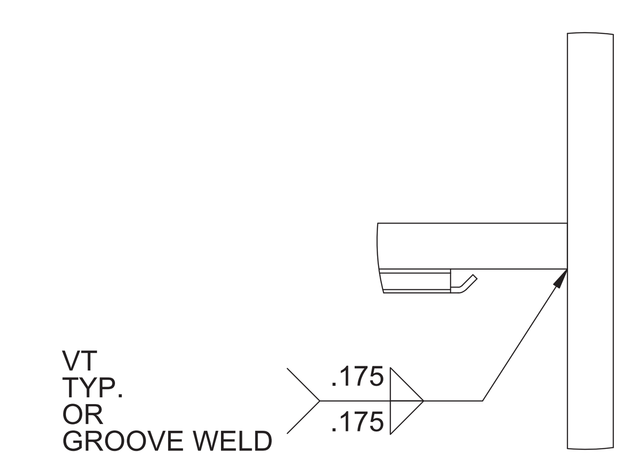




DETAIL A
SEE NOTE 6



DETAIL B



DETAIL C

DETAIL D

NOTES CONTINUED:

3. LENGTH AND SPACING OF WELD ON SIDES OF SHEATHING CAN VARY PROVIDED TOTAL VERTICAL SHEATHING WELD PER PIECE IS NOT LESS THAN 70" AND THERE IS A MINIMUM OF 19" OF WELD ALONG EACH SIDE OF THE SHEATHING PANEL.

4. METAMIC TO BE .106" (NOM.) THK. X 7 1/2" WIDE MIN. X 155 7/8" MIN. (156" NOM.) LG. THE MINIMUM METAMIC B-10 LOADING IS 0.0310 g/cm², WITH A MINIMUM B4C LOADING OF 31.5% AND MAXIMUM OF 33% METAMIC IS NOT REQUIRED TO BE PASSIVATED. SHEATHING 0.035" THK.

11 5. ONLY FUEL STORAGE LOCATIONS 1 AND 5 CAN ACCOMMODATE DAMAGED FUEL AND/OR FUEL DEBRIS. DAMAGED FUEL AND/OR FUEL DEBRIS IS NOT CERTIFIED FOR TRANSPORTATION.

6. CHAMFER CONFIGURATION MAY VARY PROVIDED THE RADIAL BASKET TO MPC SHELL GAP MEETS THE REQUIREMENTS OF DRAWING 4459 NOTE 6 (SHEET 6).

11 7. MINIMUM WELD LENGTH VERIFIED BY EVALUATION OF TOLERANCE STACK-UP, NOT DIRECT INSPECTION, FOR MPC-32 SERIAL NUMBERS 8, 10, 12, 14,17, 20, 21, AND 65. (SMDR-1073-1709).

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.2-14
MPC-32 FUEL BASKET ASSEMBLY
SHEET 3 OF 3

Revision 8 March 2020

HOLTEC INTERNATIONAL HOLTEC CENTER ONE HOLTEC DRIVE MARLTON, NJ 08053	CLIENT: PG&E
	DESCRIPTION: DIABLO CANYON MPC-32 FUEL BASKET LAYOUT AND WELD DETAILS
PROJECT NO.: 1073	COMPANION DRAWINGS: 4407,4408,4459
3100813919	
SIZE: D	DRAWING NO.: 4458
SHEET: 3	REV: 11
SCALE: NONE	FILE PATH: N:\DRAWINGS\1073\4458

LICENSING DRAWING PACKAGE COVER SHEET

CLIENT	PG&E		
PROJECT NO.	1073	P.O. NO.	3100813919
DRAWING PACKAGE I.D.	4480	TOTAL SHEETS	6

LICENSING DRAWING PACKAGE CONTENTS:

SHEET	DESCRIPTION
1	COVER SHEET
2	ASSEMBLY AND NOTES
3	DETAILS
4	WEDGE DETAILS
5	WEDGE ASSEMBLY
6	SECTION VIEW

REVISION LOG

IT IS MANDATORY AT EACH REVISION TO COMPLETE THE REVIEW & APPROVAL LOG STORED IN HOLTEC'S DIRECTORY N:\PDOXWIN\WORKING\DBAL BY ALL RELEVANT TECHNICAL DISCIPLINES. PM AND QA PERSONNEL. EACH ATTACHED DRAWING SHEET CONTAINS ANNOTATED TRIANGLES INDICATING THE REVISION TO THE DRAWING.

REV	AFFECTED DRAWING SHEET NUMBERS	SUMMARY OF CHANGES/ AFFECTED ECOs	PREPARED BY	APPROVAL DATE	VIR# †
0	INITIAL ISSUE	-	JJB	03/23/05	87442
1	ALL SHEETS	ECO 1073-19 REV. 0	JJB	09/16/05	63822
2	ALL SHEETS	ECO 1073-28 REV. 0	SLC	10/21/05	80489
3	SHEETS 4 & 5	ECO 1073-34 REV. 0	DCB	01/12/06	37025
4	SHEET 2	ECO 1073-44 REV. 0	SLC	03/01/06	46724
5	SHEET 2	ECO 1073-48 REV. 0	SLC	03/21/06	21640
6	ALL SHEETS	ECO 1073-62 REV. 0	JJB	09/13/06	16247

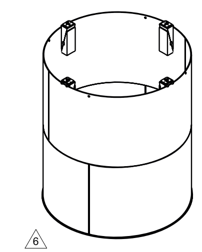
† THE VALIDATION IDENTIFICATION RECORD (VIR) NUMBER IS A COMPUTER GENERATED RANDOM NUMBER WHICH CONFIRMS THAT ALL APPROPRIATE REVIEWS OF THIS DRAWING ARE DOCUMENTED IN COMPANY'S NETWORK.

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 4.4-3
1 of 6
Cask Transfer Facility Assembly

Revision 2 June 2008

NOTES:

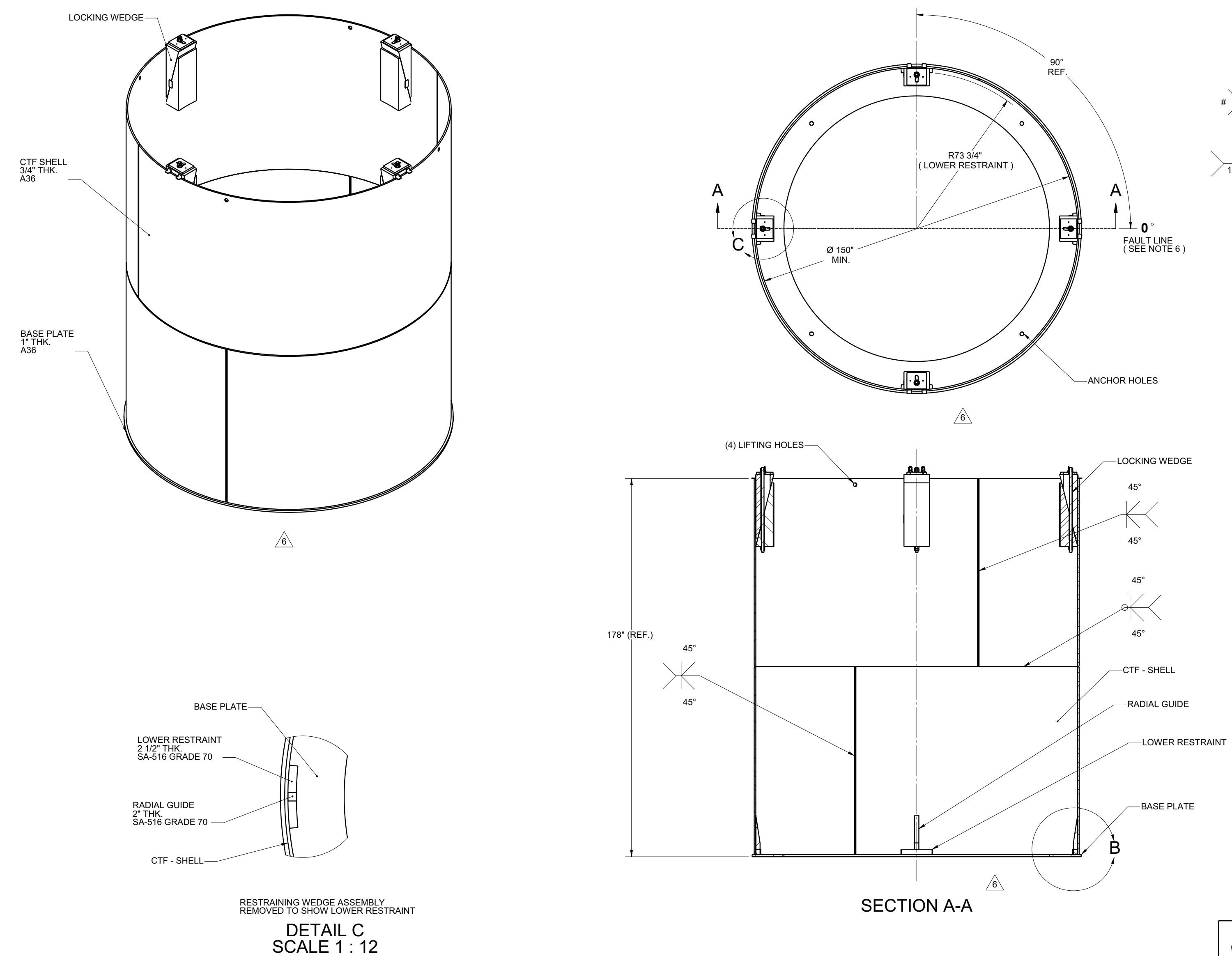
1. THE EQUIPMENT DOCUMENTED IN THIS DRAWING PACKAGE HAS BEEN CONFIRMED BY HOLTEC INTERNATIONAL TO COMPLY WITH THE SAFETY ANALYSES DESCRIBED IN THE DIABLO CANYON ISFSI FSAR.
2. DIMENSIONAL TOLERANCES ON THIS DRAWING ARE PROVIDED SOLELY FOR LICENSING PURPOSES TO DEFINE REASONABLE LIMITS ON THE NOMINAL DIMENSIONS USED IN LICENSING WORK. HARDWARE IS FABRICATED IN ACCORDANCE WITH THE FABRICATION DRAWINGS, WHICH HAVE MORE RESTRICTIVE TOLERANCES, TO ENSURE COMPONENT FIT-UP. DO NOT USE WORST-CASE TOLERANCE STACK-UP FROM THIS DRAWING TO DETERMINE COMPONENT FIT-UP.
3. THE REVISION LEVEL OF EACH INDIVIDUAL SHEET IN THE PACKAGE IS THE SAME AS THE REVISION LEVEL OF THIS COVER SHEET. A REVISION TO ANY SHEET(S) IN THIS PACKAGE REQUIRES UPDATING OF REVISION NUMBERS OF ALL SHEETS TO THE NEXT REVISION NUMBER.
4. APPLICABLE CODES AND STANDARDS ARE DELINEATED IN SECTIONS 3.3.4 & 3.4 OF THE FSAR.
5. ALL WELDS REQUIRE VISUAL EXAMINATION. ADDITIONAL NDE INSPECTIONS ARE NOTED ON THE DRAWING IF REQUIRED. NDE TECHNIQUES AND ACCEPTANCE CRITERIA ARE PROVIDED IN THE FSAR.
6. UNLESS OTHEWISE NOTED, FULL PENETRATION WELDS MAY BE MADE FROM EITHER SIDE OF A COMPONENT.
7. THIS COMPONENT IS IMPORTANT-TO-SAFETY, CATEGORY B, BASED ON THE HIGHEST CLASSIFICATION OF ANY SUBCOMPONENT. SUBCOMPONENT CLASSIFICATIONS ARE PROVIDED ON THE FABRICATION DRAWING.
8. ALL WELD SIZES ARE MINIMUMS EXCEPT AS ALLOWED BY APPLICABLE CODES AS CLARIFIED IN THE FSAR. FABRICATORS MAY ADD WELDS WITH HOLTEC APPROVAL.
9. WELDS IDENTIFIED WITH AN # ARE CONSIDERED NON-NF WELDS. WELDS MAY BE MADE USING PREQUALIFIED WELDS IN ACCORDANCE WITH AWS D1.1 OR PER ASME SECTION IX.
10. CHANGES TO THIS DRAWING AFFECT FABRICATION DRAWING 4431.



THIS COVER SHEET MUST BE MAINTAINED WITH THE LATEST REVISIONS OF THE DRAWING IDENTIFIED HEREIN.

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

CLIENT	PG&E
DESCRIPTION	UNDERGROUND CASK TRANSFER FACILITY COVER SHEET
PROJECT NO.	1073
P.O. NO.	3100813919
DRAWING NO.	4480
SHEET	1
TOTAL SHEETS	6
FILE PATH	G:\DRAWINGS\1073\4480



NOTES:

1. ALL ANGLES ARE FOR REFERENCE ONLY, UNLESS OTHERWISE STATED.
2. BLOCK TOLERANCES APPLY.
3. UNLESS OTHERWISE SPECIFIED, ALL THICKNESS DIMENSIONS ARE NOMINAL.
4. ALL WELDS REQUIRE VISUAL EXAMINATION (VT) PER ASME SECTION V, ARTICLE 9 WITH ACCEPTANCE CRITERIA PER ASME SECTION III, SUBSECTION NF, ARTICLE NF-5360.
5. SANDBLAST AND PAINT WITH SHERWIN WILLIAMS ZINC CLAD II HS OR ZINC CLAD II PLUS.
6. RESTRAINTS TO BE POSITIONED AT 90° INTERVALS AND ALIGNED PARALLEL AND PERPENDICULAR TO FAULT LINES OF H1 & H2 EARTHQUAKES.
7. TOP RESTRAINTS MAY BE FABRICATED FROM MULTIPLE PIECES PROVIDED THE SURFACE AREA OF THE WEDGES IS EQUAL TO OR UP TO 10% LARGER THAN SINGLE RESTRAINT ASSEMBLY DEPICTED HERE-IN.

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.4-3
2 of 6
Cask Transfer Facility Assembly

Revision 2 June 2008

TOLERANCES

INCHES [MILLIMETERS]		ANGULAR
.X ± .1	[2.5]	±1°
.XX ± .05	[1.27]	
.XXX ± .010	[.25]	

FRACTIONAL

0 TO LESS THAN 2 IN.	±1/8 IN.
2 IN. TO LESS THAN 3 FT.	±1/4 IN.
3 FT. AND GREATER	±3/8 IN.

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

CLIENT: PG&E

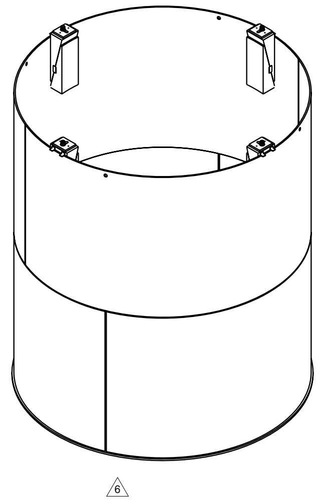
DESCRIPTION: UNDERGROUND CASK TRANSFER FACILITY ASSEMBLY AND NOTES

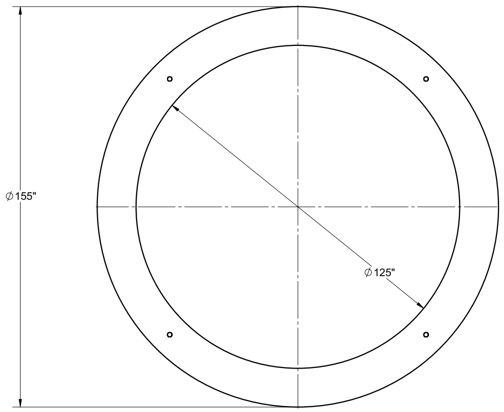
PROJECT NO. 1073
3100813919

COMPANION DRAWINGS: -

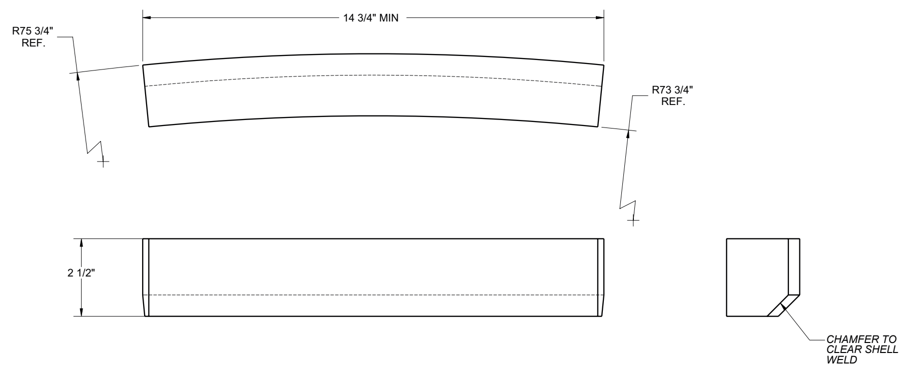
SIZE: D | DRAWING NO. 4480 | SHEET 2 | REV 6

SCALE: NONE | FILE PATH: G:\DRAWINGS\1073\4480





BASE PLATE
SCALE: 1:24



LOWER RESTRAINT
SCALE: 1:2

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.4-3
3 of 6
Cask Transfer Facility Assembly

Revision 2 June 2008

TOLERANCES

INCHES [MILLIMETERS]		ANGULAR
.X ± .1	[2.5]	±1°
.XX ± .05	[1.27]	
.XXX ± .010	[.25]	

FRACTIONAL

0 TO LESS THAN 2 IN.	±1/8 IN.
2 IN. TO LESS THAN 3 FT.	±1/4 IN.
3 FT. AND GREATER	±3/8 IN.

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

CLIENT: PG&E

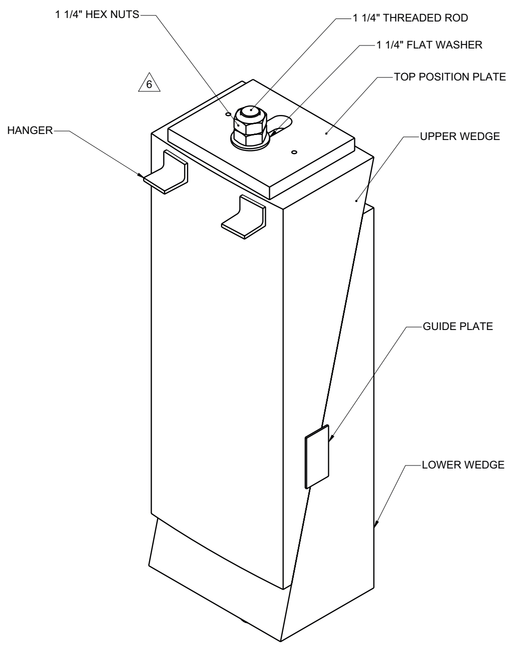
DESCRIPTION: UNDERGROUND CASK TRANSFER FACILITY DETAILS

PROJECT NO. 1073
3100813919

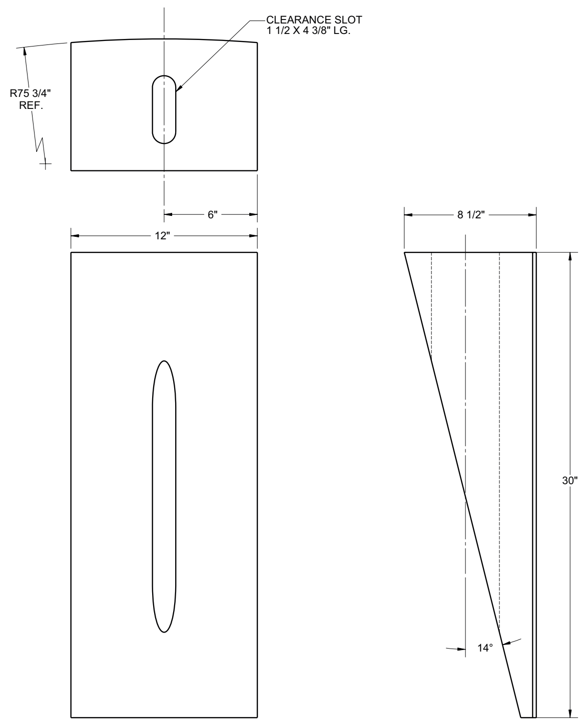
COMPANION DRAWINGS: -

SIZE: D | DRAWING NO. 4480 | SHEET 3 | REV 6

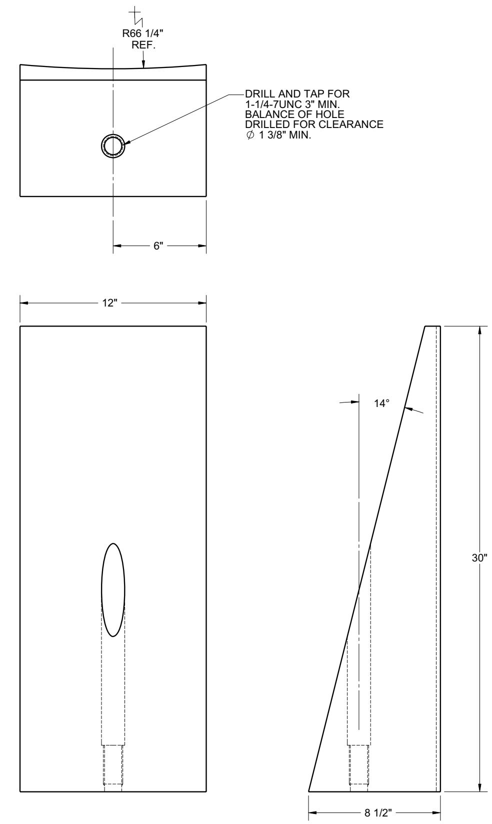
SCALE: NONE | FILE PATH: G:\DRAWINGS\1073\4480



CTF WEDGE ASSEMBLY



TOP WEDGE
SCALE 1 : 4



BOTTOM WEDGE
SCALE 1 : 4

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.4-3
4 of 6
Cask Transfer Facility Assembly

Revision 2 June 2008

TOLERANCES

INCHES [MILLIMETERS]		ANGULAR
.X ± .1	[2.5]	±1°
.XX ± .05	[1.27]	
.XXX ± .010	[.25]	

FRACTIONAL

0 TO LESS THAN 2 IN.	±1/8 IN.
2 IN. TO LESS THAN 3 FT.	±1/4 IN.
3 FT. AND GREATER	±3/8 IN.

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

CLIENT: PG&E
-

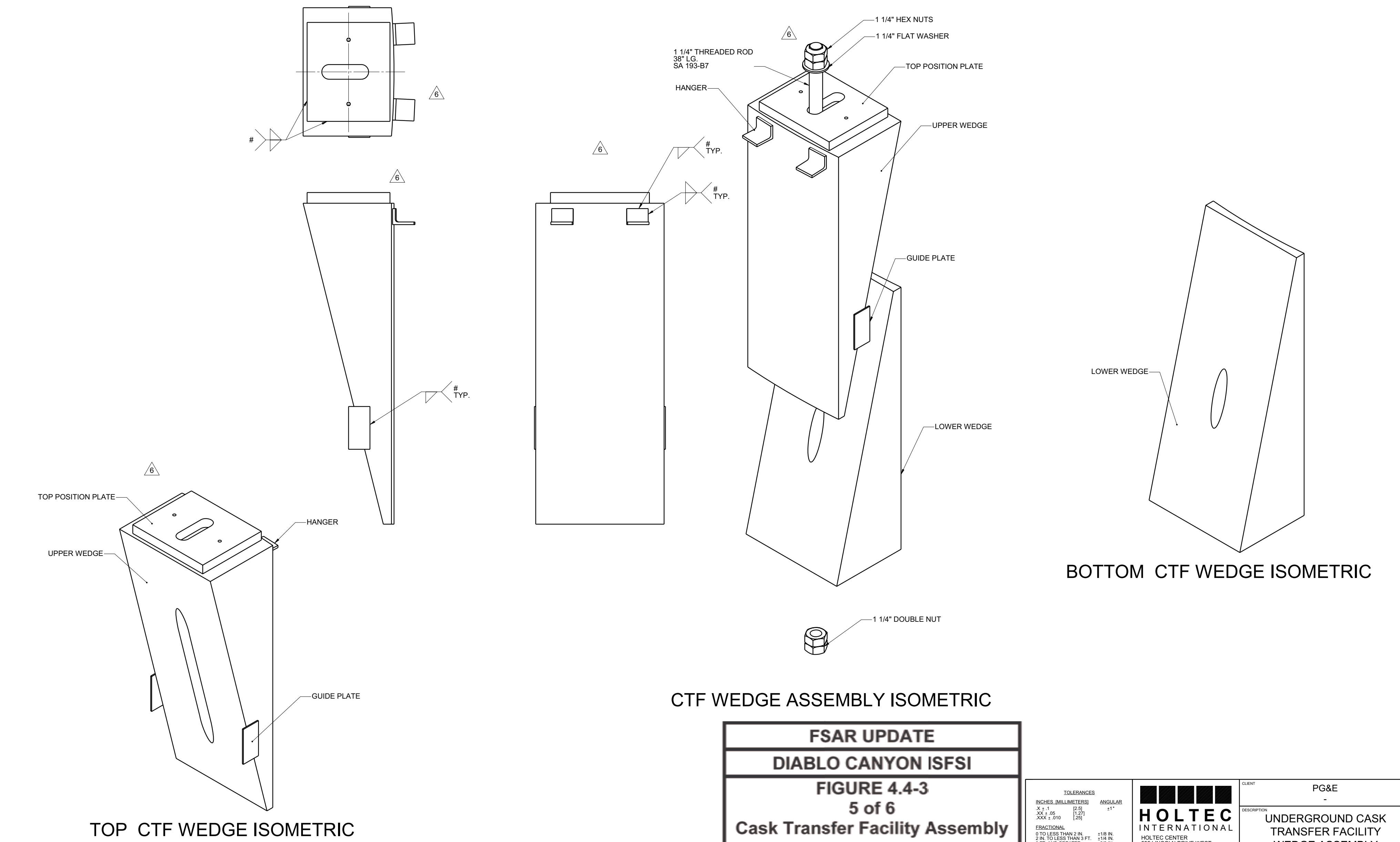
DESCRIPTION: UNDERGROUND CASK TRANSFER FACILITY WEDGE DETAILS

PROJECT NO. 1073
3100813919

COMPANION DRAWINGS: -

SIZE: D | DRAWING NO. 4480 | SHEET 4 | REV 6

SCALE: NONE | FILE PATH: G:\DRAWINGS\1073\44314480



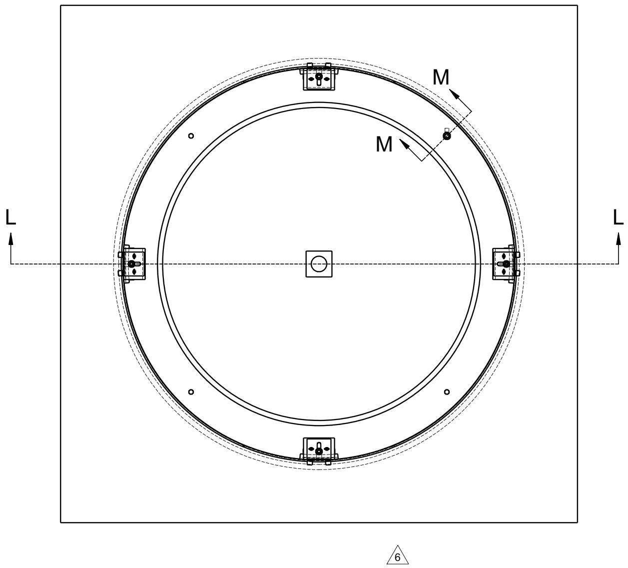
FSAR UPDATE

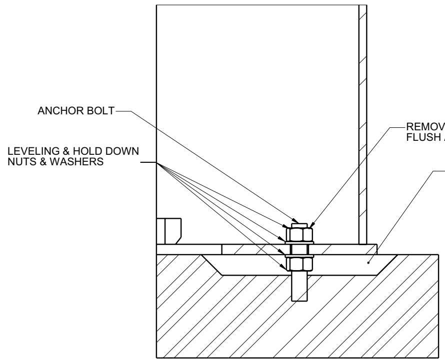
DIABLO CANYON ISFSI

FIGURE 4.4-3
5 of 6
Cask Transfer Facility Assembly

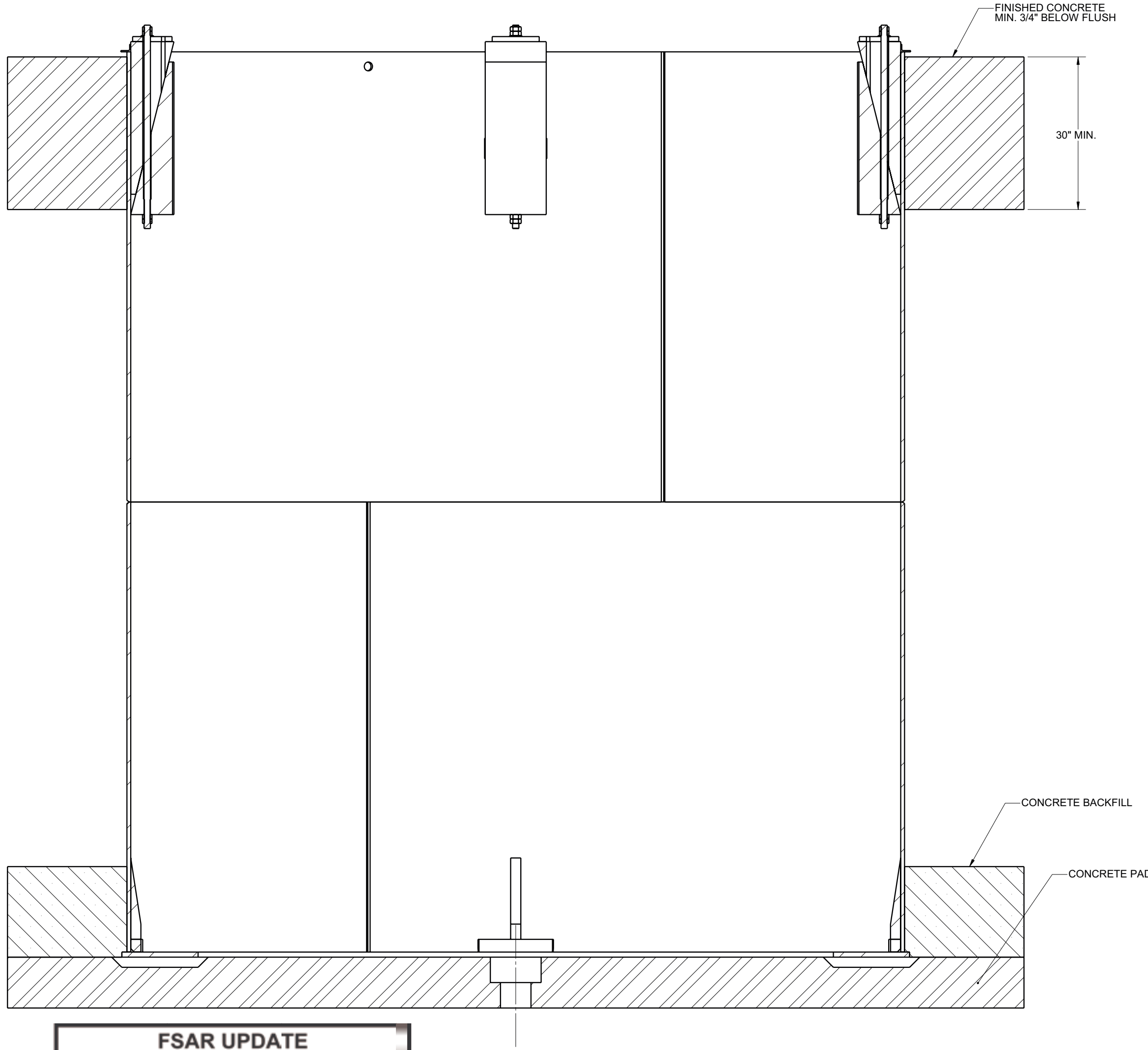
Revision 2 June 2008

TOLERANCES INCHES [MILLIMETERS] / ANGULAR .X ± .1 [2.5] ±1° .XX ± .05 [1.27] .XXX ± .010 [.25] FRACTIONAL 0 TO LESS THAN 2 IN. ±1/8 IN. 2 IN. TO LESS THAN 3 FT. ±1/4 IN. 3 FT. AND GREATER ±3/8 IN.	HOLTEC INTERNATIONAL HOLTEC CENTER 555 LINCOLN DRIVE WEST MARLTON, NJ 08053	CLIENT: PG&E - DESCRIPTION: UNDERGROUND CASK TRANSFER FACILITY WEDGE ASSEMBLY
PROJECT NO. 1073 3100813919	COMPANION DRAWINGS: -	SIZE D / DRAWING NO. 4480 / SHEET 5 / REV 6 SCALE NONE / FILE PATH: G:\DRAWINGS\1073\44314480





SECTION M-M
SCALE 1 : 6



6 SECTION L-L
SCALE 1 : 12

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 4.4-3
6 of 6
Cask Transfer Facility Assembly

Revision 2 June 200[illegible]

TOLERANCES		
INCHES [MILLIMETERS]		ANGULAR
.X ± .1	[2.5]	±1°
.XX ± .05	[1.27]	
.XXX ± .010	[.25]	
FRACTIONAL		
0 TO LESS THAN 2 IN.	±1/8 IN.	
2 IN. TO LESS THAN 3 FT.	±1/4 IN.	
3 FT. AND GREATER	±3/8 IN.	

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

CLIENT: PG&E

DESCRIPTION: UNDERGROUND CASK TRANSFER FACILITY SECTION VIEW

PROJECT NO.	COMPANION DRAWINGS:	SIZE	DRAWING NO.	SHEET	REV
1073	-	D	4480	6	6
3100813919		SCALE: NONE	FILE PATH: G:\DRAWINGS\1073\4480		

CHAPTER 5

ISFSI OPERATIONS

CONTENTS

CHAPTER 5

ISFSI OPERATIONS

TABLES

CHAPTER 5

ISFSI OPERATIONS

FIGURES

CHAPTER 5

ISFSI OPERATIONS

This chapter describes the operations associated with the Diablo Canyon ISFSI. Fuel handling and cask loading operations in the DCPP fuel handling building/auxiliary building (FHB/AB) are performed in accordance with the DCPP 10 CFR 50 license. Transfer and storage activities associated with the ISFSI are performed in accordance with the 10 CFR 72 Diablo Canyon ISFSI license. As indicated in previous chapters, the Diablo Canyon ISFSI, in its final storage configuration, is a totally passive installation. Periodic surveillance is required, by the Diablo Canyon ISFSI Technical Specifications (TS), to ensure the passive air-cooling system is properly operating. Maintenance is limited to minor, touch-up painting of the HI-STORM 100SA overpack and anchorage hardware. The operations described in this chapter relate to the loading and preparation of the multi-purpose canisters (MPCs), transport to the cask transfer facility (CTF) in the HI-TRAC transfer cask, transfer of the MPC from the transfer cask to the overpack at the CTF, and transport of the loaded overpack from the CTF to the ISFSI storage site. Also described is the process for off-normal event recovery, including unloading of fuel from a loaded overpack. An overview of activities occurring in the DCPP FHB/AB is provided. A detailed discussion of these activities is provided in the 10 CFR 50 license amendment request (LAR) and associated license amendments (References 1 and 2, respectively).

5.1 OPERATION DESCRIPTION

The methods and sequences described below provide an overview of the operational controls that the personnel performing spent fuel loading, cask transfer, and storage activities implement to ensure safe, reliable, long-term spent fuel storage at the ISFSI storage site. Site-specific procedures are used to implement these activities, including the use of existing procedures, revision of existing procedures, or the creation of new procedures. The specific number, wording, and sequence of site procedural steps may vary from the guidance provided here as long as the steps comply with assumptions and inputs in the governing, design-basis analyses.

Operations to load and place the HI-STORM 100 System at the storage location on the ISFSI pad are performed both inside and outside the DCPP FHB/AB. MPC fuel loading and handling operations are performed inside the FHB/AB using existing DCPP systems and equipment for heavy lifts, radiation monitoring, decontamination, and auxiliary support, augmented as necessary by ancillary equipment specifically designed for these functions and a single failure proof FHB/AB crane. The implementing procedures incorporate applicable 10 CFR 50 license conditions and commitments, such as those governing heavy loads. MPC transfer into the overpack at the CTF and movement of the loaded overpack to the storage location is performed using procedures developed specifically for these operations.

5.1.1 NARRATIVE DESCRIPTION

The following discussion describes the specifics of the integrated operation, including fuel loading, MPC closure operations, transfer cask handling, overpack handling, and ISFSI pad placement. As described in the HI-STORM 100 System FSAR (Reference 3) the MPC is loaded with fuel while contained in a reusable HI-TRAC transfer cask in the spent fuel pool (SFP). The MPC is welded and prepared for storage while in the FHB/AB. The MPC and transfer cask are then transported in a vertical configuration to the CTF, located adjacent to the ISFSI storage site, where the MPC is transferred into an overpack for storage on the ISFSI pads. Section 5.1.1.1 describes loading operations for damaged fuel and fuel debris. Section 5.1.1.2 describes MPC loading and sealing operations. Section 5.1.1.3 describes the operations for transferring the loaded MPC to the ISFSI storage site and into the overpack for storage. Section 5.1.1.4 describes off-normal event recovery operations.

Specific procedures identify and control the selection of fuel assemblies, and nonfuel hardware for loading into the HI-STORM 100 System. Candidate fuel assemblies are selected based on their physical characteristics (for example, dimensions, enrichment, and uranium mass) to ensure they meet the requirements of the Diablo Canyon ISFSI TS and Section 10.2. The selected fuel assemblies then are classified as intact fuel, damaged fuel, or fuel debris, in accordance with the definitions in Section 10.2. Once an assembly is found to be physically within the limits of the Section 10.2 and correctly classified, the burnup, cooling time, and decay heat of the assemblies are confirmed to be within Section 10.2 limits using existing records. Burnup uncertainty is not considered when evaluating the eligibility of fuel assemblies for storage, as an allowance for this uncertainty is not required by regulations. However, PG&E conservatively applies a 5 percent burnup uncertainty allowance when calculating the decay heat for each loaded MPC. If any selected assemblies include nonfuel hardware, the particular type of nonfuel hardware also is confirmed to meet Section 10.2.

Fuel assemblies chosen for loading are assigned a specific storage location in the MPC in accordance with the Diablo Canyon ISFSI TS and Section 10.2. Criteria such as the classification of the assembly (that is, intact, damaged, or debris), the presence of nonfuel hardware in the assembly, and the use of a uniform or regionalized storage strategy (burnup, cooling time, decay heat) as defined in Section 10.2 are used to determine the acceptable fuel storage locations for each assembly. Records are kept that track the fuel assembly, and nonfuel hardware and its assigned MPC and specific fuel storage location. Videotape (or other visual record) is used during fuel loading operations in the SFP to record fuel assembly and associated nonfuel hardware serial numbers and to provide an independent record of the MPC inventory.

Once the fuel inventory for an MPC is identified, the “time-to-boil” for that MPC is calculated based on the total decay heat rate of the fuel and the temperature of the SFP at the time of loading. This calculation establishes the time duration within which MPC sealing operations must reach the point where draining of the water in the MPC is complete and boiling of the water in the MPC is avoided. The commencement for time-to-boil starts when the MPC lid is installed in the SFP, effectively segregating the fuel in the MPC from the cooling provided by the SFP cooling system. The time-to-boil may be determined on an MPC-specific basis or a bounding time may be determined for a group of MPCs to be loaded, using a worst-case fuel decay heat value and initial water temperature. The methodology described in Section 4.5.1.1.5 of the HI-STORM 100 System FSAR (Reference 3) shall be used to determine the time-to-boil.

Additional administrative controls are used, as necessary, to govern the use of special load-handling devices, allowable travel paths, and lift heights, both inside and outside of the FHB/AB, to ensure compliance with the DCPP and Diablo Canyon ISFSI licensing and design bases, as applicable.

The loading, unloading, and handling operations described in this section have been developed based on the Holtec International field experience in loading dry cask storage systems at other ISFSIs. The equipment and operations used at these sites have been evaluated and modified, as necessary, based on this experience to reduce occupational exposures and further minimize the likelihood of human error in performing the activities needed to successfully deploy the HI-STORM 100 System at the Diablo Canyon ISFSI.

5.1.1.1 Damaged Fuel and Fuel Debris Loading

Damaged fuel containers (DFCs) are used to house damaged fuel assemblies and fuel debris in the MPC in accordance with the requirements of the Diablo Canyon ISFSI TS and Section 10.2. Any qualified fuel assembly that is classified as damaged fuel may be loaded into an MPC-24E. Up to a total of four DFCs containing damaged fuel may be stored in an MPC-24E, with the balance being intact fuel assemblies. Fuel classified as fuel debris must be stored in a DFC and must be loaded into an MPC-24EF. The MPC-24EF may also be used to store damaged fuel. Up to a total of four DFCs containing either damaged fuel or fuel debris may be stored in the MPC-24EF, with the balance being intact fuel assemblies. The fuel assembly is placed in the DFC either before or after the DFC is placed into the MPC. Storage of damaged fuel and fuel debris in the HI-STORM 100 System is discussed, and the containers analyzed, in Section 2.1.3 and Appendix 3.AS, respectively, of the HI-STORM 100 System FSAR. Figure 2.1.2B in the HI-STORM 100 System FSAR shows the Holtec pressurized water reactor (PWR) DFC.

As required by the Diablo Canyon ISFSI materials license, damaged fuel and fuel debris are only allowed in the MPC-24E and -24EF canisters. The license currently does not allow any damaged fuel or fuel debris to be stored in the MPC-32. However as a result of the modifications described in Section 1.1, the MPC-24s will likely not be used at the Diablo Canyon ISFSI and the materials license will need to be amended prior to allowing storage of damage fuel and fuel debris in the MPC-32.

5.1.1.2 MPC Loading and Sealing Operations

This section describes the general sequence of operations to load and seal the MPC, including the movement of the transfer cask within the FHB/AB. Site-specific procedures control the performance of the operations, including inspection and testing. At a minimum, these procedures control the performance of activities and alert operators to changes in radiological conditions around the cask. As described in this section, several operational sequences have important time limitations including time-to-boil following MPC lid attachment, and time limits to establish and suspend supplemental cooling. These sequences are controlled by the Diablo Canyon ISFSI TS and Section 10.2.

Several auxiliary components are used during the cask loading process. A discussion of these items is provided for the sole purpose of describing the loading process. These items, along with their design and use, are controlled under the DCPP Control of Heavy Loads Program.

A work platform in the Unit 2 cask washdown area (CWA) assists in transfer cask and MPC preparation and closure operations. The work platform is part of the transfer cask seismic restraint system.

All handling of the transfer cask inside the FHB/AB will be made using a single failure proof crane to preclude a vertical cask drop event.

Placement of loaded overpacks at the ISFSI is a cyclical process involving the movement of a loaded overpack to the ISFSI and returning with an empty transfer cask for the next loading process. The operations described herein start at the time the empty MPC is loaded into the transfer cask and is ready for movement into the FHB/AB.

An empty MPC-32 is also verified to have been cleaned, inspected, and is then raised, and inserted into the transfer cask. This insertion activity may take place either prior to entering the FHB/AB or once inside the FHB/AB. Upon completion of the insertion activity alignment marks are verified to ensure correct rotational alignment between the MPC and the transfer cask.

The transfer cask is brought into the FHB/AB through the Unit 2 roll-up door in a vertical orientation on a low-profile transporter (LPT). There is no LPT rail system for Unit 1, thus transfer casks designated for transporting spent fuel from both units enter through the Unit 2 roll-up door. If not previously installed, an empty MPC-32 will be installed when the transfer cask is in the CWA restraint. The LPT is equipped with heavy-duty rollers that engage with a set of temporary tracks that runs from inside the FHB/AB to the access road located outside the Unit 2 FHB/AB roll-up door. The track and rollers are used because dimensional limitations of the FHB/AB roll-up door prevent access of the cask transporter inside the FHB/AB.

After being transported into the FHB/AB, the transfer cask bolted to the LPT is positioned under the single failure proof FHB crane that is configured with a lift yoke. The lift yoke engages the transfer cask lifting trunnions, and the transfer cask is unbolted from the LPT. The transfer cask is then lifted above the LPT as it is moved into the Unit 2 CWA. There is no CWA seismic restraint for Unit 1, thus transfer casks designated for transporting spent fuel from both units are prepared in the Unit 2 CWA. Prior to moving the transfer cask into the CWA, the transfer cask is visually verified to have the bottom lid bolted to the cask. The transfer cask is placed within the CWA seismic restraint and secured. An empty MPC-32 is loaded into the transfer cask if not already loaded prior to entering the FHB/AB. To eliminate buoyancy effects the MPC is first filled with borated water and then the annulus between the transfer cask and the MPC is filled with demineralized water, in accordance with the ISFSI TS and Section 10.2. A seal is then installed in the top part of the annulus to minimize the risk of contaminating the external shell of the MPC.

When the transfer cask is ready for movement into the SFP, with the transfer cask engaged by the FHB crane, the transfer cask is released from the CWA seismic restraint and, along with its MPC, is raised approximately 12 inches above the floor of the FHB/AB (140 ft elevation). For Unit 1 spent fuel loading operations, the transfer cask is moved through the hot machine shop and into the FHB/AB bay area of Unit 1 and positioned adjacent to the Unit 1 SFP. For Unit 2 spent fuel loading operations, the transfer cask is positioned adjacent to the Unit 2 SFP.

The transfer cask annulus overpressure system is connected. The transfer cask is positioned over the cask recess area of the SFP and lowered using the FHB crane on to the SFP platform structure. The SFP cask restraint provides seismic restraint while the transfer cask is on the platform. The annulus over-pressure system applies a slight overpressure to the annulus to protect the MPC external shell from contamination from the SFP water in the event there is a leak in the annulus seal. When the cask is fully lowered to the platform in the cask recess area of the SFP, the lift yoke is remotely disconnected and removed from the SFP.

Fuel-loading and post-loading verification of fuel assembly identification is conducted in accordance with approved fuel-handling procedures.

For loading of damaged fuel assemblies and fuel debris in the MPC-24E or -24EF, the assembly is loaded into the DFC, and the DFC is loaded into the MPC. Optionally, an empty DFC may be first loaded into the appropriate fuel storage location in the MPC and then the damaged fuel assembly or fuel debris loaded into the DFC.

The MPC lid, with the drain line attached, is placed in position in the MPC after the completion of fuel loading, while the transfer cask is in the SFP.

The FHB crane and the lift yoke are reattached, and the transfer cask is raised until the top of the MPC just breaks the SFP water surface. Rinsing of exterior surfaces and disconnecting the annulus pressurization system is performed as the transfer cask continues to emerge from the SFP. The transfer cask is raised completely out of the SFP to clear the SFP wall and lowered to about 12 inches above the floor of the FHB/AB (140 ft elevation). For Unit 1 fuel movement, Radiation Protection will prepare the transfer cask to preclude spreading contamination, prior to moving the transfer cask through the hot shop, to the cask restraint in the Unit 2 CWA. For Unit 2 fuel movement, the transfer cask is moved directly from the Unit 2 SFP to the Unit 2 CWA. The transfer cask is moved to the Unit 2 CWA restraint system and secured. Once the transfer cask is positioned in the CWA, the lift yoke is disconnected and removed from the area. Activities involving decontamination and placement of auxiliary equipment may occur in parallel or in a different sequence based on cask-loading experience at DCPP.

Procedural controls ensure that dilution of the MPC boron concentration will not occur from removal of the HI-TRAC from the spent fuel pool, until water is removed from the MPC in the blowdown process.

A temporary shield ring may be installed in the area of the lifting trunnions to provide supplemental personnel shielding. Preparation for MPC sealing operations may now proceed. This may include the erection of scaffolding, staging of auxiliary equipment, additional cask decontamination, dose-rate surveys, and installation of temporary shielding.

As described above, fuel-assembly decay heat could eventually cause boiling of the water in the MPC after it is removed from the SFP. Therefore, MPC draining must be completed within the time-to-boil limit previously determined, which is measured beginning at the time the MPC lid is installed in the SFP and terminating at the completion of MPC draining. Should it become evident that the time-to-boil limit may be exceeded, a recirculation of the MPC water (borated as necessary in accordance with the Diablo Canyon ISFSI TS) will be performed to reduce the temperature of the water and allow a new time-to-boil value to be determined, if necessary. When the MPC water recirculation is complete, the MPC boron concentration is verified in accordance with the Diablo Canyon ISFSI TS and the time-to-boil clock is reset. This process may be repeated as necessary.

During welding operations, the MPC water volume is reduced to provide enough space between the water surface and the lid to avoid a water-weld interaction, but maintaining the fuel covered with water to ensure the fuel is not exposed to an oxidizing environment. Oxidation of Boral or Metamic panels contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during, MPC lid welding operations. In addition, the space below the MPC lid shall 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. The automated welding system is installed. The MPC-lid welding, including nondestructive examinations, is completed. Once the MPC-lid welding is complete, the MPC is filled with borated water (in accordance with the Diablo Canyon ISFSI TS), vented, and hydrostatically tested.

After an acceptable hydrostatic test has been completed, the remaining MPC water is displaced from the MPC by blowing pressurized helium gas into the vent port of the MPC, thus displacing the water through the drain line. Using helium during MPC water displacement and moisture removal ensures that there will be no oxidization of the fuel cladding during loading operations (fuel is covered with water prior to blowdown).

The moisture removal system is connected to the MPC and is used to remove the remaining liquid water from the MPC and to reduce the moisture content of the MPC cavity to an acceptable level. This is accomplished using a forced helium dehydration (FHD) system. When using the FHD system the annular gap is verified to have no water present.

When the FHD system is used, any water that has not drained from the MPC cavity is removed through introducing dry gas into the MPC cavity that absorbs the residual moisture in the MPC. This humidified gas exits the MPC and the absorbed water is removed through condensation and/or mechanical drying. During use of the FHD system, the circulated helium is monitored until it meets the dryness criteria of the DC ISFSI TS. Once this is met, the helium pressure in the MPC cavity is adjusted to within the required pressure range in accordance with the DC ISFSI TS.

Helium backfill to the required pressure and purity level ensures that the conditions for heat transfer inside the MPC are consistent with the thermal analyses and provides an inert atmosphere to ensure long-term fuel integrity.

After successful helium backfill operations, if the MPC contains any high burnup (>45,000 MWD/MTU) fuel assemblies, and temporary shielding is being utilized on the transfer cask, the supplemental cooling system (SCS) is installed and the annulus between the transfer cask and MPC is filled with demineralized water within the time required by the Diablo Canyon ISFSI TS. The MPC vent and drain port cover plates are then installed, welded, inspected, examined, and helium leak tested in accordance with ANSI N14.5-1997. The MPC closure ring is then installed, welded, and examined. The MPC closure ring provides a second welded boundary, in addition to the confinement boundary, and is described further in Section 3.3.1.1.1 that has references to the design drawings in the HI-STORM 100 System FSAR. Note that at any time after helium backfill, the supplementary cooling system (SCS) may be installed and the HI-TRAC annulus filled with demineralized water to lower the MPC temperature for transfer operations if desired.

The temporary shield ring is removed. The transfer cask and accessible portions of the MPC are checked to ensure any removable contamination is within applicable limits. Additional decontamination and surveys may be performed throughout the loading process. The transfer cask top lid is installed and secured with four bolts.

The lift yoke is re-attached to the transfer cask. The transfer cask is raised and the bottom surface of the transfer cask is decontaminated using long-handled tools or other remotely-operated devices which do not require personnel to directly access the bottom of the transfer cask.

The CWA seismic restraint is released and the FHB crane then moves the transfer cask laterally away from the CWA. The transfer cask is positioned on and bolted down to the LPT. If not performed earlier, the transfer cask and LPT are surveyed to ensure that any fixed contamination is within acceptable limits. The loaded transfer cask and LPT are then rolled out of the Unit 2 FHB/AB to an area outside of the FHB/AB where the cask transporter can access the transfer cask.

The SCS is removed from service prior to the movement of the transfer cask from the CWA restraint to the cask transporter.

5.1.1.3 Transfer to the ISFSI Storage Site

The cask transporter and associated ancillaries, described in Section 4.3, are positioned outside the FHB/AB doors to receive the transfer cask. The transporter receives preoperational testing and maintenance and is operated in accordance with the Cask Transportation Evaluation Program in the Diablo Canyon ISFSI TS, which evaluates and controls the transportation of loaded MPCs between the DCPP FHB/AB to the CTF and ISFSI. The transfer cask on the LPT is positioned under the lift beam of the cask transporter and the transfer cask lift links are rigged to the cask. The transporter lift system engages the transfer cask while the transfer cask is unbolted from the LPT. The transporter than raises the transfer cask and it is secured within the transporter for the trip to the CTF. The LPT is than rolled out of the way and the transporter transports the transfer cask to the CTF along the approved transportation route as described in Section 4.3.3 and shown in Figure 2.1-2.

The overpack is prepared for loading, which involves general visual inspections and cleaning. Following the visual inspection and cleaning, the overpack is positioned in the CTF by the transporter. In preparation for receiving the loaded MPC, the overpack lid is removed (if previously installed). The mating device is secured to the overpack. To restrain the cask against seismically-induced impact loads on the main shell of the CTF, seismic restraints are installed to transmit the load from the overpack to the CTF shell (Section 3.3.4.2.3).

At the CTF, the transporter positions the transfer cask over the mating device and the transfer cask is then secured to the mating device. During this connection process, subsequent to MPC transfer, HI-TRAC removal, and HI-STORM closure operation, temporary shielding is provided around the mating device as needed to minimize occupational dose. Use of the temporary shielding during these processes will be administratively controlled. The cask transporter seismic anchor (TSA) restraints connect the cask transporter to the CTF TSA pads. The TSA restraints are described in Section 4.2.1.2 and depicted in plan view in Reference 39 of Section 4.2. The TSAs function to prevent the transporter from seismically interacting with the storage cask while in the CTF during MPC transfer operations. The transfer cask lift links are then disconnected and the MPC lift cleats are installed. The MPC downloader slings are attached between the cask transporter towers and the MPC lift cleats, and the MPC is raised slightly to remove the weight of the MPC from the bottom lid. The bottom lid is supported by the mating device while the bottom lid bolts are removed. The bottom lid is removed from under the transfer cask.

The transporter towers are used to lower the MPC into the overpack. The MPC downloader slings are disconnected from the cask transporter and lowered onto the MPC lid. The lift links are reengaged on the transfer cask and the transporter lift system is engaged. The cask transporter TSA restraints are disconnected and the transfer cask is unfastened and lifted from the mating device and raised from the top of the overpack and placed beside the CTF. The lift cleats and MPC downloader slings are removed, and threaded inserts are installed in the MPC lid lift holes where the lift cleats were attached. The mating device containing the transfer cask bottom lid is removed from the overpack and placed in a nearby location.

The overpack lid is installed. The overpack lifting brackets are attached. The cask transporter is positioned with its lift beam above the overpack. The overpack is lifted out of the CTF by the transporter and moved to the ISFSI pad, where it is placed in its designated storage location. During the transporter lifting of the HI-STORM, the probability of an earthquake occurring is so small as to make this event non-credible. Thus, the TSAs do not need to be attached to the transporter during the overpack lifting. Specific steps involved in these operations are described in the Diablo Canyon ISFSI approved procedures.

Prior to the loaded overpack arriving at the ISFSI pad, the designated storage location has been prepared for the cask to be placed on the pad. Specifically, a small number of alignment pins are installed in the anchor stud locations. These alignment pins ensure that the cask is properly located and the holes in the cask bottom flange match with the holes in the ISFSI pad embedment plate. When the cask is properly located and seated, the alignment pins are removed and the 16 anchor studs are threaded into the top of the embedded coupling (see Figure 4.2-2). The studs are pre-tensioned using a stud tensioner and the nuts tightened in a cross-pattern, roughly 180 degrees apart, to avoid uneven loads on the baseplate.

The preload on the cask anchor studs is applied without employing a torque wrench. Therefore, no torque is induced on the embedded anchor rods or compression couplings during the preload operation. A stud tensioner is used to apply preload on the anchor studs using hydraulic pressure to elastically “stretch” the bolt. The nuts are then tightened on the “stretched” stud to maintain the pre-load. This tension is transferred to the cask base/embedment plate interface as a compressive force via the stud nut and compression coupling. There is no significant torque applied on the nuts during tightening (i.e., hand-tightening is adequate).

The cask transporter is disconnected from the overpack and the lift brackets are removed and lid studs installed on the overpack. The grounding cables are attached to the overpack. The overpack duct photon attenuators (also known as gamma shield cross plates) are installed in the upper and lower air ducts and screens are secured. The anchor studs and nuts are covered with a metal cap for protection from the elements.

5.1.1.4 Off-Normal Event Recovery Operations

The analysis of off-normal and accident events, as defined in ANSI/ANS-57.9 (Reference 6) and as applicable to the Diablo Canyon ISFSI, is presented in Chapter 8. Each postulated off-normal and accident event analyzed and discussed in Chapter 8 addresses the event cause, analysis, and consequences. Suggested corrective actions are also provided for off-normal events. The actual cause, consequences, corrective actions, and actions to prevent recurrence (if required) will be determined through the DCPP corrective action program on a case-specific basis. All corrective actions will be taken in a timely manner, commensurate with the safety significance of the event. Of primary importance in the early response to any event will be the verification of continued criticality prevention, the protection of fuel cladding integrity (that is, heat removal), and the adequacy of radiation shielding while longer-term corrective actions are developed. This may also involve the need for temporary shielding or cask cooling in accordance with the recommendations of PG&E technical staff personnel, based on the event conditions.

Should the need arise, the MPC can be returned to the SFP for unloading. To unload an overpack or transfer cask, the operations described above are effectively executed in reverse order from the point in the operation at which the event occurred. Should any MPCs loaded under Amendment 2 of this license require unloading, the use of the supplementary cooling system shall be utilized. Once the transfer cask is back in the FHB/AB, the transfer cask top lid is removed, and preparations are made to reopen the MPC in the SFP. This involves first grinding out the welds and removing the MPC closure ring and vent and drain port cover plates. A sample of the gas inside the MPC may be drawn to determine the extent of fuel cladding failure, if any. Then, the helium cooldown system is connected and used to recirculate the helium in the MPC to cool it to a temperature at or below the maximum-allowed temperature for reflooding in accordance with the Diablo Canyon ISFSI TS and Section 10.2. Cooling the helium allows the MPC to be reflooded with water (borated as necessary) with a minimal amount of flashing and the associated undesirable pressure spikes in the MPC cavity. (Using helium for cooling prior to reflood ensures the fuel is not exposed to an oxidizing environment during unloading operations.) Based on the time the cask has been in storage, a new time-to-boil may be determined using a lower decay heat value than was used when the cask was loaded. When the MPC has been reflooded, the time-to-boil clock is started. The weld removal system is used to cut the MPC lid weld, freeing the lid for subsequent removal.

Oxidation of Boral or Metamic panels and aluminum components contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during, MPC lid cutting operations. In addition, the space below the MPC lid shall be exhausted prior to, and during, MPC lid cutting operations to provide additional assurance that explosive gas mixtures will not develop in this space. When the lid weld has been successfully cut, the lid retention device and lift yoke are installed, and the transfer cask is returned to the SFP using the same procedures and equipment as used to remove the transfer cask from the SFP after fuel loading.

Once in the SFP, the MPC lid is removed, and the spent fuel assemblies are removed from the MPC and placed back into the wet storage racks. The time-to-boil consideration is stopped once the MPC lid is removed.

5.1.2 ADDITIONAL DESCRIPTIONS

A detailed description of the operations is provided in Section 5.1.1. Radiation source terms are discussed in Chapter 5 of the HI-STORM 100 System FSAR for the generic cask analyses and in Section 7.2 of this FSAR for site-specific dose analyses. Equipment descriptions, with dimensions, design and operating characteristics, materials of construction, special design features, and operating characteristics are provided in Sections 3.3, 4.2, 4.3, and 4.4. Generic cask component design drawings are found in Section 1.5 of the HI-STORM 100 System FSAR.

5.1.3 IDENTIFICATION OF SUBJECTS FOR SAFETY AND RELIABILITY ANALYSIS

5.1.3.1 Criticality Prevention

A summary description of the principal design features, procedures, and special techniques used to preclude criticality in the design and operation of the HI-STORM 100 System is provided in Section 3.3.1.4. Additional detail on the criticality design of the storage cask is provided in Section 4.2.3.3.5.

5.1.3.2 Instrumentation

No instrumentation is required to detect off-normal operations of the HI-STORM 100 System while in its final storage configuration at the ISFSI storage site. The cask system is designed to maintain confinement integrity under all design-basis normal, off-normal, and accident conditions. Detection of degradation in the HI-STORM 100 heat removal system is accomplished by a Diablo Canyon ISFSI TS that requires periodic visual surveillance of the overpack inlet and outlet air ducts to ensure they remain free of blockage. If blockage is detected, action can be taken to remove the source of the blockage in a short time period, typically within one operating shift.

Examples of measuring and test equipment (M&TE) used during the preparation of the cask for storage operations are listed in Table 5.1-1. Additional, or different M&TE, may be used as determined through the development of site-specific operating procedures, including the revision of those procedures as experience in cask loading operations is gained and the state of the art evolves.

5.1.3.3 Maintenance Techniques

The HI-STORM 100 System is designed to safely store spent nuclear fuel with no regularly required maintenance. The only expected maintenance is to apply touch-up repair coatings to the overpack and/or the anchorage hardware due to exposure to the elements and normal wear and tear.

5.1.4 REFERENCES

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

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

3. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 1A, January 2003.

4. Deleted in Revision 2.

5. ANSI N14.5-1997, Leakage Tests on Packages for Shipment, American National Standards Institute.

6. ANSI/ANS-57.9-1992, Design Criteria for an Independent Spent Fuel Storage Installation (dry type), American National Standards Institute.

5.2 CONTROL ROOM AND CONTROL AREAS

Due to the welded closure of the MPC, the passively-cooled storage cask design, and the Diablo Canyon ISFSI TS requirement for periodic checks of the casks, the Diablo Canyon ISFSI does not require continuous surveillance and monitoring or operator actions to ensure that its safety functions are performed during normal, off-normal, or postulated accident conditions. Therefore, a control room or control area is not considered necessary, as allowed by 10 CFR 72.122(j).

Normal loading and unloading operations take place in the DCPP fuel handling building/auxiliary building under local control and in coordination with the DCPP control room staff and subject to the controls established under the DCPP 10 CFR 50 license.

Operation during the transport phase is under local control by DCPP personnel.

5.3 SPENT FUEL ACCOUNTABILITY PROGRAM

Accountability and control of spent fuel are maintained at all times during loading, transfer, and storage operations. Loading, transfer, and inventory records for spent fuel moved from the DCPP spent fuel storage pools to the Diablo Canyon ISFSI storage site are maintained in accordance with existing DCPP procedures. The Diablo Canyon ISFSI storage site is treated as a separate material balance area from DCPP.

As required by 10 CFR 72.72, records are maintained showing the receipt, inventory (including location), disposal, acquisition, and transfer of all spent fuel and radioactive waste in storage. In addition, accountability records for all fuel assemblies transferred to, stored at, or removed from the Diablo Canyon ISFSI are maintained for as long as fuel assemblies are stored at the ISFSI and retained for a period of 5 years after the fuel is transferred out of the ISFSI. PG&E requested an exemption from 10 CFR 72.72(d), which requires that spent fuel and high-level radioactive waste records in storage be kept in duplicate. The NRC granted the exemption, and, as specified in License Condition 16 of the Diablo Canyon ISFSI License SNM-2511, the exemption allows PG&E to maintain records of spent fuel and high-level radioactive waste in storage either in duplicate, as required by 10 CFR 72.72(d), or, alternatively, a single set of records may be maintained at a records storage facility that satisfies the standards of ANSI N 45.2.9-1974. All other requirements of 10 CFR 72.72(d) must be met.

All nonfuel hardware associated with the DCPP spent fuel assemblies is identified by a unique serial number permanently stamped or engraved on the hardware. Verification of the nonfuel serial numbers is made to ensure that only appropriate nonfuel hardware is stored with the spent fuel assemblies. The verification includes verifying in which fuel assembly the nonfuel hardware is stored.

Material status reports are completed and submitted to the NRC as specified in 10 CFR 72.76. Nuclear material stored at the ISFSI is not expected to be transferred from PG&E until eventual transfer to DOE for transportation to a DOE storage facility. Therefore, Nuclear Transaction reports (DOE/NRC Form-741) required by 10 CFR 72.78 are not needed until that time.

5.4 SPENT FUEL TRANSPORT

Spent fuel transport from the fuel handling building/auxiliary building to the cask transfer facility and, subsequently, to the ISFSI storage pads, is accomplished using a specifically designed transporter. Design criteria for the transporter are presented in Sections 3.2 and 3.3.3. A description of the transporter is provided in Section 4.3. Operation of the transporter is described in Sections 4.4.1.2.4 and 5.1.1.3. The location and construction features of the transport route are described in Section 4.3.3.

TABLE 5.1-1

MEASURING AND TEST EQUIPMENT

Instrument	Function
Contamination Survey, Radiation Monitoring Instruments	Measures contamination levels and dose rate levels on HI-STORM 100SA overpack, MPC lid, HI-TRAC transfer cask and ancillaries.
Flow Rate Monitor	Monitors gas flow rate during assembly cool-down.
Helium Mass Spectrometer Leak Detector	Ensures leakage rates are within acceptance criteria.
Pressure and Vacuum Gauges	Ensures correct helium backfill and MPC dryness during loading operations.
Temperature Gauge	Monitors the state of fuel cooldown prior to MPC flooding and ensures MPC dryness during loading operations when FHD system is used.
Water Totalizer	Used for water pumpdown prior to lid welding operations.

CHAPTER 6

WASTE MANAGEMENT

CONTENTS

CHAPTER 6

WASTE MANAGEMENT

6.1 MPC CONFINEMENT BOUNDARY DESIGN

The MPC is designed to endure normal, off-normal, and accident conditions of storage with maximum decay heat loads without loss of confinement. The MPC confinement boundary ensures that there is no release of radioactive materials from the cask storage system under all postulated loading conditions. Refer to Chapter 3 for additional detail regarding confinement barriers and systems.

6.2 RADIOACTIVE WASTES

No radioactive wastes are generated due to transport or storage of the loaded MPC at the ISFSI. Radioactive wastes generated during MPC loading operations in the fuel handling building/auxiliary building (FHB/AB) are treated using existing DCPP radioactive waste control systems as described in the DCPP Final Safety Analysis Report (FSAR) Update, Chapter 11, “Radioactive Waste Management” (Reference 1).

Contaminated water from loaded MPCs normally is drained back into the spent fuel pool with no additional processing. A small amount of liquid waste results from transfer cask and MPC decontamination. The decontamination procedure may result in a small amount of detergent/demineralized mixture being collected in the FHB/AB. Liquid wastes in this area are directed to the liquid radwaste disposal system.

If necessary, potentially contaminated air and helium from the MPC during loading and unloading operations will be connected to the gaseous radwaste system. A small quantity of low-level solid waste may be generated during MPC loading operations. The solid waste may include disposable anti-contamination garments, paper, rags, tools, etc., and will be processed as described in the DCPP FSAR Update, Section 11.5, “Solid Waste System.”

Any water collected in the cask transfer facility sump is sampled before it is discharged. If the water is found to be contaminated, it will be disposed of in accordance with the DCPP radioactive waste management program.

6.3 REFERENCES

1. Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update.

CHAPTER 7

RADIATION PROTECTION

CONTENTS

CHAPTER 7

RADIATION PROTECTION

TABLES

CHAPTER 7

RADIATION PROTECTION

TABLES (Continued)

CHAPTER 7

RADIATION PROTECTION

TABLES (Continued)

CHAPTER 7

RADIATION PROTECTION

FIGURES

CHAPTER 7

RADIATION PROTECTION

This chapter provides information regarding the radiation protection design features of the ISFSI and the estimated onsite and offsite doses expected due to operation of the Diablo Canyon ISFSI. The generic HI-STORM 100 System, described in the HI-STORM 100 System FSAR (Reference 1), is deployed at the Diablo Canyon ISFSI. The generic shielding analyses, including methodology, computer codes, and modeling were performed and licensed in accordance with NUREG-1536. These same, previously-licensed techniques were used in performing the site-specific analyses described in this chapter.

The Diablo Canyon ISFSI was initially licensed based on HI-STORM 100 CoC, Amendment 1, and used the applicable source terms based on fuel which providing limiting dose rates within the allowed loadable contents for the canister. Additionally, the original Diablo Canyon ISFSI License utilized canisters with an allowed leakage rate from the MPC, and hence a confinement dose analysis was performed to document potential effluent doses from the allowed MPC leakage.

During the development of License Amendment 2 (LA 2), to allow loading of high burnup fuel, the dose analyses were re-performed. Although the allowed loading of fuel was based on HI-STORM 100 CoC, Amendment 3, the revised dose analysis was performed using the HI-STORM 100 CoC, Amendment 5, source terms, which results in overstated doses since the Amendment 5 fuel was allowed to be loaded at a higher heat load, and hence higher dose rate. Additionally, as part of the system changes developed for LA 2, the helium leak testing requirements for the MPC shell welds were revised to require them to meet the "leaktight" criteria of ANSI N14.5-1997. The vent and drain port cover plate welds helium leak testing requirements had been changed to the “leaktight” criteria of ANSI N14.5-1997 in LA 1. Since the lid-to-shell (LTS) weld is a large, multi-pass weld which is placed and inspected in accordance with ISG-15; therefore in accordance with ISG-18, leakage from this weld is considered non-credible. Because all the closure welds meet a leaktight criteria, the confinement boundary of the subsequently fabricated MPCs can be considered leak tight, and no dose contribution from confinement boundary leakage is required to be considered for the casks loaded to these requirements.

To preserve the previous licensing basis, where the previous analyses have not been superseded by the updated analyses, the new data is provided along side the previous data. When the information is contained in the Tables, the data supporting the current analyses is provided in the Table designated "A", and the previous data is in the Table designated "B".

7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE

7.1.1 POLICY CONSIDERATION AND ORGANIZATION

It is the policy of Pacific Gas and Electric Company (PG&E), through Nuclear Power Generation (NPG), to design, operate, and maintain the Diablo Canyon ISFSI in a manner that maintains personnel radiation doses as low as is reasonably achievable (ALARA).

DCPP's ALARA program, which complies with the requirements of 10 CFR 20 and 10 CFR 50, is considered sufficient for ISFSI operations under 10 CFR 72. The ALARA program is implemented through NPG program directives, administrative procedures, and working level procedures. These documents have been revised to address ISFSI operations.

The Health Physics Program used for operating the Diablo Canyon ISFSI is described in Section 7.6 and implements the requirements of 10 CFR 20, 10 CFR 72, and the NPG policy for implementation of the ALARA philosophy for all site activities involving potential radiation exposure. The Radiation Protection Manager is responsible for administering, coordinating, planning, and scheduling all radiation protection activities involving the ISFSI.

The primary objective of the Health Physics Program is to maintain radiation exposures to workers, visitors, and the general public below regulatory limits and otherwise ALARA.

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

Specific design- and operations-oriented ALARA considerations are described in the following sections.

7.1.2 DESIGN CONSIDERATIONS

The Diablo Canyon ISFSI storage pad site is located in an area adjacent to the raw water reservoir. The location was chosen based on two ALARA considerations as follows:

- The ISFSI is centrally located within the DCPP site boundary, thus maintaining offsite doses ALARA.

- The ISFSI is sufficiently distant from buildings and occupied spaces so that the doses to onsite personnel are maintained ALARA.

The layout of the ISFSI storage pads is designed to minimize personnel exposures during routine surveillance, maintenance, and repair activities. The overpacks are sufficiently spaced to allow adequate personnel access between the casks.

Regulatory Position 2 of NRC Regulatory Guide 8.8 (Reference 3) provides guidance regarding facility and equipment design features. This guidance has been followed in the design of the Diablo Canyon ISFSI and the HI-STORM 100 System as described below:

- Regulatory Position 2a, regarding access control, is met by the use of a restricted area fence for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials and a security perimeter fence with a locked gate that surrounds the ISFSI storage pad and prevents unauthorized access.

- Regulatory Position 2b, regarding radiation shielding, is met by the storage cask and transfer cask biological shielding that minimizes personnel exposure to the extent practicable. Fundamental design considerations that directly influence occupational exposures and which have been incorporated into the HI-STORM 100 System design include:

  - Minimization of the number of handling and transfer operations for each spent fuel assembly

  - Minimization of the number of handling and transfer operations for each MPC loading

  - Maximization of fuel capacity, thereby taking advantage of the self-shielding characteristics of the fuel and the reduction in the number of MPCs that must be stored at the ISFSI

  - Minimization of planned maintenance requirements

  - Minimization of decontamination requirements at ISFSI decommissioning

  - Optimization of the placement of shielding with respect to anticipated worker locations and fuel placement during loading and transfer operations

  - A thick-walled overpack that provides gamma and neutron shielding

- A single, thick MPC lid (rather than separate structural and shield lids) that provides effective shielding for operators during MPC loading and transfer operations

- Multiple welded barriers to confine radionuclides

- Smooth surfaces to reduce decontamination times

- MPC penetrations located and configured to reduce streaming paths

- Overpack and transfer cask designed to reduce streaming paths

- MPC vent and drain ports, with remotely operated valves, to prevent the release of radionuclides during loading and unloading operations and to facilitate draining, drying, and backfill operations

- Use of an annulus overpressure system to minimize contamination of the MPC shell outer surfaces during loading operations

- Minimization of maintenance to reduce doses during storage operation

- Use of a dry environment inside the MPC cavity to preclude the possibility of release of contaminated liquids.

• Regulatory Position 2c, regarding process instrumentation and controls, is met since there are no radioactive systems at the ISFSI.

• Regulatory Position 2d, regarding control of airborne contaminants, is met since the HI-STORM 100 System is designed to withstand all normal, off-normal, and accident design-basis conditions without loss of confinement function, as described in Chapter 7 of the HI-STORM 100 System FSAR (Reference 1). Therefore, no gaseous releases are anticipated. No significant surface contamination is expected since the exterior of the MPC is kept clean by using clean demineralized water in the transfer cask MPC annulus and by using an inflatable annulus seal to preclude spent fuel pool (SFP) water contacting the exterior surface of the MPC.

• Regulatory Position 2e, regarding crud control, is not applicable to the Diablo Canyon ISFSI since there are no radioactive systems at the ISFSI that could transport crud.

• Regulatory Position 2f, regarding decontamination, is met since the exterior of the loaded transfer cask is decontaminated prior to being removed from the DCPP fuel handling building/auxiliary building (FHB/AB). The exterior surface of the transfer cask is designed with a minimal number of crud traps and a smooth, painted surface for ease of decontamination. In addition, an inflatable annulus seal and annulus overpressure system are used to prevent SFP water from contacting and contaminating the exterior surface of the MPC.

- Regulatory Position 2g, regarding monitoring of airborne radioactivity, is met since the MPC provides confinement for all design basis conditions. There is no need for monitoring since no airborne radioactivity is anticipated to be released from the casks at the ISFSI.

- Regulatory Position 2h, regarding resin treatment systems, is not applicable to the Diablo Canyon ISFSI since there are no treatment systems containing radioactive resins.

- Regulatory Position 2i, regarding other miscellaneous features, is met because the ISFSI storage pad is located in a cut into an existing hill and located away from normally-occupied power plant areas. The hill provides natural shielding on one side and partial shielding on two sides, and the ISFSI pads are set back a sufficient distance from the controlled area boundary to ensure low dose rates in the uncontrolled area. In addition, the MPC is constructed from stainless steel. This material is resistant to corrosion and the damaging effects of radiation, and is well proven in spent nuclear fuel storage cask service.

7.1.3 OPERATIONAL CONSIDERATIONS

Operating procedures for the Diablo Canyon ISFSI, including cask loading, unloading, transfer to the cask transfer facility (CTF), MPC transfer, and movement to the ISFSI storage pad are detailed in Chapter 5. The operating procedures were developed with an underlying ALARA philosophy and have been modified, as appropriate, to incorporate lessons learned from actual loading campaigns conducted at Diablo Canyon and other nuclear power plants. ISFSI personnel follow site-specific implementing procedures consistent with the philosophy of Regulatory Guides 8.8 and 8.10. Personnel radiation exposure during ISFSI operations is minimized through the incorporation of the following concepts:

- Fuel loading procedures that follow accepted practice and build on lessons learned from operating experience

- Preparation of the loaded MPC and transfer cask inside the FHB/AB using existing plant equipment and procedures, where possible

- Use of an optional regionalized loading strategy, where feasible, to take advantage of shielding provided by placing lower burnup and longer cooled fuel assemblies on the periphery of the MPC basket

- Filling of the annulus between the MPC and the transfer cask with clean demineralized water and using the inflatable annulus seal and annulus overpressure system to minimize contamination of the outer surface of the MPC

- Performance of as many MPC preparation activities as possible with water in the MPC cavity

- Maintaining the transfer cask water jacket filled with water during MPC processing

- Use of temporary portable shielding, as appropriate

- Use of power-operated tools, when possible, to install and remove bolts on the transfer cask and overpack

- Consideration of the ALARA philosophy in job briefings prior to fuel movement, cask loading, and MPC preparation

- Use of classroom training, mock-ups and dry-run training to verify equipment operability and procedure adequacy and efficiency.

7.1.4 REFERENCES

1. Final Safety Analysis Report for the HI-STORM 100 Cask System, Holtec International Report No. HI-2002444, Revision 1A, January 2003.

2. Deleted in Revision 2.

3. Regulatory Guide 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable, USNRC, June 1978.

4. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 3, May 29, 2007.

5. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 5, July 14, 2008.

6. Holtec International Report No. HI-2002563, "Dose Evaluation for the ISFSI at Diablo Canyon Power Plant," Revision 10.

7.2 RADIATION SOURCES

The source terms presented in this section were developed specifically for use in the Diablo Canyon ISFSI shielding analyses, HI-2002563, Revision 10 (Reference 10). Other sections of this FSAR reference dose analyses from the HI-STORM 100 System FSAR (Reference 1) and HI-STORM 100 System FSAR Revision 7 (Reference 11). The source terms used for the dose analyses referenced from the HI-STORM 100 System FSAR are contained in those documents and, therefore, are not repeated in this section.

7.2.1 CHARACTERIZATION OF SOURCES

Shielding analyses for dose rates from direct radiation were performed assuming that the overpacks contain MPC-32s completely loaded with fuel assemblies having identical burnup and cooling times. In the original analysis, burnup was assumed to be 32,500 MWD/MTU with an initial cooling time of 5 years. To allow the HI-STORM 100 system at Diablo Canyon to be loaded with high burnup fuel, the shielding analysis was reperformed in support of License Amendment 2 (LA 2). The burnup assumed was increased to 69,000 MWD/MTU for assemblies of 4.8 wt% U-235 initial enrichment, with an initial cooling time of 5 years.

In the estimation of the doses presented in Sections 7.4 and 7.5, credit was taken for additional cooling time from 5 years to 20 years as the casks are placed at the ISFSI over time. An annual loading campaign of eight casks each year was assumed. This initial burnup and cooling time value is based on Section 10.2 for uniform fuel loading. It is demonstrated in Section 7.3 that the dose rates on the surface of the overpack calculated using this burnup and cooling time bound the dose rates calculated using the other allowable burnup and cooling times. In addition, it is demonstrated that the dose rates calculated for an overpack containing an MPC-32 bound the dose rates calculated for an overpack containing an MPC-24, MPC-24E, or MPC-24EF.

The original shielding analysis for the transfer cask that is presented in this chapter was performed for the MPC-24 using a burnup and cooling time of 55,000 MWD/MTU and 12 years, respectively, based on Section 10.2 for uniform loading. It is demonstrated in Section 7.3 that the dose rates on the surface of the transfer cask using this burnup and cooling time bound the dose rates using other allowable burnup and cooling times. It is also demonstrated that the dose rates from a transfer cask containing an MPC-24 bounded the dose rates from a transfer cask containing an MPC-32.

In the revised analysis for high burnup fuel, the transfer cask shielding analysis used the MPC-24 analysis from the HI-STORM 100 FSAR Revision 7 (Reference 11), which used a burnup of 75,000 MWD/MTU and cooling time of 5 years. This combination provides conservative doses as it exceeds the fuel allowed for loading in the system allowed by Section 10.2. To estimate the dose for an MPC-32, these doses are multiplied by the ratio of assemblies contained, which provides conservative results since it does not take into consideration the increased self-shielding in the MPC-32.

A review of the fuel inventory, as of November 2000, indicates that fuel assemblies with burnups between 30,000 and 35,000 MWD/MTU have an average initial enrichment of 3.01 wt percent ^{235}U and that assemblies with burnups between 50,000 and 55,000 MWD/MTU have an average initial enrichment of 4.2 wt percent ^{235}U. Since lower enrichments result in slightly higher neutron source terms, enrichments of 2.9 and 4.0 wt percent ^{235}U were conservatively used for the original analysis of the 32,500 and 55,000 MWD/MTU burnups, respectively.

The principal sources of direct radiation in the HI-STORM 100 System are:

- Gamma radiation originating from the following sources
  - Decay of radioactive fission products
  - Secondary photons from neutron capture in fissile and nonfissile nuclides
  - Hardware activation products generated during power operations
- Neutron radiation originating from the following sources
  - Spontaneous fission
  - Alpha, neutron (α, n) reactions in fuel materials
  - Secondary neutrons produced by fission from subcritical multiplication
  - Gamma, n (γ, n) reactions (this source is negligible)
  - Neutron Source Assemblies

The foregoing can be grouped into three distinct sources, each of which is discussed below: fuel-gamma source, fuel-neutron source, and nonfuel-hardware-activation source. The source terms for the analyses presented in this FSAR were calculated using the same methods described in the HI-STORM 100 System FSAR. The neutron and gamma source terms, along with the quantities of radionuclides available for release, were calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system (References 3 and 4, respectively).

7.2.1.1 Design-Basis Fuel Assembly

The physical characteristics of the fuel used at DCPP are summarized in Table 3.1-1 and Section 10.2.

Section 5.2 of the HI-STORM 100 System FSAR describes the design basis pressurized water reactor (PWR) fuel assembly based on a comparison of source terms from the PWR fuel assembly classes permitted for storage under the HI-STORM 100 System general certification. It was determined that the B&W 15-by-15 fuel assembly, which has the highest uranium mass of the allowable fuel assemblies, was the assembly with the highest radiation source and therefore was the design-basis fuel assembly. Since the fuel assemblies used for DCPP are permitted for storage under the HI-STORM 100 general certification, they are bounded by the determination of the design-basis fuel assembly in the HI-STORM 100 System FSAR. Therefore, for conservatism, the B&W 15-by-15 design basis PWR fuel assembly described in Table 5.2.1 of the HI-STORM 100 System FSAR was used for the analysis presented in this chapter. Tables 5.3.1 and 5.3.2 of the HI-STORM 100 System FSAR describe the axial location of the sources in the fuel assembly and the material composition of the assembly. The axial burnup profile used in these analyses and the position of the assembly within the MPC were identical to those described in Chapter 2 of the HI-STORM 100 System FSAR.

The HI-STORM 100 System FSAR describes the shielding analysis to qualify generic damaged fuel assemblies. The discussion in Section 5.4.2 of the HI-STORM 100 System FSAR describes the effect of damaged fuel assemblies on the external dose rates. This discussion indicates that the change in dose rate associated with the storage of damaged fuel assemblies is not significant. Based on that analysis and the reasonable expectation that there will be few damaged fuel assemblies stored in the Diablo Canyon ISFSI, a specific evaluation of damaged fuel assemblies was not performed. Rather, all assemblies in all casks were assumed to be intact at the design basis burnup and cooling times.

7.2.1.2 Fuel-Gamma Source

Tables 7.2-1A and 7.2-1B and Tables 7.2-2A and 7.2-2B present the gamma source terms that were used for the active fuel portion of the design basis assemblies for the overpack and transfer cask analyses, respectively. The source is presented in both MeV/sec and photons/sec for an energy range of 0.45 MeV to 3.0 MeV. Section 5.2.1 of the HI-STORM 100 System FSAR provides the justification that only photons in this energy range need to be considered in the dose evaluation. The HI-STORM 100 System FSAR states: “Photons with energies below 0.45 MeV are too weak to penetrate the overpack or transfer cask, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose.”

As mentioned above, the cooling time was varied from 5 to 20 years for the HI-STORM analysis to account for residency time on the ISFSI storage pad as the casks are assumed to be deployed in annual, 8-cask increments. In order to minimize the volume of data presented, Table 7.2-1 only presents the source term for the odd-year cooling times beginning at 5 years and ending at 15 years. This approach is also used in presenting the other source terms described below.

7.2.1.3 Fuel-Neutron Source

Table 7.2-3A and 7.2-3B and Tables 7.2-4A and 7.2-4B present the neutron source term used for the active fuel portion of the design-basis fuel assemblies for the overpack and transfer cask analyses, respectively. The neutron source is presented in neutrons/sec. Section 5.2.2 of the HI-STORM 100 System FSAR provides additional discussion on the calculation of the neutron source.

The neutron source term increases as the ^{235}U enrichment decreases for the same burnup and cooling time. Therefore, as discussed earlier in this section, a bounding low enrichment was chosen for the source term calculations. The neutron source strength also varies with burnup, by the power of 4.2 (Reference 1). Since this relationship is nonlinear and since burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.11 of the HI-STORM 100 System FSAR was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnup listed in Table 2.1.11 of the HI-STORM 100 System FSAR for the PWR fuel is 1.105. Using the power of 4.2 relationship results in a 37.6 percent ($1.105^{4.2}/1.105$) increase in the neutron source strength in the peak nodes and the total neutron source strength listed in Tables 7.2-3 and 7.2-4 increases by 15.6 percent. This increase in neutron source term is not reflected in the data presented in Tables 7.2-3 and 7.2-4, but is accounted for in the shielding analysis.

7.2.1.4 Nonfuel-Hardware Source

As mentioned above, the nonfuel hardware of a fuel assembly (for example, steel and inconel in the end fittings) activate during in-core operations to produce a radiation source. The primary radiation from these portions of the fuel assembly is ^{60}Co activity. Radiation from other isotopes within the steel and inconel has a negligible impact on the radiation dose rate compared with the ^{60}Co activity. Therefore, ^{60}Co was the only isotope considered in the analysis. The method used to calculate the activity in the nonfueled regions of the assembly is fully described in Section 5.2.1 of the HI-STORM 100 System FSAR. The ^{59}Co impurity level assumed in the steel and inconel of the fuel assembly was 1.0 g/kg or 1000 ppm. It was also assumed for this analysis that the fuel assemblies contained nonzircaloy grid spacers with a ^{59}Co impurity level of 1.0 g/kg. This assumption also conservatively bounds nonzircaloy fuel clips, which are present on a limited number of fuel assemblies. The HI-STORM 100 System FSAR (Chapter 8) discusses how this ^{59}Co impurity level value is conservative relative to fuel manufactured since the late 1980s.

Tables 7.2-5A and 7.2-5B and Tables 7.2-6A and 7.2-6B list the ^{60}Co source that was used in the nonfuel portions of the fuel assemblies for the overpack and transfer cask analyses, respectively. Tables 5.2.1 and 5.3.1 of the HI-STORM 100 System FSAR describe the mass and dimensions of these nonfuel portions of the fuel assembly. The HI-STORM 100 System FSAR includes burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), and rod cluster control assemblies (RCCAs) in the authorized contents of the HI-STORM 100 System. Since the DCPP fuel inventory includes assemblies containing all of these devices in some quantity, they were considered in the analysis. The HI-STORM 100 System FSAR describes the design-basis BPRA, RCCA, and TPD. The results demonstrate that the design-basis BPRA results in the highest dose rates compared to the TPD and RCCA. This is because the BPRA and TPD are very similar with the exception that the BPRA has an activated portion within the active fuel region. Since the RCCAs are limited to a quantity of four per cask in the center four locations, their contribution to the external dose rate is negligible compared to that of the BPRAs, which can be stored in any position. Therefore, only the BPRAs were considered in this analysis. As described above, the only isotope of concern in the activation of the BPRA is ^{60}Co. Consistent with the analysis in the HI-STORM 100 System FSAR the ^{59}Co impurity level was assumed to be 0.8 g/kg or 800 ppm in stainless steel and 4.7 g/kg or 4700 ppm in inconel. Table 7.2-7 provides the source term that was calculated for the BPRAs. This source was calculated using the design basis BPRA from the HI-STORM 100 System FSAR . An associated burnup of 40,000 MWD/MTU and a cooling time of 13 years were used for the BPRA. This burnup and cooling time bounds the current inventory of BPRAs at DCPP. DCPP has stopped using BPRAs and TPDs. Therefore, the number of these devices in the SFP is not increasing. However, for conservatism, it was assumed that all overpacks were filled with design-basis BPRAs. In the calculation of the dose rate from the ISFSI storage pads, the source shown in Table 7.2-7 was decayed (similar to the neutron and gamma source) to credit the additional cooling time arising from the assumption of eight casks per year being loaded and deployed at the ISFSI storage pads.

Neutron source assemblies (NSAs) are used in reactors for startup. During in-core operations, the stainless steel and Inconel portions of the NSAs become activated, producing a significant amount of Co-60. Using the masses of steel and Inconel for the NSAs it was determined that the total activation of a primary or secondary source is bound by the total activation of a BPRA (see Table 5.2.31 of Reference 11). Therefore, storage of NSAs is acceptable and a detailed dose rate analysis using the gamma source from activated NSAs is not performed.

Antimony-beryllium sources are used as secondary (regenerative) neutron sources in reactor cores. The Sb-Be source produces neutrons from a gamma-n reaction in the beryllium, where the gamma originates from the decay of neutron-activated antimony. The very short half-life of ^{124}Sb, 60.2 days, however results in a complete decay of the initial amount generated in the reactor within a few years after removal from the reactor. The production of neutrons by the Sb-Be source through regeneration in the MPC is orders of magnitude lower than the design-basis fuel assemblies. Therefore Sb-Be sources do not contribute to the total neutron source in the MPC.

Primary neutron sources (californium, americium-beryllium, plutonium-beryllium and polonium-beryllium) are usually placed in the reactor with a source-strength on the order of 5E+08 n/s. This source strength is similar to, but not greater than, the maximum design-basis fuel assembly source strength listed in Tables 5.2.15 and 5.2.16 of Reference 11.

By the time NSAs are stored in the MPC, the primary neutron sources will have been decaying for many years since they were first inserted into the reactor (typically greater than 10 years). For the ^{252}Cf source, with a half-life of 2.64 years, this means a significant reduction in the source intensity; while the ^{210}Po-Be source, with a half-life of 138 days, is virtually eliminated. The ^{238}Pu-Be and ^{241}Am-Be sources, however, have a significantly longer half-life, 87.4 years and 433 years, respectively. As a result, their source intensity does not decrease significantly before storage in the MPC. Since the ^{238}Pu-Be and ^{241}Am-Be sources may have a source intensity similar to a design-basis fuel assembly when they are stored in the MPC, only a single NSA is permitted for storage in the MPC. Because storage of a single NSA would not significantly increase the total neutron source in an MPC, storage of NSAs is acceptable and detailed dose rate analysis of the neutron source from NSAs is not performed.

For ease of implementation, the restriction concerning the number of NSAs is being applied to all types of NSAs. In addition, conservatively NSAs are required to be stored in the inner region of the MPC basket as specified in Section 10.2.

Instrument tube tie rods (ITTRs), which are installed after core discharge and do not contain radioactive materials, may also be stored in the assembly. ITTRs are authorized for unrestricted storage in an MPC.

7.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

Loading of spent fuel into the MPC in the transfer cask is performed under water in the SFP cask loading pit, which prevents the spread of effluent radioactivity during fuel loading. The MPC is sealed and dried within the FHB/AB allowing the liquid and gaseous waste released from the MPC during the draining and drying to be processed by the appropriate DCPP systems. Therefore, no airborne releases to the environment from the spent nuclear fuel assemblies are expected to occur during loading and handling operations.

The MPC, which provides the confinement boundary for the HI-STORM 100 System, is a welded pressure vessel and has no bolted closure or mechanical seals. Chapter 3 of the HI-STORM 100 System FSAR demonstrates that all confinement boundary components are maintained within Code-allowable stress limits under all design-basis normal, off-normal, and accident conditions. The all-welded construction of the MPC in conjunction with the extensive inspections and testing performed during closing operations ensures that no release of radioactive effluents will occur from the HI-STORM 100 System.

The above discussion notwithstanding, an analysis has been performed to calculate the dose to an individual at the Diablo Canyon site boundary due to an effluent release based on the Section 10.2 limit for leakage of 5.0 x 10^{-6} atm-cm^3/sec under the conditions of the helium leak rate test. This calculation is based on the guidance of NUREG-1536 (Reference 5), ISG-5 (Reference 6) and ISG-11 (Reference 7), as applicable, and is discussed in Sections 7.5 (for normal conditions), Section 8.1.3 (for off-normal conditions), and Section 8.2.7 (for accident conditions).

When the dose analysis was updated (Reference 10) to support loading of high-burnup fuel, the criteria for allowed leakage from the MPC was reduced to the leaktight criteria of ANSI N14.5-1997 and as such effluent releases do not need to be considered for casks tested to this criteria. The original effluent analysis is maintained for the 16 casks loaded to the original leakage criteria, however as noted in Reference 10, the off-site dose analysis for the original casks with the reduced source term and effluent release is bounded by the updated dose analysis with only direct dose. Therefore, the values for off-site dose assume all casks are loaded with the higher source term, and do not include a contribution from effluent release.

7.2.2.1 External Contamination Control

The external surface of the MPC is protected from contamination by preventing it from coming into contact with the SFP water. Prior to submergence in the SFP, an inflatable seal is installed at the top of the annulus formed between the MPC shell and the transfer cask cavity. This annulus is filled with clean, demineralized water and the seal is inflated. An annulus water overpressurization system is used to maintain the water behind the inflated seal at a slight positive pressure. This system, in the unlikely event of a leak in the inflated seal, will preclude the entry of contaminated water into the annulus. These steps ensure that the MPC surface is free of contamination that could become airborne during storage. Additionally, following fuel-loading operations and removal from the SFP, the MPC lid, the upper end of the MPC shell, and the exterior surfaces of the transfer cask are decontaminated, to the extent practicable, and then surveyed for any remaining, loose surface contamination.

7.2.2.2 Confinement Vessel Releasable Source Term

The inventory for isotopes other than ^{60}Co is calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system, as described in Chapter 5 of the HI-STORM 100 System FSAR. The isotopic inventory for PWR fuel in the MPC-32 was based on the design-basis fuel assembly with a burnup of 55,000 MWD/MTU, 5-years cooling time, and an enrichment of 4.0 wt percent ^{235}U. These assumed burnup and cooling times were chosen to conservatively bound the actual burnup and cooling times for all spent fuel currently at the DCPP site. This burnup is different from that used for the direct radiation source, because unlike the direct radiation source, where the dose rate decreases as the burnup and cooling time increase, the dose rate from effluent release is primarily driven by burnup and is not significantly affected by cooling time.

The enrichment chosen for the confinement evaluation, 4.0 wt percent ^{235}U, is a conservatively low enrichment for the burnup of 55,000 MWD/MTU. The dose to all organs, with the exception of the lung, and the whole body either increases or remains constant with decreasing enrichment. Therefore, a lower enrichment is generally conservative. The dose rate to the lung increases less than 5 percent for a 1 percent increase in enrichment. Section 7.5 presents the offsite dose due to a non-mechanistic normal effluent release. In that section, the dose rate to the lung is bounded by the dose rate to the bone and therefore the slight increase in dose rate for the lung that would be expected from a higher enrichment is not considered.

The 55,000 MWD/MTU burnup bounds the allowable burnups for the MPC-32 as specified in the Diablo Canyon ISFSI TS and Section 10.2. This burnup, though, does not bound all the allowable burnups for the MPC-24 or MPC-24E. However, the reduced fuel contained in an MPC-24 versus an MPC-32 offsets the slight increase in isotopic inventory associated with the slightly higher allowable burnups in the MPC-24. Therefore, the confinement analysis in Section 7.5 of an MPC-32 with a burnup of 55,000 MWD/MTU and a cooling time of 5 years is conservative.

All isotopes that contribute greater than 0.1 percent to the total curie inventory for the fuel assembly are considered in the evaluation as fines. This analysis also includes those actinides that contribute greater than 0.01 percent to the total curie inventory as the dose conversion factors for these isotopes are in general, greater than other isotopes (for example, isotopes of plutonium, americium, curium, and neptunium). A summary of the isotopes available for release is provided in Table 7.2-8.

7.2.2.3 Crud Radionuclides

The majority of the activity associated with crud is due to ^{60}Co (Reference 8). The inventory for ^{60}Co was determined by using the crud surface activity for PWR rods (140×10^{-6} Ci/cm^2) provided in NUREG/CR-6487, multiplied by the surface area per assembly (3×10^5 cm^2 for PWR fuel, also provided in NUREG/CR-6487). The source terms were then decay corrected 5 years using the basic radioactive decay equation:

$$A(t) = A_0 e^{-\lambda t}$$

where:

A(t)	=	activity at time t (Ci)
A_0	=	the initial activity (Ci)
λ	=˜	the $\ln2/t_{1/2}$ (where $t_{1/2}$ = 5.272 years for ^{60}Co (Reference 9))
t	=	the time in years (5 years)

A summary of the ^{60}Co inventory available for release is provided in Table 7.2-8.

7.2.3 REFERENCES

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7.3 RADIATION PROTECTION DESIGN FEATURES

7.3.1 STORAGE SYSTEM DESIGN FEATURES

The Diablo Canyon ISFSI is described in Chapters 1, 2, and 4. The HI-STORM 100 System dry storage casks will be stored on up to seven concrete pads. Each pad contains a 4-by-5 array of casks. Therefore, the ISFSI has a total capacity of 140 casks (138 plus 2 spare locations). Figures 2.1-2 and 4.1-1 illustrate the ISFSI location and pad layout. The casks are positioned on a 17 ft, center-to-center pitch and the pads are positioned such that the pitch between casks on adjacent pads is also 17 ft. As discussed in Section 4.1, the restricted area fence surrounding the ISFSI is positioned to ensure that the dose rate at the fence is below 2 mrem/hr, in accordance with the requirements of 10 CFR 20 for unrestricted areas.

The ISFSI and dry storage system has a number of design and administrative control features that ensure that radiation exposures are ALARA.

- There are no radioactive systems at the ISFSI storage pads other than the overpacks containing MPCs.
- The fuel is stored dry inside the MPC, so that no radioactive liquid is available for leakage.
- The MPCs are loaded, welded, and the upper lid decontaminated in the DCPP FHB/AB prior to being moved to the CTF located near the ISFSI storage pads.
- The overpacks are loaded and the lids installed prior to movement from the CTF to the ISFSI pads.
- Fuel is not removed from the MPCs at either the ISFSI storage pads or the CTF. Unloading of the fuel from the MPC, if necessary, would only occur in the SFP in the FHB/AB.
- The MPCs are heavily shielded by the overpack.
- A locked restricted area fence surrounds the ISFSI storage pads to prevent unauthorized access.
- The ISFSI storage area is typically not occupied.
- Lastly, the MPC design includes a 9.5-inch thick steel lid for shielding of workers.

The HI-STORM 100 System FSAR (Reference 1) describes the transfer cask and overpack in detail. The design features of the HI-STORM 100 System and CTF that ensure radiation exposures are ALARA follow:

- The overpack has a large concrete body encased in steel. The concrete is over 2-ft thick and the steel on the inside and outside of the concrete is each more than 0.5-inch thick. The concrete provides both neutron and gamma radiation shielding while the steel provides predominantly gamma radiation shielding.

- The use of the short overpack eliminates the need for the upper vent duct shield inserts during MPC loading operations. This is accomplished by incorporating the upper vent ducts into the lid.

- The cask transporter places the overpack below ground prior to the MPC transfer. This minimizes the time involved in loading the overpack by significantly reducing the lift height of the transfer cask above the overpack. This contributes to reduced dose rates during MPC transfer operations.

- The HI-STORM 100 System and the CTF have been designed for ease of operation to minimize the duration of the operational sequences.

- In order to minimize dose to personnel consistent with the ALARA philosophy, procedures will be reviewed and dry runs will be performed prior to loading the first cask.

7.3.2 SHIELDING

The design of the HI-STORM 100 System, including the transfer cask, as it relates to the shielding evaluation, is described in Section 5.3 of the HI-STORM 100 System FSAR (Reference 1). Summary design targets are given in Table 3.4-2. Besides the overpack and transfer cask, no other radiation shielding features are required for the Diablo Canyon ISFSI. However, due to the choice of the ISFSI storage pad location, which is excavated into the side of a hill, there is a partial natural earth berm located around three sides of the ISFSI storage pads. The terrain around the Diablo Canyon ISFSI storage pads is naturally hilly, which will also provide additional radiation shielding. Conservatively, the analysis documented in this FSAR does not take credit for any additional radiation shielding, which would be provided by the surrounding terrain. Rather, the calculations conservatively assume that the ISFSI storage pads are located on flat ground. The details of the calculations are described in Sections 7.4 and 7.5.

The HI-STORM 100SA overpack design is used at the Diablo Canyon ISFSI. The overpack anchorage hardware has no significant impact on the shielding evaluation. Therefore, the shielding analyses and models emulate the HI-STORM 100S overpack and are applicable to the HI-STORM 100SA overpacks used at the Diablo Canyon ISFSI.

7.3.2.1 Surface and One Meter Dose Rates

As described in Section 7.2, the design-basis MPC for the HI-STORM analysis is the MPC-32. In the original analysis a burnup and cooling time of 32,500 MWD/MTU and 5 years, respectively, was used for all fuel assemblies in the MPC. When the analysis was updated for high burnup fuel, a burnup and cooling time of 69,000 MWD/MTU and 5 years, respectively, was used for all fuel assemblies in the MPC. The design-basis MPC for the transfer cask analysis is the MPC-24 with a burnup and cooling time of 55,000 MWD/MTU and 12 years in the original analysis, and an MPC-32 with a burnup and cooling time of 75,000 MWD/MTU and 5 years in the high burnup analysis, respectively, for all fuel assemblies in the MPC. These MPCs and burnup/cooling time combinations were chosen to bound all models of MPC in each case, as noted in the associated HI-STORM FSAR (References 1 and 6). Figures 7.3-1 and 7.3-2 show the overpack and the transfer cask with dose rate locations marked. These are the same dose locations for which values were reported in the HI-STORM 100 System FSAR. Tables 7.3-1A and 7.3-1B, 7.3-2A and 7.3-2B present the surface and 1-meter dose rates for the overpack and the transfer cask loaded with the MPC-32 and MPC-24, respectively, and design basis fuel, including BPRAs. The dose from the individual source components (neutron, photon, and cobalt) is explicitly listed.

7.3.2.2 Dose Versus Distance

The dose rate versus distance from both an overpack and the Diablo Canyon ISFSI were calculated using the Monte Carlo N-Particle (MCNP) transport code (Reference 3). Figure 7.3-3 provides a pictorial representation of the ISFSI with all seven storage pads completely filled with loaded overpacks. The cooling time of the fuel assemblies assumed in the shielding analysis is superimposed on the cask locations in Figure 7.3-3. Based on the storage capacity of the ISFSI (138 plus 2 spare locations), it is not practical to try to model the entire ISFSI in MCNP or any other computer code. Therefore, a methodology similar to that described in Section 5.4 of the HI-STORM 100 System FSAR was used in the calculation of the dose rate versus distance from the ISFSI. The dose rate versus distance was calculated first for a single overpack. Then numerous MCNP calculations, using relatively small models, were performed to develop ratios for the dose rate contribution from casks situated behind other casks. These ratios were used in conjunction with the dose rate versus distance from a single overpack to estimate the dose rate from the entire ISFSI storage area.

The dose rate from the radiation source was separated into two components. For the purposes of this discussion, the first is referred to as the top-dose. This is the dose rate from radiation that leaves the top of the overpacks. The second component is referred to as side-dose. This is the dose rate from radiation that leaves the sides of the overpacks. In both cases, top-dose and side-dose, in-air scattering of radiation (skyshine) were accounted for in the dose calculations.

In calculating the dose rate from the entire ISFSI storage area, the cask array geometry impacted each of the dose components (top and side) in a different fashion. The total top-dose rate was a summation of the top-dose rates from all 140 casks where the actual distance from the dose location to the individual cask was accounted for.

The total side-dose rate was a summation of the side-dose rates from all 140 casks where the distances within the facility and the self-shielding of one row of casks to another row were accounted for. Since the side-dose rate is from particles leaving the side of the overpack, this dose contribution is greatly reduced if the cask is situated behind another cask. The front cask blocks some, but not all of the radiation from the back cask from reaching the site-boundary. The fraction of radiation blocked was therefore calculated with MCNP, as mentioned above, and used in the determination of the total side-dose.

Dose locations along the long side of the cask array are facing 28 casks directly, that is, without being shielded by other casks. Dose locations along the short side of the array only face five casks directly. Dose rates at dose points along the long side of the array will, therefore, always be higher than dose rates at dose points along the short side of the array. As a bounding approach, all dose rates from the ISFSI storage area reported in this chapter are calculated perpendicular to the long side of the array, regardless of the actual orientation of the dose location relative to the cask array. The results of the dose rate calculations are discussed in Sections 7.4 and 7.5.

As mentioned earlier, the models assumed a flat terrain surrounding the overpack and the ISFSI storage area. The MCNP models consisted of the overpack surrounded by 1,050 meters of air in the radial direction and 700 meters of air in altitude. The cask was assumed to be sitting on an infinite slab of soil. The dose rate versus distance from a single overpack was calculated for the top and side of the overpack separately. Tables 7.3-4A and 7.3-4B show the dose rate versus distance from a single overpack for the design basis burnup and cooling time. The dose rate due to radiation exiting the top and radiation exiting the side of the overpack are explicitly listed in addition to the total dose rate.

7.3.2.3 ISFSI Loading Plan

As mentioned in Section 7.2, it was assumed for the purpose of the dose rate analysis that eight overpacks are loaded per year every year until the ISFSI storage pads are completely filled. Credit for source-strength reduction was taken for the additional cooling time that occurs as a result of this loading plan. At a rate of 8 casks per year, it takes 17.5 years to fill the ISFSI to capacity for a total minimum cooling time after core discharge of 22.5 years for the first casks deployed. However, the oldest fuel in the casks in the ISFSI was conservatively assumed to be 20 years old. No credit was taken for additional cooling from 20 to 22.5 years. Note that this approach also conservatively assumes that all fuel is loaded in the HI-STORM 100 System casks at 5-years cooling time, which is the shortest cooling time allowed by the Diablo Canyon ISFSI TS and

Section 10.2. Since the fuel in the casks on the ISFSI pads have different cooling times after the ISFSI is filled, the position of the casks relative to the dose locations is important.

Section 4.1 states that up to 7 ISFSI pads will be constructed and each pad will contain a 4-by-5 array of casks. The pads will be constructed beginning at the east end of the ISFSI and progressing west, as needed. This loading plan was credited in the shielding analysis. However, it was conservatively assumed that the casks with the "youngest" fuel were positioned on the pads closest to the dose locations. Figure 7.3-3 shows the ISFSI in its final configuration after all seven storage pads have been filled. The age of the fuel in the casks assumed for the analysis is shown in the center of the circle representing a cask. Since it is assumed that 8 casks are loaded per year and credit is taken for the additional cooling time up to 20 years, the age of the fuel in the casks on Pad 1 (the first pad to be used) is assumed to be 20 years. The age of the fuel in the casks on the last pad loaded, Pad 7, is assumed to be 5 to 7 years. Since the highest dose rate from the ISFSI will occur after the ISFSI is completely loaded, this was the only configuration analyzed. As discussed earlier, the dose rate was conservatively calculated perpendicular to the long side of the ISFSI. However, because of the loading pattern of the casks, the location of highest dose rate is not in the center of the ISFSI. Calculations determined that the highest dose rate occurs at approximately the center of Pad 6. Therefore, the dose versus distance calculations from the ISFSI were conservatively performed for distances perpendicular to the center of Pad 6.

When the analysis was updated for implementation of storage of high burnup fuel, all casks were assumed loaded with the new higher source term. No credit was taken for lower design/actual source term from the initial 16 casks.

7.3.3 VENTILATION

10 CFR 72.122(h)(3) requires that ventilation systems and offgas systems be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal and off-normal conditions. However, as discussed in Section 3.3.1.2, the HI-STORM 100 System is designed to prevent the release of radioactive materials and gases during normal and off-normal conditions. Thus, there are no offgas systems required once the spent fuel is enclosed in the welded MPCs.

Nonetheless, Section 7.5 provides an evaluation of the offsite dose consequences from the hypothetical leakage of all loaded MPC-32s in the ISFSI under normal and off-normal conditions. The hypothetical leakage of a single, loaded MPC-32 under accident conditions, where the cladding of 100 percent of the fuel rods is postulated to have ruptured, is described in Section 8.2.7.

7.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

Permanent area radiation and airborne radioactivity monitors are not needed at the Diablo Canyon ISFSI since the storage system is passive. Temporary, hand-held radiation protection instruments and self-reading dosimeters will be used during transfer operations at the CTF and routine maintenance at the ISFSI storage area. Thermoluminescent dosimeters will be used to monitor, record, and trend area doses at appropriate intervals in all four directions around the ISFSI restricted area fence. Neutron radiation detection devices may also be used if deemed necessary by the DCPP radiation protection organization.

During fuel loading, existing SFP monitors 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. An area radiation monitoring system is provided for personnel protection and general surveillance of the SFP area (Reference 4, Section 11.4.2.1.4). Continuous monitoring, recorded readouts, and high radiation level alarms are available in the control room, plus local audible and visual indicators are in place to alert personnel of high radiation conditions during fuel movement in the FHB/AB. In addition to the monitoring equipment, radiation protection coverage with hand-held radiation protection instruments and self-reading dosimetry for fuel movement evolutions is provided, which is standard practice for these activities.

7.3.5 REFERENCES

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7.4 ESTIMATED ONSITE COLLECTIVE DOSE ASSESSMENTS

The results presented in this section are based on the analysis of the overpack and the transfer cask using original analysis design basis fuel, including BPRAs (bounding nonfuel hardware). The discussion in Section 7.2 states that the transfer cask was analyzed with the MPC-24 and the overpack was analyzed with the MPC-32 because these were the bounding MPCs for those overpacks. Consistent with that approach, the analysis presented in this section assumed the transfer cask was loaded with an MPC-24 with a design basis burnup and cooling time of 55,000 MWD/MTU and 12 years, respectively. This analysis also conservatively assumed that the overpack was loaded with an MPC 32 with a design basis burnup and cooling time of 32,500 MWD/MTU and 5 years, respectively (Reference 1).

These values were not revised for the analysis performed in support of loading high burnup fuel (HBF). Since the actual burnup/cooling time allowed to be loaded by Section 10.2 is not appreciably different (burnup < 20% higher for same cooling time) from the values used in the original analysis, and the actual loading experience has shown that actual loading is done for less than 30% of the estimated dose, the following values will continue to be used to estimate the occupational exposure for loading and operations of the ISFSI at Diablo Canyon.

The estimated occupational exposure during overpack loading operations is approximately 2.1 rem. Refer to Holtec Report HI-2002563, Revision 10 (Reference 1).

The estimated occupational exposure during overpack unloading operations is approximately 1.5 rem (Reference 1).

The list of operation steps is also provided in Reference 1, Appendix K. Numerous operations have been lumped together for ease of presentation. The duration of the operation and the time the personnel are located in the higher dose rate areas are based on industry experience with the Holtec HI-STAR and HI-STORM casks and casks from other vendors. The dose rates used for this analysis are conservatively estimated using design-basis fuel. Diablo Canyon radiation protection personnel assure that the appropriate radiation monitoring is performed and that all operations are performed in a manner consistent with ALARA.

The occupational exposures during overpack loading and unloading operations are conservatively estimated. ALARA practices take advantage of experience in loading HI-STORM 100 Systems at Diablo Canyon as well as at other utilities. Based on the experience gained and the lessons learned, it is expected that the dose rates from loading an overpack will be less than those listed here (that is, fewer activities, strategically placed shielding and shorter durations).

The estimated total annual per person occupational exposure as a result of daily ISFSI walkdowns, occasional maintenance repairs, and construction of additional ISFSI pads are 1.8 rem, 0.8 rem, and 2.9 rem, respectively (Reference 1). The dose associated with the clearing of debris from a blocked ventilation duct is presented in Sections 8.1.4 and 8.2.15.

The daily walkdown of the ISFSI requires a person to walk the full length of the ISFSI outside each pad of casks and between each row of casks. This walkdown is to look for obstructions that may be blocking the air vents of the overpack. It was assumed, based on a walking speed of 2 miles/hour, that it would take a person 20 minutes to perform the walk-down at the completion of the ISFSI when all pads are filled with overpacks. This results in a total occupancy time of 122 hours per year. A dose rate of 15 mrem/hr for the walk-downs is conservatively based on the 1-meter dose rates, times 4 casks.

The doses for the repair operations assume 1 repair operation per month of 1-hour duration with 2 people performing the operation. A dose rate of 65 mrem/hr for repair operations is conservatively based on an infinite array of casks.

The dose during construction of additional storage pads was calculated for the construction of Pad 7. It was assumed that the previous six pads were completely filled. Doses estimated for the construction of Pad 7 bound the construction of any other pad. The dose rate was conservatively estimated at the center of Pad 7 with no credit for temporary shielding. It was assumed that construction would take 3 months at 40 hours per week in the dose field. The number of personnel and dose rate were assumed to be 15 and 6 mrem/hr, respectively.

The estimated dose rate at the assumed location for the restricted area fence, the makeup water facility (the nearest normally occupied location), and the power plant are 1.9 mrem/hr, 0.51 mrem/hr, and 0.022 mrem/hr, respectively. The occupancy time was assumed to be 2,080 hours, which is the equivalent of a 40-hour workweek for 52 weeks per year. Also, the dose rates at these locations were conservatively calculated perpendicular to the long side of the storage array. The dose rate at the restricted area fence for the assumed location will be below 2 mrem/hr. Also, the dose rates in the normally-occupied locations, due to the ISFSI, are well below the 10 CFR 20 limits for monitored radiation workers. The workers at the makeup water facility may have to become monitored workers as the storage pad approaches the full capacity. Compliance with 10 CFR 20 for these and other workers is assured via personnel dose monitoring in accordance with the DCPP Radiation Protection Program (Reference 1).

The dose rates for ISFSI walkdowns, occasional maintenance repairs, and construction of additional ISFSI pads and at the restricted area fence, the makeup water facility, and the power plant demonstrate that the estimated occupational exposures from the Diablo Canyon ISFSI meet the regulatory requirements of 10 CFR 20. The actual doses from the ISFSI are expected to be considerably less than the above conservatively estimated values. This information is included in the annual report to regulatory agencies.

7.4.1 REFERENCES

1. Holtec International Report No. HI-2002563, “Dose Evaluation for the ISFSI at Diablo Canyon Power Plant,” Revision 10.

7.5 OFFSITE COLLECTIVE DOSE

The annual offsite dose is calculated for both direct radiation (neutrons and gammas) and from radionuclide releases from the MPC (Reference 8). Since the MPC is welded and designed to maintain confinement integrity under all normal, off-normal, and accident conditions of storage, there will not be any release of radionuclides during normal operation. Nonetheless, an analysis of the offsite dose consequences from a nonmechanistic confinement boundary leak from the ISFSI was calculated for normal, off-normal, and accident conditions. This section addresses doses for normal conditions. Off-normal and accident analyses are provided in Sections 8.1.3 and 8.2.7, respectively. The direct radiation dose from the ISFSI is the same for normal and off-normal conditions.

Since the loading of the MPC into the overpack occurs outside the FHB/AB at the CTF, the offsite dose due to these activities was also calculated and included in the total annual dose estimate.

The controlled area boundary is located 1,400 ft (427 m) from the ISFSI. However, the nearest resident is located 1.5 mi (7,920 ft or 2,414 m) from the ISFSI. Therefore, consistent with ISG-13 (Reference 1), the occupancy time at the controlled area boundary for the dose calculation was assumed to be 2,080 hr based on a 40-hr work week and 52 weeks per yr while the occupancy time at the nearest resident location was assumed to be 8,760 hr (24 hr per day 365 days per yr).

7.5.1 DIRECT RADIATION DOSE RATES

Table 7.5-1 presents the dose rate and annual doses at the site boundary and the nearest residence from direct radiation from the Diablo Canyon ISFSI after it is completely filled with 140 overpacks loaded with the MPC-32 at design-basis burnup and cooling times. As described in Section 7.3.2.3, these dose rates and doses were calculated at distances that were perpendicular to the long side of the ISFSI and it was assumed that eight overpacks were loaded per year.

7.5.2 DOSE RATES FROM NORMAL OPERATION EFFLUENT RELEASES

The source term used for the offsite dose assessment from the effluent release from the MPC is discussed in Section 7.2.2. The dose assessment from effluent release was calculated for normal conditions. Effluent doses for off-normal operations are discussed in Section 8.1.3. Effluent doses for an accident condition are discussed in Section 8.2.7.

As noted in Section 7.2.2, when the dose analysis was updated to support the loading of high burnup fuel at the Diablo Canyon ISFSI, the need to consider effluent releases under conditions of normal storage was eliminated. Therefore, the remainder of this section is presented to document the historical licensing basis of the initial 16 casks only.

7.5.2.1 Release of MPC Contents Under Normal Occurrences

The MPC is designed to maintain confinement boundary integrity under all normal, off-normal, and accident conditions of storage. Nevertheless, for the original dose analysis, a hypothetical, non-mechanistic confinement boundary leak was evaluated in the effluent dose analysis. For normal conditions, it was assumed that 2.5 percent of the total source term of each assembly is available for release to the MPC cavity. This was based on the assumption, from ISG-5 (Reference 2), that 1 percent of the fuel rods have ruptured. In addition to the 1 percent, it was assumed, consistent with ISG-11 (Reference 3), that an additional 3 percent of fuel rods had cladding oxide thicknesses greater than 70 micrometers and therefore had 50 percent of the source term in these rods available for release. The spent fuel is stored in a manner such that the spent fuel cladding is protected during storage against degradation that could lead to fuel cladding ruptures. The MPC cavity is filled with the inert gas helium after the MPC has been evacuated of air and moisture that might produce long-term degradation of the spent fuel cladding. The HI-STORM 100 System is additionally designed to provide for long-term heat removal to ensure that the fuel is maintained at temperatures below those at which cladding degradation occurs. It is therefore highly unlikely that a spent fuel assembly with intact fuel cladding will undergo cladding failure during storage, and the assumption that 2.5 percent of the source term is available for release is conservative.

The assumption that 10 percent of the fuel rods have ruptured was incorporated into the postulated pressure increase within the MPC cavity to determine a bounding pressure of the MPC cavity for effluent release calculations for the normal and off-normal cases. This pressure, combined with the maximum MPC cavity temperature was used to determine a postulated leakage rate. This leakage rate was based on an assumed leakage of 5.0×10^{-6} atm-cm^3/sec during the helium leak rate test and was adjusted for the higher temperature and pressure during the off-normal condition to result in a calculated leak rate of 7.37×10^{-6} atm-cm^3/sec.

The radionuclide release fractions, which account for the radionuclides trapped in the fuel matrix and radionuclides that exist in a chemical or physical form that is not releasable to the environment, were based on ISG-5 and are presented in Table 7.2-8. Additionally, only 10 percent of the fines released to the MPC cavity were assumed to remain airborne long enough to be available for release from the cask MPC (Reference 4). It was conservatively assumed that 100 percent of the volatiles, crud, and gases remain airborne and available for release. The release rate for each radionuclide was calculated by multiplying the quantity of radionuclides available for release in the MPC cavity by the leakage rate calculated above, divided by the MPC cavity volume.

7.5.2.2 Effluent Dose Calculations for Normal Conditions

The nearest distance from the ISFSI to the DCPP site boundary is 1,400 ft. A χ/Q value of 3.44×10^{-6} sec/m^3 (Reference 5) at the site boundary was used for this analysis. This χ/Q value is the highest χ/Q in any direction and is based on duration of an entire year.

The dose conversion factors for internal doses due to inhalation and submersion in a radioactive plume were obtained from the EPA Federal Guidance Report No. 11 (Reference 6) and EPA Federal Guidance Report No. 12 (Reference 7), respectively. An adult breathing rate of 3.3 x 10^{-4} m^3/sec was assumed (Reference 2). For site boundary dose, an annual occupancy of 2,080 hr was assumed. For the nearest resident, full-time occupancy was assumed (8,760 hr).
The annual dose equivalent for the whole body, thyroid, and other critical organs to an individual at the DCPP site boundary as a result of a non-mechanistic normal effluent release were calculated for an ISFSI containing 140 overpacks, each loaded with an MPC-32. Table 7.5-2 summarizes the dose results for normal conditions. As can be concluded from Table 7.5-2, the estimated doses are a fraction of the limits specified in 10 CFR 72.104(a) for normal operations.

7.5.3 OFFSITE DOSE FROM OVERPACK LOADING OPERATIONS

The transfer of the MPC from the transfer cask to the overpack occurs outside the FHB/AB at the CTF. As a result, the impact of this operation on the offsite dose was considered. The only condition that needs to be considered in this analysis is the condition of the MPC inside the transfer cask when outside the FHB/AB. Table 7.5-3 presents the results of this analysis.

7.5.4 TOTAL OFFSITE COLLECTIVE DOSE

Table 7.5-4 presents the annual dose at the site boundary and for the nearest resident from the combined dose rates from direct radiation and non-mechanistic effluent release for normal ISFSI operations and off-normal operations. The dose rates from other uranium fuel cycle operations (that is, DCPP) are also shown in this table to demonstrate compliance with 10 CFR 72.104. Table 7.5-4 demonstrates that the Diablo Canyon ISFSI will meet the 10 CFR 72.104 regulatory requirements. However, ultimate compliance with the regulations is demonstrated through the DCPP environmental monitoring program.

The actual dose from the ISFSI will be considerably less than the conservatively estimated values in Table 7.5-4. The following are some of the conservative assumptions used in the calculating the dose rates presented.

- The design basis assembly and design basis burnup and cooling time were conservatively chosen.

- All fuel assemblies in the MPC are assumed to be identical with the design basis burnup and cooling time.

- BPRAs are assumed to be present in all fuel assemblies in all casks.

- The assumed ISFSI loading plan was conservatively chosen to result in the highest offsite dose rate.

- The dose rate was calculated at the most conservative location around the ISFSI.

7.5.5 REFERENCES

1. Real Individual, USNRC, Interim Staff Guidance Document-13, Revision 0, June 2000.

2. Normal, Off-Normal and Hypothetical Dose Estimate Calculations, USNRC, Interim Staff Guidance Document-5, Revision 1, June 1999.

3. Transportation and Storage of Spent Fuel Having Burnups in Excess of 45 GWD/MTU, USNRC, Interim Staff Guidance Document-11, Revision 1, May 2000.

4. Y.R. Rashid, et al, An Estimate of the Contribution of Spent Fuel Products to the Releasable source Term in Spent Fuel Transport Casks, SAND88-2778C, Sandia National Laboratories, 1988.

5. 1999 Annual Radioactive Effluent Release Report, PG&E Letter DCL-00-061, April 28, 2000.

6. Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, US EPA, Federal Guidance Report No. 11, DE89-011065, 1988.

7. External Exposure to Radionuclides in Air, Water, and Soil, US EPA, Federal Guidance Report No. 12, EPA 402-R-93-081, 1993.

8. Holtec International Report No. HI-2002563, “Dose Evaluation for the ISFSI at Diablo Canyon Power Plant,” Revision 10.

7.6 HEALTH PHYSICS PROGRAM

7.6.1 ORGANIZATION

The health physics program, which is described in the DCPP FSAR Update, Section 12.3, is considered sufficient for ISFSI activities. The Manager, Radiation Protection, is responsible for health physics activities related to ISFSI operations for the life of the facility, including all decontamination and decommissioning activities. The radiation protection manager is independent of the operations manager.

7.6.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

DCPP health physics program objectives, equipment, instrumentation, and facilities are described in the DCPP FSAR Update, Section 12.3.2. Except for access control to the ISFSI, many of the same facilities identified in the DCPP FSAR Update will be used for ISFSI operations and surveys. Once the storage site is operational, entrance to and work within the ISFSI protected area will be controlled by radiation protection and security personnel. Radiation work permits will be required in accordance with applicable DCPP procedures.

Available equipment and instrumentation includes personal monitoring equipment, portable radiation measuring instruments, portable air sampling equipment, facilities for internal radiation monitoring, count room equipment, personnel protective equipment, and decontamination equipment and facilities.

7.6.3 POLICIES AND PROCEDURES

The health physics program is carried out 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 revised procedures will help to maintain exposure ALARA to personnel consistent with operating the ISFSI in a safe, reliable, and efficient manner and will ensure compliance with all applicable regulations and PG&E policies pertaining to radiation protection and release of radioactive materials.

The operation and use of radiation monitoring instrumentation at the Diablo Canyon ISFSI, including personnel monitoring equipment and measurement and sampling techniques, will be described in written procedures.

7.7 ENVIRONMENTAL MONITORING PROGRAM

The DCPP radiological environmental monitoring program will also be used for the Diablo Canyon ISFSI. This program will be augmented to include additional thermoluminescent dosimeters. Since there are no effluents from the ISFSI, there will be no additional radiological effluent monitoring.

TABLE 7.2-1A

CALCULATED HI-STORM PWR GAMMA SOURCE PER ASSEMBLY FOR A BURNUP OF 69,000 MWD/MTU

Lower Energy	**Upper Energy**	**5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5E-01	7.0E-01	3.26E+15	5.67E+15
7.0E-01	1.0	1.23E+15	1.44E+15
1.0	1.5	2.69E+14	2.15E+14
1.5	2.0	1.41E+13	8.08E+12
2.0	2.5	7.56E+12	3.36E+12
2.5	3.0	3.56E+11	1.29E+11
Totals		4.78E+15	7.34E+15

NOTE:
Table values obtained from Reference 11, Table 5.2.4

TABLE 7.2-1B

CALCULATED HI-STORM PWR GAMMA SOURCE PER ASSEMBLY FOR A BURNUP OF 32,500 MWD/MTU

Lower Energy	**Upper Energy**	**5-Year Cooling**		**7-Year Cooling**		**9-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
4.5E-01	7.0E-01	1.47E+15	2.56E+15	1.17E+15	2.04E+15	1.02E+15	1.77E+15
7.0E-01	1.0	4.49E+14	5.28E+14	2.40E+14	2.83E+14	1.35E+14	1.59E+14
1.0	1.5	1.07E+14	8.53E+13	6.85E+13	5.48E+13	4.96E+13	3.97E+13
1.5	2.0	7.51E+12	4.29E+12	3.63E+12	2.07E+12	2.48E+12	1.42E+12
2.0	2.5	6.42E+12	2.86E+12	1.23E+12	5.46E+11	2.49E+11	1.11E+11
2.5	3.0	2.38E+11	8.67E+10	6.08E+10	2.21E+10	1.58E+10	5.73E+09
Totals		2.04E+15	3.18E+15	1.49E+15	2.38E+15	1.20E+15	1.97E+15
Lower Energy	**Upper Energy**	**11-Year Cooling**		**13-Year Cooling**		**15-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
4.5E-01	7.0E-01	9.20E+14	1.60E+15	8.53E+14	1.48E+15	8.02E+14	1.39E+15
7.0E-01	1.0	7.99E+13	9.40E+13	5.06E+13	5.95E+13	3.44E+13	4.05E+13
1.0	1.5	3.86E+13	3.08E+13	3.12E+13	2.50E+13	2.59E+13	2.07E+13
1.5	2.0	1.99E+12	1.14E+12	1.69E+12	9.67E+11	1.46E+12	8.36E+11
2.0	2.5	5.75E+10	2.55E+10	1.81E+10	8.05E+09	9.47E+09	4.21E+09
2.5	3.0	4.29E+09	1.56E+09	1.37E+09	4.99E+08	6.27E+08	2.28E+08
Totals		1.04E+15	1.73E+15	9.36E+14	1.57E+15	8.63E+14	1.46E+15

TABLE 7.2-2A

CALCULATED HI-TRAC PWR GAMMA SOURCE PER ASSEMBLY FOR A BURNUP OF 75,000 MWD/MTU

Lower Energy	**Upper Energy**	**5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5E-01	7.0E-01	3.55E+15	6.17E+15
7.0E-01	1.0	1.36E+15	1.60E+15
1.0	1.5	2.94E+14	2.35E+14
1.5	2.0	1.50E+13	8.59E+12
2.0	2.5	7.63E+12	3.39E+12
2.5	3.0	3.72E+11	1.35E+11
Totals		5.23E+15	8.02E+15

NOTE:
Table values obtained from Reference 11, Table 5.2.5

TABLE 7.2-2B

CALCULATED HI-TRAC PWR GAMMA SOURCE PER ASSEMBLY FOR A BURNUP OF 55,000 MWD/MTU

Lower Energy	**Upper Energy**	**12-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5E-01	7.0E-01	1.48E+15	2.58E+15
7.0E-01	1.0	1.30E+14	1.52E+14
1.0	1.5	7.07E+13	5.65E+13
1.5	2.0	3.64E+12	2.08E+12
2.0	2.5	4.08E+10	1.81E+10
2.5	3.0	4.01E+09	1.46E+09
Totals		1.69E+15	2.79E+15

TABLE 7.2-3A

CALCULATED HI-STORM PWR NEUTRON SOURCE PER ASSEMBLY FOR A BURNUP OF 69,000 MWD/MTU

Lower Energy (MeV)	**Upper Energy** (MeV)	**5-Year Cooling** (Neutrons/s)
1.0E-01	4.0E-01	5.31E+07
4.0E-01	9.0E-01	2.71E+08
9.0E-01	1.4	2.48E+08
1.4	1.85	1.82E+08
1.85	3.0	3.21E+08
3.0	6.43	2.92E+08
6.43	20.0	2.60E+07
Total		1.39E+09

NOTE:
Table values obtained from Reference 11, Table 5.2.15

TABLE 7.2-3B

CALCULATED HI-STORM PWR NEUTRON SOURCE PER ASSEMBLY FOR A BURNUP OF 32,500 MWD/MTU

Lower Energy (MeV)	**Upper Energy** (MeV)	**5-Year Cooling** (Neutrons/s)	**7-Year Cooling** (Neutrons/s)	**9-Year Cooling** (Neutrons/s)	**11-Year Cooling** (Neutrons/s)	**13-Year Cooling** (Neutrons/s)	**15-Year Cooling** (Neutrons/s)
1.0E-01	4.0E-01	6.35E+06	5.89E+06	5.46E+06	5.07E+06	4.70E+06	4.36E+06
4.0E-01	9.0E-01	3.24E+07	3.01E+07	2.79E+07	2.59E+07	2.40E+07	2.23E+07
9.0E-01	1.4	2.98E+07	2.76E+07	2.56E+07	2.38E+07	2.21E+07	2.05E+07
1.4	1.85	2.20E+07	2.04E+07	1.90E+07	1.76E+07	1.64E+07	1.53E+07
1.85	3.0	3.90E+07	3.63E+07	3.38E+07	3.15E+07	2.94E+07	2.74E+07
3.0	6.43	3.52E+07	3.27E+07	3.04E+07	2.83E+07	2.63E+07	2.44E+07
6.43	20.0	3.11E+06	2.88E+06	2.67E+06	2.48E+06	2.30E+06	2.13E+06
Total		1.68E+08	1.56E+08	1.45E+08	1.35E+08	1.25E+08	1.16E+08

TABLE 7.2-4A

CALCULATED HI-TRAC PWR NEUTRON SOURCE PER ASSEMBLY FOR A BURNUP OF 75,000 MWD/MTU

Lower Energy (MeV)	**Upper Energy** (MeV)	**5-Year Cooling** (Neutrons/s)
1.0E-01	4.0E-01	6.82E+07
4.0E-01	9.0E-01	3.48E+08
9.0E-01	1.4	3.18E+08
1.4	1.85	2.34E+08
1.85	3.0	4.11E+08
3.0	6.43	3.75E+08
6.43	20.0	3.34E+07
Total		1.79E+09

NOTE:
Table values obtained from Reference 11, Table 5.2.16

TABLE 7.2-4B

CALCULATED HI-TRAC PWR NEUTRON SOURCE PER ASSEMBLY FOR A BURNUP OF 55,000 MWD/MTU

Lower Energy (MeV)	**Upper Energy** (MeV)	**12-Year Cooling** (Neutrons/s)
1.0E-01	4.0E-01	2.31E+07
4.0E-01	9.0E-01	1.18E+08
9.0E-01	1.4	1.08E+08
1.4	1.85	7.97E+07
1.85	3.0	1.41E+08
3.0	6.43	1.28E+08
6.43	20.0	1.13E+07
Total		6.09E+08

TABLE 7.2-5A

CALCULATED HI-STORM ^{60}Co SOURCE PER ASSEMBLY FOR A BURNUP OF 69,000 MWD/MTU

Location	**5-Year Cooling** (curies)
Lower End Fitting	208.12
Gas Plenum Springs	15.88
Gas Plenum Spacer	9.11
Incore Grid Spacers	539.00
Upper End Fitting	102.08

NOTE:
Table values obtained from Reference 11, Table 5.2.11

TABLE 7.2-5B

CALCULATED HI-STORM ^{60}Co SOURCE PER ASSEMBLY FOR A BURNUP OF 32,500 MWD/MTU

Location	5-Year Cooling (curies)	7-Year Cooling (curies)	9-Year Cooling (curies)	11-Year Cooling (curies)	13-Year Cooling (curies)	15-Year Cooling (curies)
Lower End Fitting	139.25	106.90	82.30	63.19	48.62	37.27
Gas Plenum Springs	10.62	8.16	6.28	4.82	3.71	2.84
Gas Plenum Spacer	6.10	4.68	3.60	2.77	2.13	1.63
Incore Grid Spacers	360.64	276.85	213.15	163.66	125.93	96.53
Upper End Fitting	68.30	52.43	40.37	31.00	23.85	18.28

TABLE 7.2-6A

CALCULATED HI-TRAC ^{60}Co SOURCE PER ASSEMBLY FOR A BURNUP OF 75,000 MWD/MTU

Location	**5-Year Cooling** (curies)
Lower End Fitting	219.47
Gas Plenum Springs	16.74
Gas Plenum Spacer	9.61
Incore Grid Spacers	568.40
Upper End Fitting	107.65

NOTE:
Table values obtained from Reference 11, Table 5.2.12

TABLE 7.2-6B

CALCULATED HI-TRAC ^{60}Co SOURCE PER ASSEMBLY FOR A BURNUP OF 55,000 MWD/MTU

Location	**12-Year Cooling** (curies)
Lower End Fitting	75.11
Gas Plenum Springs	5.73
Gas Plenum Spacer	3.29
Incore Grid Spacers	194.53
Upper End Fitting	36.84

TABLE 7.2-7

CALCULATED ^{60}Co SOURCE PER BPRA PER ASSEMBLY
FOR A BURNUP OF 40,000 MWD/MTU AND A COOLING TIME OF 13 YEARS

Region	**Curies Co-60**
Upper End Fitting	12.1
Gas Plenum Spacer	1.8
Gas Plenum Springs	3.3
Incore	313.8

TABLE 7.2-8

ISOTOPE INVENTORY AND RELEASE FRACTION
Ci/ASSEMBLY

Nuclide	**PWR Fuel Ci/Assembly**	**Release Fraction[(a)]**
Gases		
^{3}H	2.97E+02	0.30
^{129}I	2.64E-02	0.30
^{85}Kr	4.82E+03	0.30
Crud		
^{60}Co	2.18E+01	0.15 normal/offnormal 1.0 accident
Volatiles		
^{90}Sr	5.10E+04	2.0E-04
^{106}Ru	1.44E+04	2.0E-04
^{134}Cs	3.01E+04	2.0E-04
^{137}Cs	7.82E+04	2.0E-04
Fines		
^{241}Pu	7.75E+04	3.0E-05
^{90}Y	5.10E+04	3.0E-05
^{147}Pm	2.57E+04	3.0E-05
^{154}Eu	4.51E+03	3.0E-05
^{244}Cm	5.57E+03	3.0E-05
^{238}Pu	3.76E+03	3.0E-05
^{125}Sb	1.99E+03	3.0E-05
^{155}Eu	1.28E+03	3.0E-05
^{241}Am	8.06E+02	3.0E-05
^{240}Pu	3.65E+02	3.0E-05
^{239}Pu	1.99E+02	3.0E-05
^{137m}Ba	7.38E+04	3.0E-05
^{106}Rh	1.44E+04	3.0E-05

TABLE 7.2-8

Nuclide	**PWR Fuel Ci/Assembly**	**Release Fraction[a]**
^{144}Ce	8.14E+03	3.0E-05
^{144}Pr	8.14E+03	3.0E-05
^{125m}Te	4.86E+02	3.0E-05

[a] B.L. Anderson, et al., Containment Analysis for Type B Packages Used to Transport Various Contents, NUREG/CR-6487, UCRL-ID-124822, Lawrence Livermore National Laboratory, November 1996.

NOTE:
The isotopes, which contribute greater than 0.1 percent to the total curie inventory for the fuel assembly, are considered in the evaluation as fines. The analysis also includes actinides, which contribute greater than 0.01 percent to the total curie inventory for the fuel assembly. This is in accordance with ISG-5.

TABLE 7.3-1A

SURFACE AND 1 METER DOSE RATES FOR THE
OVERPACK WITH AN MPC-32
69,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	**Fuel Gammas** (mrem/hr)	**^{60}Co Gammas** (mrem/hr)	**Neutrons** (mrem/hr)	**Totals** (mrem/hr)
Surface Dose Rate				
1	16.7	25.4	24.1	66.2
2	68.0	0.1	7.1	75.2
3	23.3	25.9	21.1	70.3
4	5.3	0.4	11.0	16.7
4a	6.8	19.0	107.8	133.6
1 Meter Dose Rate				
1	10.6	8.0	2.8	21.4
2	34.1	0.8	2.9	37.8
3	10.3	7.1	3.1	20.5
4	2.0	0.5	4.0	6.5

NOTES:
- Refer to Figure 7.3-1 for the dose locations.
- Gammas generated by neutron capture are included with fuel gammas.
- Gammas from BPRAs are included in the fuel gammas for the portion of the BPRA in the active fuel zone and included in the ^{60}Co gammas for the portion of the BPRA above the active fuel zone.
- Dose location 4a is located directly above the top duct. This is a very localized area of increased dose. Dose location 4a was only calculated at the surface of the lid.
- These values are taken from Reference 5, Appendix Q.

TABLE 7.3-1B

SURFACE AND 1 METER DOSE RATES FOR THE OVERPACK WITH AN MPC-32 32,500 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	**Fuel Gammas** (mrem/hr)	**^{60}Co Gammas** (mrem/hr)	**Neutrons** (mrem/hr)	**Totals** (mrem/hr)
Surface Dose Rate				
1	6.8	17.0	2.9	26.7
2	33.9	0.1	0.8	34.8
3	9.9	18.3	2.6	30.8
4	1.6	1.5	0.9	3.9
4a	2.5	13.4	13.0	28.9
1 Meter Dose Rate				
1	4.9	5.3	0.3	10.5
2	17.0	0.6	0.4	18.0
3	4.6	5.0	0.4	10.0
4	0.4	0.5	0.4	1.3

NOTES:

- Refer to Figure 7.3-1 for the dose locations.
- Gammas generated by neutron capture are included with fuel gammas.
- Gammas from BPRAs are included in the fuel gammas for the portion of the BPRA in the active fuel zone and included in the ^{60}Co gammas for the portion of the BPRA above the active fuel zone.
- Dose location 4a is located directly above the top duct. This is a very localized area of increased dose. Dose location 4a was only calculated at the surface of the lid.
- These values are taken from Reference 5, page A-7.

TABLE 7.3-2A

SURFACE AND 1 METER DOSE RATE ESTIMATES FOR THE
TRANSFER CASK WITH THE MPC-32
75,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	**Fuel Gammas** (mrem/hr)	**(n,γ) Gammas** (mrem/hr)	**^{60}Co Gammas** (mrem/hr)	**Neutrons** (mrem/hr)	**Totals** (mrem/hr)	**Totals with BPRAs** (mrem/hr)
Surface Dose Rate						
1	8.4	82.5	134.2	554.5	779.6	780.6
2	151.1	244.3	0.01	383.9	779.3	800.5
3	1.9	8.7	83.0	884.9	978.5	1004.8
4	55.4	11.2	454.2	1023.9	1544.8	1698.7
4 (outer)	6.5	8.0	56.4	21.5	92.3	111.3
5 (pool)	73.0	4.9	606.1	3844.7	4528.7	4539.0
1 Meter Dose Rate						
1	19.9	32.9	17.2	91.3	161.3	164.0
2	67.3	79.2	0.7	131.0	278.1	287.6
3	7.5	18.6	16.8	81.4	124.3	130.9
4	15.4	2.7	109.4	105.5	232.9	269.8
5 (pool)(est)	28.8	0.9	293.0	665.9	1603.9	1611.5

NOTES:

- Refer to Figure 7.3-2 for the dose locations.
- Dose location 4 (outer) is the radial segment at dose location 4, which is 18-24 inches from the center of the overpack.
- Dose rates are based on no water within the MPC. During the MPC lid welding the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- The dose rate below the bottom lid is calculated in the center of the lid. The HI-STORM 100 System FSAR demonstrates that this dose rate will be greatly reduced at the outer edge of the overpack.
- These values are based on Reference 5, Appendix P. As noted in the report to obtain values for an MPC-32, the MPC-24 values of HI-STORM FSAR Rev. 7 Table 5.1.8 are ratioed by the number of fuel assemblies (i.e. 32/24) to obtain the values in this Table.
- 1-meter dose rates for point 5 are estimated based on applying the ratio of dose rates, surface and 1-meter, for point 5 (transfer) to the surface dose rate for 5 (pool), as only the pool lid is used.
- Values in tables are nominal based on design basis fuel.
- Accident analysis values for complete loss of water in water jacket are provided in Section 8.2.11.3.

TABLE 7.3-2B

SURFACE AND 1 METER DOSE RATES FOR THE TRANSFER CASK WITH THE MPC-24
55,000 MWD/MTU AND 12-YEAR COOLING

Dose Point Location	**Fuel Gammas** (mrem/hr)	**(n,γ) Gammas** (mrem/hr)	**^{60}Co Gammas** (mrem/hr)	**Neutrons** (mrem/hr)	**Totals** (mrem/hr)
Surface Dose Rate					
1	1.9	23.3	36.5	133.6	195.3
2	38.0	63.0	0.0	160.0	261.0
3	0.3	4.2	25.2	261.4	291.1
4	9.9	2.9	115.1	261.4	389.3
4 (outer)	1.1	2.0	14.5	5.5	23.1
5 (pool)	12.9	1.2	155.6	982.0	1151.7
5 (pool with temp. shield)	7.2	11.5	110.0	58.5	187.2
1 Meter Dose Rate					
1	3.6	8.5	4.1	20.1	36.3
2	12.1	20.1	0.3	31.7	64.2
3	2.0	5.4	4.0	16.8	28.2
4	2.7	0.7	27.9	26.9	58.2

NOTES:
- Refer to Figure 7.3-2 for the dose locations.
- Gammas from BPRAs are included in the fuel gammas for the portion of the BPRA in the active fuel zone and included in the ^{60}Co gammas for the portion of the BPRA above the active fuel zone.
- Dose location 4 (outer) is the radial segment at dose location 4, which is 18-24 inches from the center of the overpack.
- Dose rates are based on no water within the MPC. During the MPC lid welding the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- The dose rate below the bottom lid is calculated in the center of the lid. The HI-STORM 100 System FSAR demonstrates that this dose rate will be greatly reduced at the outer edge of the overpack.
- These values are taken from Reference 5, Appendix J.
- Values in tables are nominal based on design basis fuel.
- Accident analysis values for complete loss of water in water jacket are provided in Section 8.2.11.3.

TABLE 7.3-3

TOTAL SURFACE AND 1 METER DOSE RATES FOR THE TRANSFER CASK WITH VARIOUS MPCs

Dose Point Location	MPC-24 55,000 MWD/MTU 12-yr. Cooling (mrem/hr)	MPC-24 75,000 MWD/MTU 5-yr. Cooling (mrem/hr)	MPC-32 75,000 MWD/MTU 5-yr. Cooling (mrem/hr)
Surface Dose Rate			
1	195.3	585.4	780.6
2	261.0	600.4	800.5
3	291.1	753.6	1004.8
4	389.3	1274.0	1698.7
4 (outer)	23.1	83.5	111.3
5 (pool)	1151.7	3404.2	4539.0
1 Meter Dose Rate			
1	36.3	123.0	164.0
2	64.2	215.7	287.6
3	28.2	98.2	130.9
4	58.2	202.3	269.8

NOTES:
- Refer to Figure 7.3-2 for the dose locations.
- Dose location 4 (outer) is the radial segment at dose location 4, which is 18-24 inches from the center of the overpack.
- Dose rates are based on no water within the MPC. During the MPC lid welding the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- The dose rate below the bottom lid is calculated in the center of the lid. The HI-STORM 100 System FSAR demonstrates that this dose rate will be greatly reduced at the outer edge of the overpack.
- These values are taken from Reference 5.
- Values in tables are nominal based on design basis fuel.
- Accident analysis values for complete loss of water in water jacket are provided in Section 8.2.11.3.

TABLE 7.3-4A

DOSE RATE VERSUS DISTANCE FROM A SINGLE OVERPACK
WITH THE MPC-32
69,000 MWD/MTU AND 5-YEAR COOLING

Distance		**mrem/hr**		
m	**ft**	**Side-dose rate**	**Top-dose rate**	**Total dose rate**
12.19	40.00	2.03E+00	1.42E-02	2.05E+00
18.29	60.00	1.00E+00	1.02E-02	1.01E+00
24.38	80.00	5.80E-01	7.48E-03	5.87E-01
30.48	100.00	3.73E-01	5.86E-03	3.79E-01
45.72	150.00	1.61E-01	3.23E-03	1.64E-01
50.00	164.04	1.35E-01	2.84E-03	1.38E-01
60.96	200.00	8.68E-02	2.03E-03	8.88E-02
91.44	300.00	3.45E-02	9.18E-04	3.54E-02
100.00	328.08	2.79E-02	7.51E-04	2.86E-02
121.92	400.00	1.68E-02	4.94E-04	1.73E-02
150.00	492.13	9.67E-03	2.83E-04	9.95E-03
200.00	656.17	4.33E-03	1.22E-04	4.45E-03
250.00	820.21	2.15E-03	5.70E-05	2.20E-03
300.00	984.25	1.18E-03	2.82E-05	1.20E-03
350.00	1148.29	6.59E-04	1.50E-05	6.74E-04
400.00	1312.34	3.81E-04	8.33E-06	3.90E-04
450.00	1476.38	2.29E-04	4.63E-06	2.34E-04
500.00	1640.42	1.44E-04	2.68E-06	1.47E-04
550.00	1804.46	9.35E-05	1.53E-06	9.50E-05
600.00	1968.50	6.42E-05	9.16E-07	6.51E-05
650.00	2132.55	4.13E-05	6.13E-07	4.19E-05
700.00	2296.59	2.80E-05	3.63E-07	2.83E-05
750.00	2460.63	1.84E-05	2.34E-07	1.86E-05
800.00	2624.67	1.26E-05	1.61E-07	1.27E-05
850.00	2788.71	9.10E-06	1.07E-07	9.20E-06
900.00	2952.76	6.16E-06	7.19E-08	6.24E-06

NOTE:
Table values are derived from the annual dose numbers at 5 years cooling from Reference 5, Appendix P, adjusted for the annual exposure of 8760 hours/yr.

TABLE 7.3-4B

DOSE RATE VERSUS DISTANCE FROM A SINGLE OVERPACK
WITH THE MPC-32
32,500 MWD/MTU AND 5-YEAR COOLING

Distance		mrem/hr		
m	**ft**	**Side-dose rate**	**Top-dose rate**	**Total dose rate**
12.19	40.00	1.02E+00	1.97E-03	1.02E+00
18.29	60.00	5.00E-01	1.42E-03	5.01E-01
24.38	80.00	2.88E-01	1.05E-03	2.89E-01
30.48	100.00	1.86E-01	8.31E-04	1.87E-01
45.72	150.00	8.04E-02	4.68E-04	8.09E-02
50.00	164.04	6.64E-02	4.14E-04	6.68E-02
60.96	200.00	4.27E-02	2.99E-04	4.30E-02
91.44	300.00	1.69E-02	1.40E-04	1.71E-02
100.00	328.08	1.37E-02	1.16E-04	1.38E-02
121.92	400.00	8.33E-03	7.71E-05	8.41E-03
150.00	492.13	4.76E-03	4.56E-05	4.81E-03
200.00	656.17	2.12E-03	2.02E-05	2.14E-03
250.00	820.21	1.05E-03	9.63E-06	1.06E-03
300.00	984.25	5.80E-04	4.88E-06	5.85E-04
350.00	1148.29	3.19E-04	2.61E-06	3.22E-04
400.00	1312.34	1.84E-04	1.45E-06	1.86E-04
450.00	1476.38	1.11E-04	8.08E-07	1.11E-04
500.00	1640.42	6.88E-05	4.61E-07	6.93E-05
550.00	1804.46	4.45E-05	2.63E-07	4.47E-05
600.00	1968.50	2.99E-05	1.58E-07	3.01E-05
650.00	2132.55	1.90E-05	1.01E-07	1.91E-05
700.00	2296.59	1.25E-05	5.96E-08	1.26E-05
750.00	2460.63	8.24E-06	3.71E-08	8.28E-06
800.00	2624.67	5.42E-06	2.48E-08	5.45E-06
850.00	2788.71	3.90E-06	1.61E-08	3.91E-06
900.00	2952.76	2.66E-06	1.06E-08	2.67E-06

TABLE 7.5-1

NORMAL OPERATION DOSE RATES AND ANNUAL DOSES AT THE SITE BOUNDARY AND NEAREST RESIDENT FROM DIRECT RADIATION FROM THE 140 CASKS AT THE DIABLO CANYON ISFSI

Location	**Dose Rate (mrem/hr)**	**Occupancy (hours/year)**	**Annual Dose (mrem)**
Site Boundary (1,400 ft / 427 m)	8.5E-03	2,080	17.6
Nearest Resident (1.5 mi / 7,920 ft / 2414 m)	3.4E-07	8,760	3.0E-03

TABLE 7.5-2

NORMAL OPERATION ANNUAL DOSES AT THE SITE BOUNDARY AND NEAREST RESIDENT FROM AN ASSUMED EFFLUENT RELEASE FROM THE 140 CASKS AT THE DIABLO CANYON ISFSI

	Annual Dose[a] (mrem)
Site Boundary (1,400 ft / 427 m)	
Whole body ADE[b]	0.064
Thyroid ADE	0.010
Critical Organ ADE (Max)	0.35
Nearest Resident (1.5 mi / 7,920 ft / 2,414 m)	
Whole body ADE	0.27
Thyroid ADE	0.043
Critical Organ ADE (Max)	1.46

NOTE:
This Table is provided for historical information only. See discussion in Section 7.2.2.

(a) The effluent release dose for the nearest resident is conservatively chosen to be the site boundary dose, adjusted for full-time occupancy (8,760/2,080). This is conservative since the χ/Q for the nearest resident would be less than that used for the site boundary. The occupancy time for the site boundary is 2,080 hours and the occupancy time for the nearest resident is 8,760 hours.

(b) ADE is annual dose equivalent.

TABLE 7.5-3

DOSE RATES AT THE SITE BOUNDARY
FROM OVERPACK LOADING OPERATIONS

Condition	**Dose Rate** (mrem/hr)	**Event Duration** (hours)	**Loadings per year**	**Annual Dose** (mrem)
MPC in transfer cask	2.0E-03[(a)]	12	8	3.22E-01

NOTE:
HI-TRAC contribution at the site boundary is estimated by scaling the HI-TRAC dose at 1 meter (dose location 2 in Table 7.3-2A is used for this purpose) by the dose rates reduction obtained for HI-STORM between 1 and 400 meters (the site boundary is at 426.72 meters).

TABLE 7.5-4

TOTAL ANNUAL OFFSITE COLLECTIVE DOSE (MREM) AT THE SITE BOUNDARY AND NEAREST RESIDENT FROM THE DIABLO CANYON ISFSI

	Normal Operations				**Off-Normal Operations**		
Organ	**Effluent Release[c]**	**Direct Radiation[c]**	**Overpack Loading Operations[d]**	**Other Uranium Fuel Cycle Operations[a]**	**Effluent Release[e]**	**Total (normal + off-normal)**	**10 CFR 72.104 Regulatory Limit**
Site Boundary (1,400 ft / 427 m)							
Whole body ADE[b]	0	17.6	0.3	4.357E-02	1.27E-03	17.9	25
Thyroid ADE	0	17.6	0.3	1.260E-01	1.02E-04	18.0	75
Critical organ ADE (Max)	0	17.6	0.3	5.590E-02	9.31E-03	18.0	25
Nearest Resident (1.5 miles / 7,920 ft / 2414 m)							
Whole body ADE	0	3.0E-03	5.1E-05	4.357E-02	5.33E-03	0.047	25
Thyroid ADE	0	3.0E-03	5.1E-05	1.260E-01	4.31E-04	0.13	75
Critical organ ADE (Max)	0	3.0E-03	5.1E-05	5.590E-02	3.92E-02	0.098	25

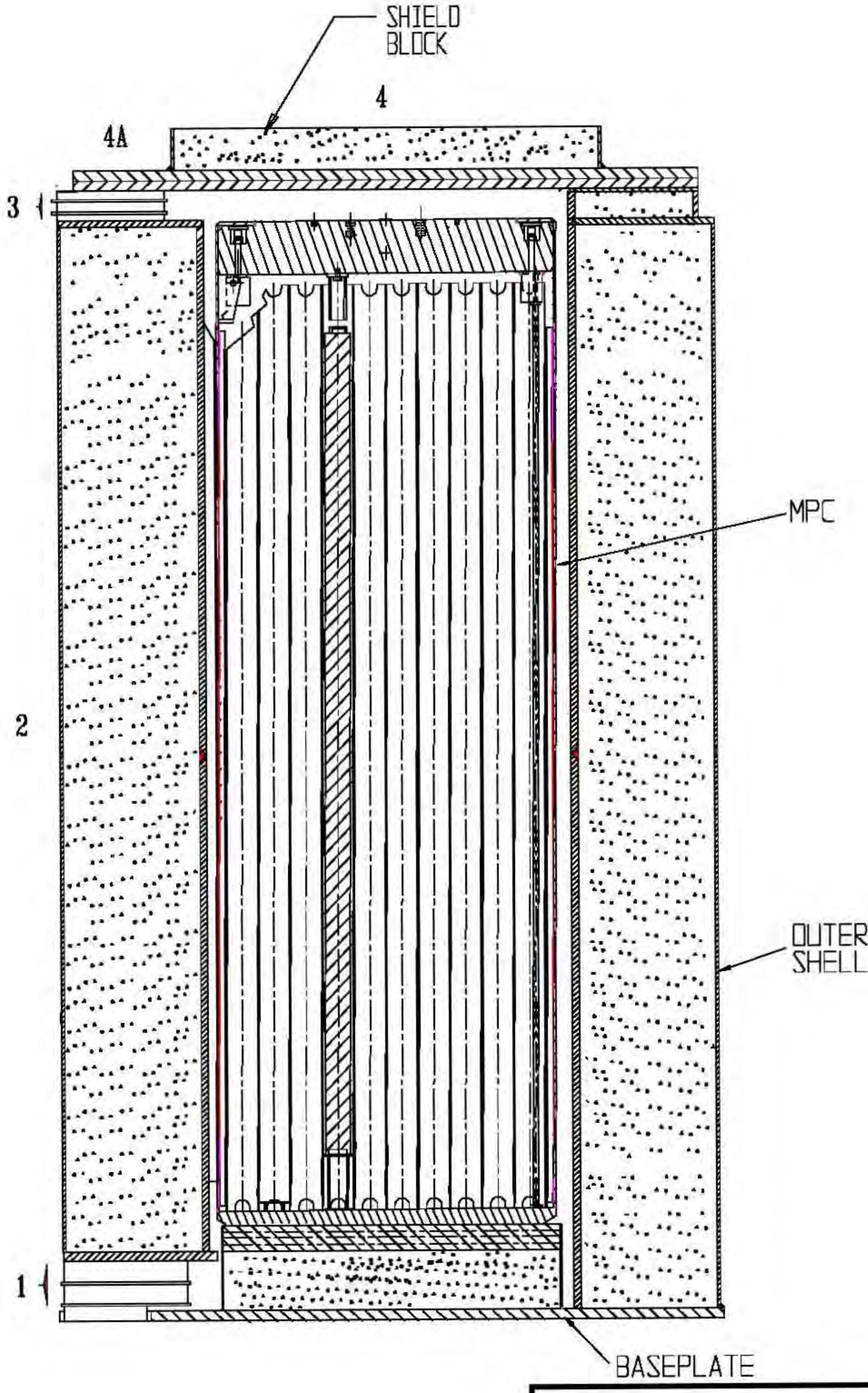
(a) Data for uranium fuel cycle operations were obtained from the DCPP FSAR Update, Rev. 11, Table 11.3-32. Table 11.3-32 was selected based on the highest dose values in the sectors at the site boundary (0.5 miles). These dose values for the site boundary were conservatively applied to the nearest resident. The critical organ dose listed was based on the total liver dose in Table 11.3-32. The values listed in Table 11.3-32 should bound the results calculated from effective dose equivalent methodology.

(b) ADE is annual dose equivalent.

(c) 140 casks

(d) From Table 7.5-3. For nearest resident, the value is scaled by the ratio of direct radiation dose from the site boundary to the nearest resident.

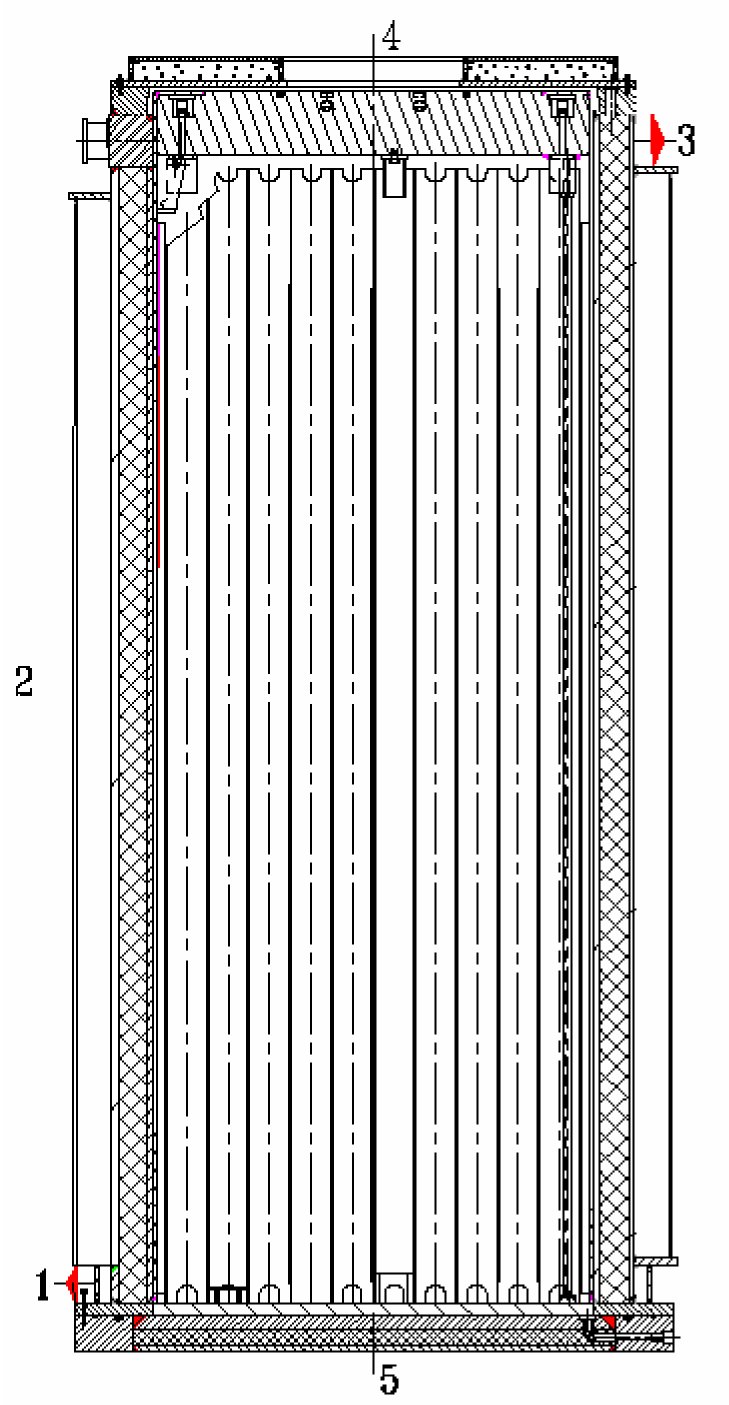
(e) From Table 8.1-1.



FSAR UPDATE

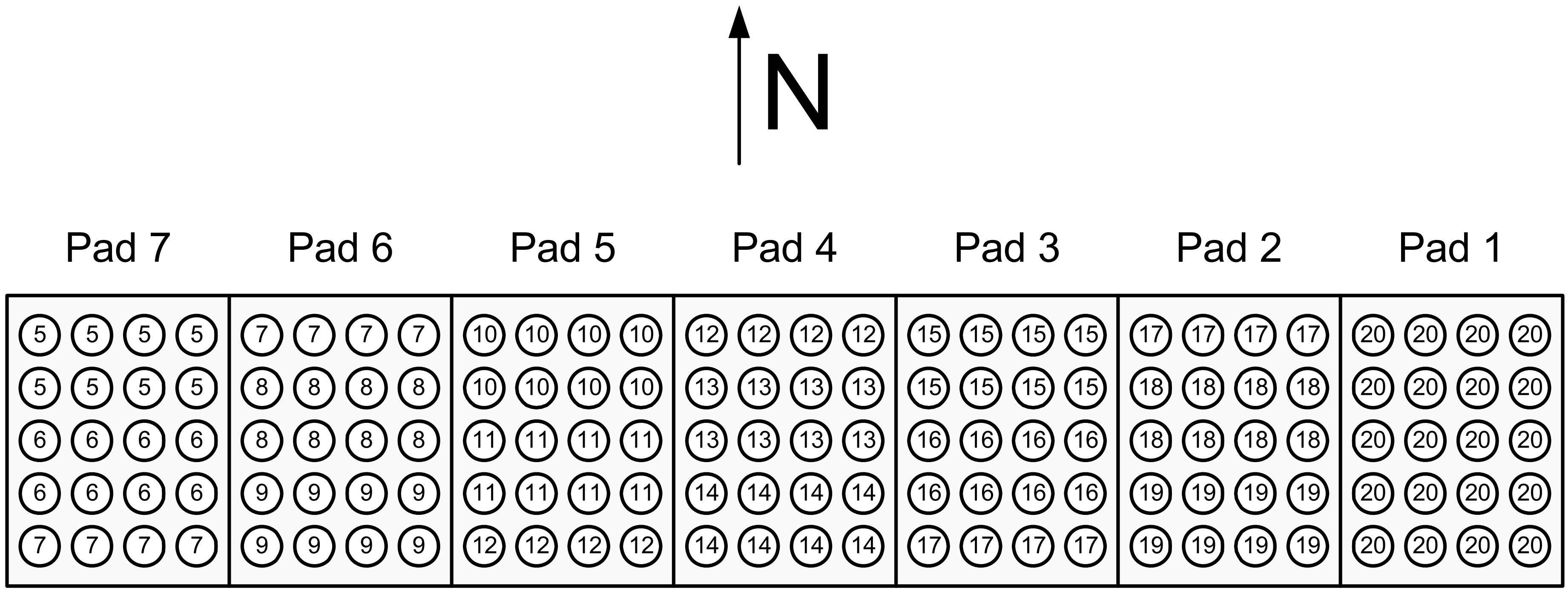
DIABLO CANYON ISFSI

FIGURE 7.3-1
CROSS SECTION ELEVATION VIEW OF THE GENERIC HI-STORM 100S OVERPACK WITH DOSE POINT LOCATIONS



Note:
Numbers refer to FSAR Table 7.3-2

FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 7.3-2 CROSS SECTION ELEVATION VIEW OF TYPICAL HI-TRAC TRANSFER CASK WITH DOSE POINT LOCATIONS



○ = Cask showing cooling time, in years, of the fuel assemblies

FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 7.3-3
A PLAN VIEW OF THE ISFSI AT THE COMPLETION OF LOADING OPERATIONS

CHAPTER 8

ACCIDENT ANALYSES

CONTENTS

CHAPTER 8

ACCIDENT ANALYSES

TABLES

CHAPTER 8

ACCIDENT ANALYSES

FIGURES

CHAPTER 8

ACCIDENT ANALYSES

This chapter describes the accident analyses for the Diablo Canyon ISFSI. Sections 8.1 and 8.2 evaluate the safety of the ISFSI under off-normal operations and accident conditions, respectively. For each event, the postulated cause of the event, detection of the event, and evaluation of the event effects and consequences, corrective actions, and radiological impact are presented. Unless otherwise identified in Chapter 8 or other FSAR sections, the MPC 32 was evaluated as a bounding condition. The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of off-normal events and accidents without affecting function and are in compliance with the applicable acceptance criteria. Section 8.3 summarizes site characteristics that affect the safety analysis.

As discussed in Section 1.1 the licensed HI-STORM 100 System at the Diablo Canyon ISFSI has been modified to facilitate fuel-loading campaigns. These modifications were performed in accordance with 10 CFR 72.48 and involve the MPC-32 canister, HI-TRAC 125D transfer cask, HI-STORM 100SA overpack, CTF, low profile transporter, cask transporter, and auxiliary components used in the loading and transport to the ISFSI facility. The originally-licensed MPC-24s will likely not be used at the ISFSI and would require modifications, analyses and associated 10 CFR 72.48 evaluations similar to the MPC-32 prior to their use. Most of the accident and off-normal analyses and evaluations performed for the licensed HI-STORM 100 system remain bounding for the modified system. However, in cases where they do not and a re-analysis or site specific analysis was required, those analyses are identified and referenced in their related sections below.

8.1 OFF-NORMAL OPERATIONS

This section addresses events designated as Design Event II, as defined by ANSI/ANS-57.9 (Reference 1). The following are considered off-normal events for the Diablo Canyon ISFSI:

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

- Cask transporter off-normal operation

For each event, the postulated cause of the event, detection of the event, an evaluation of the event effects and consequences, corrective actions, and radiological impact are presented. The results of the evaluations performed herein demonstrate that the HI-STORM 100 System used at Diablo Canyon can withstand the effects of off-normal events without affecting function and are in compliance with the applicable acceptance criteria. The following sections present the evaluation of the HI-STORM 100 System for the design-basis, off-normal conditions that demonstrate that the requirements of 10 CFR 72.122 are satisfied and that the corresponding radiation doses satisfy the requirements of 10 CFR 72.104(a).

8.1.1 OFF-NORMAL PRESSURES

The HI-STORM 100SA overpack is a ventilated cask design. The sole pressure boundary of the storage system is the multi-purpose canister (MPC). The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure, variations in the helium temperature, and leakage of any gases contained within the fuel rods. The analyzed off-normal environmental temperature is 100°F and peak solar insolation is assumed. This bounds the Diablo Canyon ISFSI maximum off-normal site ambient temperature and solar insolation values. The MPC off-normal pressure evaluation includes the conservative assumption that 10 percent of the fuel rods rupture, allowing 100 percent of the fill gas and 30-percent of the fission gases from these fuel rods to be released to the MPC cavity. This assumption is consistent with the guidance in NUREG-1536 for the review of dry storage cask designs (Reference 2).

8.1.1.1 Postulated Cause of Off-Normal Pressure

After fuel assembly loading, the MPC is drained, dried, and backfilled with an inert gas (helium) to ensure long-term fuel cladding integrity during dry storage. The pressure of the gas in the MPC cavity is affected by the initial fill pressure, the MPC cavity volume, the decay heat emitted by the stored fuel, the presence of nonfuel hardware, fuel-rod gas leakage, ambient temperature, and solar insolation. Of these, the initial fill pressure, presence of non-fuel hardware, and MPC cavity volume do not vary with time in storage and can be ignored as a cause of off-normal pressure. The decay heat emitted by the stored fuel decreases with time and is conservatively accounted for in the analysis by using the highest rate of decay heat for a given fuel cooling time. Off-normal pressure is conservatively evaluated considering a concurrent non-mechanistic rupture of 10 percent of the stored fuel rods during a time of maximum off-normal ambient temperature (100°F) and full solar insolation.

8.1.1.2 Detection of Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the MPC off-normal internal pressure without any effects on its ability to perform its design safety functions. No personnel actions or equipment are required to respond to an off-normal pressure event. Therefore, no detection instrumentation is required.

8.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

A new analysis of MPC pressure for this off-normal event was performed for two scenarios. The first scenario is for uniform loading and the second is for regionalized loading. In addition, the analysis used two normal ambient temperatures depending on the specific configurations. A normal ambient temperature of 65°F was assumed for a loaded MPC contained in a HI-STORM overpack on the ISFSI pad and for a loaded MPC contained in a HI-STORM overpack within the CTF. All transport configurations with a loaded MPC contained within the HI-TRAC assumed a normal ambient temperature of 100°F. The analysis for both HI-STORM configurations also assume 10 percent of the fuel rods ruptured, peak insolation, maximum decay heat, maximum backfill pressure, IFBA fuel and the effect of nonfuel hardware.

The MPC-32 was used as the bounding MPC in this analysis because it provides the maximum internal pressure for all MPCs to be used at the Diablo Canyon ISFSI (see Section 4.2.3.3.2.2 for justification). The resulting pressure for the MPC-32 with 65°F ambient temperature is 92.6 psig (Reference 14, Table B.5.10). The added effect of increasing the ambient temperature from 65°F to the maximum off-normal temperature of 100°F on the internal pressure was included in the calculation in Reference 13 for both the HI-STORM configurations. For the transport conditions the added effect of increasing the ambient temperature from 80°F to the maximum off-normal temperature of 100°F was conservatively evaluated using the Ideal Gas Law. Assuming the MPC cavity gas temperature increased by the full 20°F, the resulting absolute pressure P2 for the transport conditions is computed as follows:

$$P2 = P1 \times [(T1 + \triangle T)/T1]$$

Where,

P_1 = Absolute pressure at T_1 = 80.9 psig (95.6 psia)

T_1 = Absolute bulk temperature of the MPC cavity gas with design basis fuel decay heat = 514°K (Reference 14,Table C.5)

$\triangle T$ = Absolute bulk MPC cavity gas temperature increase = 20°F, or 11.1°K

The resulting absolute pressure (P_2) was computed to be 82.6 psig for the transport condition. Pressure values for both the storage and transport conditions are below the normal/off-normal MPC internal design pressure of 100 psig.

8.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. There are no corrective actions associated with off-normal pressure.

8.1.1.5 Radiological Impact from Off-Normal Pressure

The off-normal pressure event has no radiological impact because the confinement barrier and shielding integrity are not affected.

8.1.2 OFF-NORMAL ENVIRONMENTAL TEMPERATURES

The off-normal temperature ranges for which the HI-STORM 100 System is designed are summarized in the HI-STORM 100 System FSAR (Reference 3) Section 2.2.2. The off-normal temperature evaluation is described in HI-STORM 100 System FSAR Section 11.1.2. Off-normal environmental temperature ranges of -40 to 100°F (for the HI-STORM 100SA overpack and ISFSI storage pads) and 0 to 100°F (for the HI-TRAC transfer cask, cask transporter, and cask transfer facility) conservatively bound off-normal temperatures at the Diablo Canyon ISFSI site (24°F to 97°F). The off-normal environmental temperature ranges are used as the design criteria for the concrete storage pad, cask transporter, and CTF. The ranges of off-normal temperatures evaluated bound the historical temperature variations at the Diablo Canyon ISFSI.

This off-normal event is of a short duration. Therefore, the resultant fuel cladding temperatures for the cask evaluations are compared against the accident condition (short-term) temperature limits.

8.1.2.1 Postulated Cause of Off-Normal Environmental Temperatures

The off-normal environmental temperature is postulated as a constant ambient temperature caused by unusual weather conditions. To determine the effects of off-normal temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

8.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There are no personnel actions or equipment required for mitigation of an off-normal temperature event. Deleterious effects of off-normal temperatures on the cask transporter, CTF, and concrete storage pad are precluded by design. Administrative procedures based on Diablo Canyon ISFSI TS 5.1.3 prohibit cask handling if temperatures fall outside the off-normal temperature limits. Ambient temperature is available from thermometers used for the DCPP site meteorological measurement program.

8.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

There are no adverse safety effects resulting from off-normal environmental temperatures on the cask transporter, CTF, or concrete storage pads, since they are designed for these temperature ranges.

The off-normal event, considering a maximum off-normal ambient temperature of 100°F has been evaluated for the HI-STORM 100 System and is described in the HI-STORM 100 System FSAR Section 11.1.2.3. The evaluation was performed for the loaded transfer cask and the loaded overpack, assuming design-basis fuel with the maximum decay heat and the most restrictive thermal resistance. The 100°F environmental temperature was applied with peak solar insolation. Thermal analysis contained in the HI-STORM 100 System FSAR indicates that the MPC-32 has the highest design-basis decay heat load and always yields the highest cask system component and content temperatures. As such, only the MPC-32 is evaluated since the MPC-24 and MPC-24E thermal performance will be bounded by that of the MPC-32 under all conditions.

The HI-STORM 100 System maximum temperatures for components close to the design-basis temperatures are conservatively calculated at both environmental temperatures of 65°F and 80°F as an initial condition for this off-normal event. These temperatures (for MPC-32 and the overpack) are shown in Tables B.5.2 and C.2 of Reference 14. The maximum off-normal environmental temperature is 100°F, which is an increase of 20°F to 35°F, depending on the configuration, over the normal design temperature. The limiting component maximum off-normal temperatures are shown in Table B.5.4 of Reference 14. The temperatures are all below the applicable material short-term temperature limits.

The off-normal event considering a limiting low environmental temperature of -40°F and no insolation for a duration sufficient to reach thermal equilibrium has been evaluated with respect to overpack material brittle fracture at this low temperature. The overpack and MPC are conservatively assumed to reach -40°F throughout the structure. The minimum off-normal environmental temperature specified for the transfer cask is 0°F and the transfer cask is conservatively assumed to reach 0°F throughout the structure.

This evaluation is discussed in the HI-STORM 100 System FSAR Section 3.1.2.3 and the results are acceptable. Administrative procedures based on Diablo Canyon ISFSI TS 5.1.3 prohibit cask handling operations at environmental temperatures below 0°F.

8.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. The cask transporter, CTF, and ISFSI pad are designed for temperature ranges consistent with the dry storage cask components used at these facilities. Therefore, no corrective actions are required for off-normal environmental temperature conditions.

8.1.2.5 Radiological Impact of Off-Normal Temperatures

Off-normal environmental temperatures have no radiological impact as the integrity of the confinement barrier and shielding are unaffected by off-normal temperatures. The effect of elevated temperatures does not significantly increase the doses associated with the design-basis leak rate from the MPCs and is bounded by the results of the off-normal failure of fuel cladding event assessed in Section 8.1.3.

8.1.3 CONFINEMENT BOUNDARY LEAKAGE

The HI-STORM 100 System MPC has a welded confinement boundary to contain radioactive fission products under all design-basis normal, off-normal, and accident conditions. The radioactivity confinement boundary is defined by the MPC shell, baseplate, MPC lid, and vent and drain port cover plates. A non-mechanistic failure of fuel cladding in conjunction with allowable leakage in the MPC confinement boundary has been evaluated as both an off-normal and an accident condition (Reference 7). The difference between the two evaluations is in the radioactive source term, the bounding temperature and pressure determined in the thermal analysis of Reference 11 and the χ/Q value used for each of the two conditions. The analytical technique and assumptions used in both evaluations are consistent with Interim Staff Guidance (ISG) Document 5 (Reference 5). All other inputs to the confinement boundary leak dose analysis are identical for the off-normal and accident analyses. The accident condition is addressed in Section 8.2.7 of this FSAR and is not discussed further here.

Since this event is applicable only to the MPC, the evaluation is applicable for all locations (that is, in the cask transporter, at the CTF, or on the ISFSI pad) and is independent of whether the MPC is inside the transfer cask or the overpack. Due to the close proximity of these three locations, the two χ/Q values used for the off-normal and accident condition evaluations are the same for all three postulated release locations.

This section only applies to the initial 16 casks loaded at the Diablo Canyon ISFSI. Following construction of the first 16 casks, the testing requirement for the MPC boundary welds was changed to the leaktight criteria of ANSI N14.5-1997. The vent and drain port cover plate welds helium leak testing requirements had been changed to the “leaktight” criteria of ANSI N14.5-1997 in LA 1. The lid-to-shell (LTS) weld is a large, multi-pass weld which is placed and inspected in accordance with ISG-15; therefore, in accordance with ISG-18, leakage from this weld is considered non-credible. Since all the closure welds meet a leaktight criteria, the confinement boundary of the subsequently fabricated MPCs can be considered leak tight.

8.1.3.1 Postulated Cause of Confinement Boundary Leakage

Based on the design of the MPC vessel and the protection provided by the transfer cask and the overpack, a leak in the MPC confinement boundary is not considered credible, so no cause is identified. Also, there is no credible mechanism for inducing the level of fuel failure assumed for this event. This off-normal condition is evaluated as a non-mechanistic event.

8.1.3.2 Detection of Confinement Boundary Leakage

The MPC is a welded cylindrical enclosure. There are no mechanical joints or seals in the confinement boundary. The confinement boundary is designed to maintain its integrity under all design basis normal, off-normal, and accident conditions. Therefore, leakage detection equipment is not required.

8.1.3.3 Analysis of Effects and Consequences of Confinement Boundary Leakage

The MPC confinement boundary is designed to remain intact under all design basis normal, off-normal, and accident conditions. However, as a defense-in-depth measure, the MPC closure ring, which provides a redundant weld for the MPC lid-to-shell weld and the vent and drain port cover plate welds, is designed to withstand full MPC cavity pressure. Therefore, the closure ring would provide the confinement boundary in this event. The dose consequences of a hypothetical, non-mechanistic confinement boundary leak are discussed in Section 8.1.3.5.

8.1.3.4 Corrective Action for Confinement Boundary Leakage

There is no corrective action required for the assumed leakage in the MPC confinement boundary because leakage in excess of allowable is not considered credible. Also, the assumed level of fuel failure is not considered credible.

8.1.3.5 Radiological Impact of Confinement Boundary Leakage

The dose consequences of a non-mechanistic leak in the MPC confinement boundary have been analyzed on a site-specific basis for the Diablo Canyon ISFSI using appropriate source terms, release fraction, leak rate, meteorology, breathing rate, and occupancy times. The analysis of this abnormal event considers the rupture of 10 percent of the stored fuel rods. The evaluation of this event under normal conditions is discussed in Section 7.5.2. The same methodology with the unique off-normal source is used here. Annual doses at the site boundary and nearest resident were calculated. The results are provided in Table 8.1-1 for the analysis of a single HI-STORM cask in the off-normal condition. The calculated doses are less than the regulatory limits in 10 CFR 72.104(a).

8.1.4 PARTIAL BLOCKAGE OF AIR INLETS

The HI-STORM 100 System overpack is designed with inlet and outlet air ducts, four each at the top and bottom of the overpack structure with the lid installed. Each duct opening includes a stainless steel perforated plate (screen) across its outer face. These perforated plates (screens) ensure the air ducts are protected from the incursion of foreign objects. Each set of four air inlet and outlet air ducts are spaced 90 degrees apart around the circumference of the overpack and it is highly unlikely that blowing debris during normal or off-normal operation could block all of the air inlet ducts. It is conservatively assumed, as an off-normal condition, that two of the four air inlet ducts are blocked. Blockage of the inlet air ducts is assumed to be thermally equivalent to blockage of the outlet air ducts. The evaluation of this off-normal event is included in Section B.5.4 of References 13 and 14. In this evaluation this condition is defined as 50 percent blockage of all the inlet ducts. Per the evaluation the resulting decrease in flow area increases the inlet air flow resistance. The results in Table B.5.4 of References 13 and 14, show that the fuel cladding, MPC, and HI-STORM 100SA component temperatures are below their temperature limits. The MPC-32 pressure for this condition is provided in Table B.5.10 of Reference 14 and the result is below the off-normal design pressure specified in DC ISFSI UFSAR.

8.1.4.1 Postulated Cause of Partial Blockage of Air Inlets

It is assumed that all the air inlet ducts are 50 percent blocked, although the protective perforated plates (screens) prevent foreign objects from entering into the ducts. The perforated plates (screens) are inspected periodically, as required by the Diablo Canyon ISFSI TS. Any duct blockage would be detected by visual inspection and removed to restore the heat removal system to full operational condition.

8.1.4.2 Detection of Partial Blockage of Air Inlets

Detection of partial blockage of air inlet ducts would occur during the routine visual surveillance of the storage cask air duct perforated plates (screens) required by the Diablo Canyon ISFSI TS. The frequency of inspection is conservatively based on an assumed complete simultaneous blockage of all four air inlet ducts (Diablo Canyon ISFSI TS Bases).

8.1.4.3 Analysis of Effects and Consequences of Partial Blockage of Air Inlets

Blockage of the overpack air inlet ducts can affect the heat removal process of the dry storage system. The magnitude of the effect is dependent upon the rate of decay heat emission from the stored fuel (itself dependent upon the fuel burnup and cooling time) and the ambient air temperature. A bounding evaluation was performed for 50 percent blockage of all the inlet air ducts with the MPC-32 inside the overpack, at its maximum decay heat load at the ambient air temperature of 65°F. As stated above, the Diablo Canyon site-specific evaluation (Reference 14) assumes an annual-average ambient air temperature of 65°F, which bounds the actual annual-average ambient air temperature for the Diablo Canyon site of 55°F. The MPC-32 decay heat load bounds the MPC-24, MPC-24E, and MPC-24EF heat loads due to the presence of eight additional fuel assemblies. Computed component temperatures for 50 percent blockage of all air inlet ducts are less than the allowable component short-term temperature limits. Blocking of four ducts is treated as an accident in Section 8.2.15. The results are shown in Table B.5.6 of Reference 14.

The MPC cavity pressure for 50 percent blocked air ducts was also evaluated. The computed MPC internal pressure, using the maximum heat load and fill pressure, was 78.2 psig, which is less than the normal condition MPC design pressure of 100 psig (Reference 14, Table B.5.10).

8.1.4.4 Corrective Action for Partial Blockage of Air Inlets

The corrective action for the partial blockage of air inlet ducts is the removal of the cause of the blockage, and the cleaning, repair, or replacement, as necessary, of the affected perforated plates (screens). After clearing of the blockage, the cask heat removal system is restored to its design condition, and temperatures will return to the normal range. Partial blockage of air inlet ducts does not affect the ability of the H-STORM 100 System to safely store spent fuel for the long term.

Inspection of the overpack air duct perforated plates (screens) is performed at a 24-hour frequency as required by the Diablo Canyon ISFSI TS. This inspection ensures blockage of air inlet ducts is detected and appropriately corrected.

8.1.4.5 Radiological Impact of Partial Blockage of Air Inlets

For partial blockage of air inlet ducts, it is estimated that the removal, cleaning, and replacement of the affected perforated plates (screens) will take two people approximately 1 hour. The dose rate at this location is estimated to be 58 mrem/hr. The total exposure for personnel to perform these corrective actions is 0.116 man-rem.

8.1.5 CASK DROP LESS THAN ALLOWABLE HEIGHT

Cask drops outside the fuel handling building/auxiliary building (FHB/AB) are not credible due to the design of the cask transporter and LPT, as discussed in Section 8.2.4. The structural load path members of the cask transporter used in Diablo Canyon ISFSI operations are designed, operated, fabricated, tested, inspected, and maintained in accordance with the applicable guidelines of NUREG-0612 (Reference 6). The LPT has been designed to preclude tipover or drops of the transfer cask (Reference 12). In addition, an evaluation was performed for four short-term transient lifting and lowering activities during the transport operation where the cask transporter may not maintain its complete seismic capability. These involve:

(1) lifting or lowering the loaded HI-TRAC transfer cask between its bolted configuration on the LPT and its transport configuration on the transporter;

(2) lifting or lowering the loaded HI-TRAC transfer cask between the transport configuration on the transporter and its bolted configuration on the mating device at the CTF;

(3) lifting or lowering the HI-STORM overpack between the transport configuration on the transporter and entry into the CTF shell; and

(4) lifting or lowering the HI-STORM overpack between the transport configuration on the transporter and the anchored configuration on the ISFSI pad

This evaluation shows that based on the minimal height of the lifts and the duration of these activities, the probability of a design basis event during those lifts is not credible. Therefore, a drop of the loaded MPC during inter-cask transfer operations is not a credible event.

Inside the FHB/AB the cask and any ancillary components are lifted, handled, and moved in accordance with DCPP procedures and the DCPP Control of Heavy Loads Program, as applicable, which provides assurance of safe heavy load handling. In addition the LPT has been designed to preclude tipover or drops of the transfer cask (Reference 12).

8.1.6 LOSS OF POWER

Electric and pneumatic power supplies are used at the CTF during MPC and cask handling activities. A loss of power is postulated to occur as a result of the failure of the electric and pneumatic power supplies supplying the ISFSI storage site. A loss of power does not affect the cask handling or the cask transporter because all active functions of the transporter, such as cask lifting and MPC downloading, are driven from the onboard transporter diesel engine. Section 8.1.7 discusses the cask transporter off normal operation.

8.1.6.1 Postulated Cause of Loss of Power

Loss of the power supplies may occur as the result of natural phenomena, such as lightning strike or high winds. Loss of electrical power may also result from an electrical system fault in the power supplies.

8.1.6.2 Detection of Loss of Power

Loss of electrical or pneumatic power will be detected by the failure of systems that require the power supplies.

8.1.6.3 Analysis of Effects and Consequences of Loss of Power

8.1.6.3.1 ISFSI Storage Site

There is no effect on the ability of the HI-STORM 100 System to safely continue storing the spent fuel at the ISFSI storage site during a loss of power event because the dry storage system is a completely passive design. No electric or pneumatic-powered equipment is used with the storage overpack while it is in its storage configuration on the concrete storage pads.

8.1.6.3.2 Mating Device

The mating device hydraulic and airbag systems require electric and pneumatic power to operate. If the mating device hydraulic system is in operation at the time of loss of electrical power, the hydraulic system can be operated manually. If the mating device airbag system is in operation at the time of loss of pneumatic power, removal of the MPC from the transfer cask will be restricted until the system is returned to service. Maintaining the MPC in the transfer cask is considered the safest condition. There are no functions at the CTF related to safe operation of the ISFSI that are electrically powered. All lifting of the overpack and transfer cask are performed by the transporter, which is powered by an onboard diesel engine. The transporter also performs the downloading of the MPC into the overpack. As a result, there is no effect from a loss of power.

8.1.6.4 Corrective Action for Loss of Power

The corrective action following a loss of power to the mating device hydraulic system includes manually operating the hydraulic system. The corrective action following a loss of pneumatic power to the mating device air bag system includes maintaining the MPC within the transfer cask, which is considered the safest condition.

8.1.6.5 Radiological Impact of Loss of Power

The off-normal event of loss of power has no radiological impact because the MPC confinement barrier is not breached and shielding is not affected. The transfer cask is designed to provide adequate shielding and decay heat removal from the canisters. The sides of the transfer cask have both gamma and neutron shields, and the bottom lid is designed to prevent excessive dose rates below the transfer casks. In the event the transfer operation is interrupted due to a loss of power, operators would take measures as necessary to assure adequate distance and/or additional shielding between themselves and the transfer cask to minimize doses until power is restored and the transfer process can resume.

8.1.7 CASK TRANSPORTER OFF-NORMAL OPERATION

Off-normal operation of the cask transporter includes postulation of the following human performance and active component failures during transport of the loaded transfer cask and the loaded overpack:

- Driver error
- Driver incapacitation
- Transporter engine failure
- Loss of hydraulic fluid

8.1.7.1 Postulated Cause of Cask Transporter Off-Normal Operation

Cask transporter driver error may be caused by driver inattentiveness, poor visibility, incorrect instructions, poor training, or any of several human performance-related causal factors. Driver incapacitation would be most likely caused by a sudden medical emergency. Transporter engine failure may be caused by a variety of mechanical problems typical of combustion engines. A loss of hydraulic fluid may be caused by a leak anywhere in the hydraulic system.

8.1.7.2 Detection of Cask Transporter Off-Normal Operation

Driver error or driver incapacitation would be detected by the support staff walking along with the transporter on the transport route observing the driver in distress or erratic transporter motion. Transporter engine failure would be detected by the halt of any engine-driven activity taking place at the time. A hydraulic fluid leak would be detected by the pressure instrumentation in the hydraulic system and possibly by visual observation of leaking fluid.

8.1.7.3 Analysis of Effects and Consequences of Cask Transporter Off-Normal Operation

In addition to the transporter driver, transport operations are conducted with a support team consisting of security and other personnel affiliated with the fuel movement walking along with the transporter to ensure a safe and efficient move of the loaded cask from its point of origin to its destination. These personnel observe the movement of the transporter to ensure the designated travel path is being followed. Should the transporter start to veer from the travel path, the transporter will be stopped (either by the driver or by a support team member using either of two external stop switches mounted on the outside of the transporter), the cause investigated, and corrective actions taken to get the vehicle back on the correct path.

Incapacitation of the driver is addressed by the design of an automatic shutoff control where the vehicle will stop whenever the control is released. The same control is used to move the transporter vehicle and operate the cask lifting apparatus integral to the transporter. A selector switch is used to ensure only one function can be performed by the transporter at a time. Also, either of two emergency stop switches, mounted on the outside of the transporter, can be operated to stop the transporter.

A loss of hydraulic pressure (e.g. due to transporter engine failure) or loss of electrical power results in automatic application of the brakes (if the vehicle is moving) or stoppage of load movement and engagement of the mechanical locks (if lift operations are in progress). Once the transporter engine is operating again or hydraulic pressure restored, the transporter controls are designed to require the operator to re-engage the controls to allow resumption of any function of the transporter.

A loss of hydraulic fluid causes a loss of pressure in the hydraulic system that engages the hydraulic brakes and stop movement of the lifting apparatus. Once the hydraulic system is operating again, the transporter controls are designed to require the operator to re-engage the controls to allow resumption of any function of the hydraulic system.

8.1.7.4 Corrective Action for Cask Transporter Off-Normal Operation

The corrective action for cask transporter off-normal operation will be developed and implemented based on the nature and safety significance of the problem. Corrective actions may include additional training for the driver, replacement of the driver, improved operating procedures, and repair or replacement of failed mechanical parts.

The transporter is designed “fail-safe” to preclude uncontrolled lowering of the loaded transfer cask or overpack if a failure of an active component occurs, so during the transport operation to the CTF no corrective actions related to the cask are necessary. If necessary, cribbing could be used to support the loaded transfer cask or overpack if the transporter needs to be replaced or detached from the load for repairs.

Corrective actions following a transporter engine failure or a loss of hydraulics during the lifting functions at the CTF may vary widely, depending on the cause of the failure or loss. Restoration activities are generally straightforward. If there is an engine failure or hydraulics loss at the CTF while the loaded overpack is below grade, it can remain that way indefinitely. If the failure occurs during the MPC transfer it is desirable to transfer the MPC back into the transfer cask or complete the transfer to the overpack. If time permits, the first option would be to repair the transporter. As an alternative, a skid-mounted backup power unit will be available at the CTF site to actuate the transporter towers lifting function. The backup unit is diesel powered with a 2.5 gallon fuel tank. These lifting functions may also be accomplished using a mobile crane, which would meet the requirements of Section 4.3.4.b of the DC ISFSI TS.

8.1.7.5 Radiological Impact of Cask Transporter Off-Normal Operation

The cask transporter off-normal event has no radiological impact since the confinement barrier is not breached and shielding is not affected.

The transfer cask is designed to provide adequate shielding and decay heat removal from the canisters. The sides of the transfer cask have both gamma and neutron shields, and the combination of the bottom lid and bottom shield are designed to prevent excessive dose rates below the transfer casks. In the event the transfer operation is interrupted due to a loss of the transporter engine or hydraulics, operators would take measures as necessary to assure adequate distance and/or additional shielding between themselves and the transfer cask to minimize doses until transporter function is restored and the transfer process can resume.

The loaded overpack is designed to provide adequate shielding and decay heat removal from the canisters for long term storage. In the event the transfer operation is interrupted due to a loss of the transporter engine or hydraulics and the loaded overpack must remain at the CTF, no additional measures would be necessary other than those taken while the loaded overpack in on the ISFSI pad.

8.1.8 REFERENCES

1. ANSI/ANS 57.9-1992, Design Criteria for an Independent Spent Fuel Storage Installation (dry type), American National Standards Institute.

2. Standard Review Plan for Dry Cask Storage Systems, USNRC, NUREG-1536, January 1997.

3. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 1A, January 2003.

4. Deleted in Revision 2.

5. Normal, Off-Normal, and Hypothetical Dose Estimate Calculations, USNRC, Interim Staff Guidance Document-5, May 2000.

6. Control of Heavy Loads at Nuclear Power Plants, USNRC, NUREG-0612, July 1980.

7. PG&E Calculation STA-140 (HI-2002513, Rev. 7), “Diablo Canyon ISFSI Site Boundary Confinement Analysis.”

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

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

10. Holtec International HI-STORM 100 Cask System Certificate of Compliance Number 1014, Amendment 1 dated 7/15/02

11. Holtec International Report No. HI-2053376, “Thermal-Hydraulic Analysis for Diablo Canyon Site-Specific HI-STORM System Design,” Revision 7.

12. Holtec International Report No. HI-2053390, “Structural Evaluation of the Low Profile Transporter,” Revision 4.

13. Holtec International Document No. HI-2104625, “Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System Design”, Revision 10.

14. Holtec International Document No. HI-2125191, “Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System with up to 28.74 kW Decay Heat”, Revision 6.

8.2 ACCIDENTS

As discussed in Section 1.1 the licensed HI-STORM 100 System at the Diablo Canyon ISFSI has been modified to facilitate the first fuel-loading campaign. These modifications were performed in accordance with 10 CFR 72.48 and 10 CFR 50.59 and involve the MPC-32 canister, HI-TRAC 125D transfer cask, HI-STORM 100SA overpack, CTF, low profile transporter, cask transporter, and auxiliary components used in the loading and transport to the ISFSI facility. The originally-licensed MPC-24s will likely not be used at the ISFSI and would require modifications, analyses and associated 10 CFR 72.48 evaluations similar to the MPC-32 prior to their use. Most of the accident and off-normal analyses and evaluations performed for the licensed HI-STORM 100 System remain bounding for the modified system. However, in cases where they do not and a re-analysis or site specific analysis was required, those analyses are identified and referenced in their related sections below.

8.2.1 EARTHQUAKE

An earthquake is classified as a natural phenomenon Design Event IV as defined in ANSI/ANS-57.9 (Reference 1). The effects of seismic events on cask loading operations inside the fuel handling building/auxiliary building (FHB/AB) are discussed in the 10 CFR 50 license amendment request (Reference 53) and license amendments (Reference 54), both of which support the Diablo Canyon ISFSI license. This section addresses the effect of a seismic event on the operations related to the Diablo Canyon ISFSI that occurs outside the FHB/AB. Cask handling activities outside the FHB/AB were reviewed to identify potential risk significant configurations during a seismic event. The seismic evaluations address the following potentially seismic risk significant configurations (all configurations are analyzed with an MPC loaded with spent fuel):

(1) HI-TRAC transfer cask suspended vertically from the cask transporter on the transport route between the FHB/AB and the cask transfer facility (CTF).

(2) HI-TRAC transfer cask suspended vertically from the cask transporter at the CTF, prior to being placed atop the HI-STORM 100SA overpack.

(3) HI-TRAC transfer cask mounted atop the HI-STORM 100SA overpack at the CTF and the transporter restrained to the ground. The overpack is sitting on the CTF baseplate.

(4) HI-STORM 100SA overpack being transported to the ISFSI storage pad, suspended vertically from the cask transporter. In terms of seismic stability, this configuration bounds configuration (2) because the HI-STORM 100SA overpack is heavier than the HI-TRAC transfer cask.

(5) HI-STORM 100SA overpack anchored to the ISFSI storage pad in its long-term storage configuration.

Additionally, the slopes above the ISFSI and transport route were analyzed for stability during a seismic event (see Section 2.6.5).

8.2.1.1 Cause of Accident

Earthquakes are natural phenomena caused by the movement of large geological plates under the earth's surface.

8.2.1.2 Earthquake Accident Analysis

Two methods were used for seismic analysis of SSCs, that is, equivalent static analysis load method and dynamic analysis method. These methods were used as follows:

Equivalent Static Analysis Method

(1) Design of CTF reinforced concrete support structure.

(2) Pad design.

(3) Design of CTF shell structural steel.

Dynamic Analysis Method

(1) Determination of slope stability.

(2) Determination of transporter stability while carrying a transfer cask or loaded overpack.

(3) Determination of ISFSI storage pad sliding.

(4) Design of storage cask anchorage to the pad.

(5) Determination of low profile transporter (LPT) stability while carrying a transfer cask.

As discussed in Section 2.6.2.2, the design earthquake (DE), double-design earthquake (DDE), Hosgri earthquake (HE) and Long Term Seismic Program (LTSP) earthquakes are the DCPP seismic licensing basis. The DE and DDE spectra are defined for periods up to 1 second. The Hosgri spectra are defined for periods up to 0.8 seconds. The LTSP spectra are defined for periods up to 2 seconds.

The statistically independent free-field DE, HE and LTSP ground acceleration time histories in two horizontal and vertical directions were regenerated and updated based on the free-field response spectra and time histories from strong ground motion recorded at the Lucerne Valley site from the June 28, 1992 Landers magnitude 7.3 earthquake and from a rock site located approximately 8 km fault rupture distance from the September 20, 1999 Chi Chi magnitude 7.6 earthquake. These time histories are referred in this FSAR as the DE, DDE, HE and LTSP time histories. The DDE is twice the DE. The regenerated DE, DDE, HE and LTSP free-field time histories meet the NRC Standard Review Plan (SRP) spectral matching criteria, Section 3.7.1 of NUREG-0800, (Reference 2) and the 3 components of the time-histories for each earthquake were verified to be statistically independent in accordance with ASCE 4-86 (Reference 3). The spectra generated from the time-histories were compared to existing DCPP DE, DDE, HE, and LTSP ground spectra. The regenerated DE, DDE, HE, and LTSP time histories were used in the seismic time history analysis of the cask anchorage; since the storage cask is anchored to the ISFSI storage pad and long period energy has a negligible impact on the analysis results.

As discussed in Section 2.6.2, PG&E developed the ISFSI Long Period (ILP) earthquake spectra to be used for the analyses of transporter stability, slope stability and ISFSI storage pad sliding to provide extra design margin since these analyses' results could be affected by long period energy. The ILP are 84th percentile spectras at damping values of 2 percent, 4 percent, 5 percent and 7 percent for the horizontal and vertical components that extend out to 10 seconds and which include near fault effects of directivity and fling. The ILP spectra envelops the DDE spectra at 2 percent and 5 percent damping, the Hosgri spectra at 4 percent, 5 percent, and 7 percent damping, and the LTSP spectra at 5 percent damping. Five sets of spectrum compatible time histories were generated from recordings of large magnitude earthquakes (M>6.7) recorded at short distances (<15 km from the fault), and they contain a range of characteristics of the near fault effects.

The modal damping ratios expressed as a percentage of critical damping for the seismic analyses are provided in Table 8.2-1. These damping values are from the DCPP FSAR Update (Reference 4). The analysis approach, results, and conclusions for each of the configurations are discussed separately below.

8.2.1.2.1 Seismic Evaluation of Operations Involving the Cask Transporter - Seismic Configurations 1, 2 and 4

This section discusses the seismic stability evaluation of the spent fuel cask transporter used at the Diablo Canyon ISFSI.

The HI-TRAC transfer cask, containing a loaded MPC, exits the FHB/AB on the LPT in a vertical orientation to a position just outside the FHB/AB. Analysis (Reference 60) demonstrates that during the design basis seismic events, overturning of the loaded transfer cask attached to the LPT is not credible under all scenarios. It is also shown that the LPT meets structural integrity limits when carrying a loaded transfer cask from the FHB/AB to the staging area where the transfer cask is transferred to the cask transporter. Following the LPT delivering the transfer cask to the cask transporter staging area, the transfer cask is connected to the cask transporter using lift links. The cask transporter lifts the transfer cask to the transport position, a few inches above the LPT. Securing the transfer cask by a seismic strap, the transporter moves the transfer cask in a vertical orientation along the road approximately 1.2 miles to the ISFSI storage site, in the process traversing an 8.5 percent (nominal) grade decline and climbing an 8 percent grade (for approximately 600 ft) and then an approximate 6 to 8 percent grade (for approximately 3,000 ft). Once at the CTF, the seismic strap is removed from the transfer cask and the transporter lifts the transfer cask about 5 feet above the ground. The transfer cask is aligned directly above the mating device, lowered onto the mating device and secured. After the transfer cask is secured onto the mating device the MPC is transferred by the transporter into the overpack located in the CTF. After the MPC transfer operation is executed at the CTF, the cask transporter lifts the loaded overpack out of the CTF and carries the loaded overpack in a vertical orientation to its final position on the ISFSI storage pad. During this transport a seismic strap is secured to the overpack.

An evaluation (Reference 62) was performed for four short term transient lifting activities during the transport operation where the cask transporter may not maintain its complete seismic capability. These involve:

(1) lifting or lowering the loaded HI-TRAC transfer cask between its bolted configuration on the LPT and its transport configuration on the transporter;

(2) lifting or lowering the loaded HI-TRAC transfer cask between the transport configuration on the transporter and its bolted configuration on the mating device at the CTF;

(3) lifting or lowering the HI-STORM overpack between the transport configuration on the transporter and entry into the CTF shell; and

(4) lifting or lowering the HI-STORM overpack between the transport configuration on the transporter and the anchored configuration on the ISFSI pad

This evaluation shows that based on the minimal height of the lifts and the duration of these activities, the probability of a design basis event during those lifts is not credible.

The transport route is approximately 1.2 miles long, approximately one third on bedrock and the remaining two thirds crossing surficial deposits over bedrock (Section 2.6.2.6). Because the transport route is about the same distance from the Hosgri fault zone as the DCPP and the ISFSI sites, the ILP spectra are appropriate for use along the transport route where the route is constructed on bedrock and where the transport route crosses surficial deposits over bedrock (approximately two-thirds of the route) as described in Reference 50. Seismic stability analyses of the transporter on the road are provided in References 31 and 46. A transporter stability analysis (Reference 31), described below, was performed for ground acceleration associated with the ILP earthquake. The analysis determined that the transporter would not overturn or leave the roadway (Configurations 1, 2, and 4, and a portion of Configuration 1).

PG&E also performed a seismic evaluation of the cask transporter with the cask in a horizontal configuration under ground accelerations twice those of the ILP earthquake accelerations to account for any amplification due to surficial deposits over bedrock beneath the road. This evaluation is documented in Reference 46 and demonstrates that even under these hypothetical conditions, the cask transporter would remain stable and would not overturn or leave the roadway. Reference 46 determined that this analysis is bounding for carrying a cask in a vertical orientation.

PG&E evaluated the risk of an earthquake causing ground accelerations twice those of the ILP occurring simultaneously with cask transport activities (12 hours per year) and concluded that the risk is not credible (less than 10^{-7}) as described in Reference 51.

In summary, the transporter remains stable under seismic conditions while the transporter is traversing the portions of the route over bedrock and the surficial deposits over bedrock. In addition, Reference 46 determined that for ground accelerations twice those of the ILP demonstrate that the transporter remains stable and the risk of such a ground acceleration occurring during cask transport is not credible (less than 10^{-7}).

Methodology

The ILP seismic events for the Diablo Canyon ISFSI, described in Section 8.2.1.2, were evaluated and analyzed for the transporter stability analysis. Five sets of ILP time-histories were used to demonstrate transporter stability as it carries a loaded cask on the transport route. As discussed in Section 2.6.2.6, the ILP spectra and associated time histories are appropriate for use along the transport route.

Visual Nastran 4-D (VN) (formerly Working Model 4-D) (Reference 5) serves as the simulation engine to obtain the response to the 3-dimensional seismic events. This computer code has previously been used in licensing the HI-STORM 100 System as described in the HI-STORM 100 System FSAR (Reference 6).

The time-domain dynamic simulations model the cask transporter, the HI-STORM 100SA overpack, the HI-TRAC transfer cask, the MPC (including the fuel basket, fuel, and lid), and the cask lids as rigid bodies. The mass of the MPC and the contained spent fuel is lumped in a free-standing rigid cylinder that, during the earthquake, is free to rattle in the cask cavity.

The cask transporter sits on grade that is subjected to a ground acceleration time history appropriate to the free field ILP event. The simulations use the Holtec generic model of the cask transporter with a track length and width identical to that planned for the Diablo Canyon cask transporter.

Acceptance Criteria

The cask transporter plus its carried load must remain stable (not overturn) and remain on the travel path under all seismic events applicable to the Diablo Canyon ISFSI site.

The minimum roadway width is 22 ft, which sets the allowable transporter lateral sliding distance. The maximum acceptable sliding movement along the roadway is limited to the DCPP cask transporter track length to ensure that the transporter will remain on the roadway after exiting a turn in the roadway.

Assumptions

The following key assumptions were employed to construct the models for the simulations:

(1) The time domain dynamic analyses of the transporter seismic stability simulate the modeled components (cask transporter, transfer cask, overpack and MPC) as rigid bodies with specified geometry and bounding mass. The connections between the cask body and the lids were assumed to be rigid. These are conservative assumptions for the seismic analysis since energy dissipation in the dynamic system is neglected by virtue of the rigid body modeling.

(2) The time domain dynamic simulations model the MPC and the contained fuel by a solid cylinder with total mass that bounds the heaviest PWR MPC-32 (90,000 lb). This is conservative since all energy dissipation due to fuel assembly rattling inside the MPC is neglected and any reduction in amplitude due to chaotic fuel assembly motion over time is ignored.

(3) The analyses in time domain are simplified by assuming the rigid bodies to have uniform mass density when calculating their mass moments of inertia and mass center locations. Any shift in the centroid due to this assumption has a negligible effect on the results of the analysis.

(4) The coefficient of restitution for the internal contact surfaces (MPC/overpack) is set to zero. The coefficient of restitution between the transporter treads and the ground was set to 0.0 - 0.25 (the exact value has no influence on the solution when sliding motions predominate). For the coefficient of friction at the transporter tread/ground interface, an upper bound value of 0.8 was conservatively assumed to emphasize tipping action. A lower bound value for the tread/roadway surface of 0.4 was assumed to determine the sliding behavior of the transporter. The coefficient of friction between the MPC and the HI-TRAC transfer cask cavity side surfaces is set at 0.5. This is realistic because experience indicates a variation from 0.8 down to 0.2 for steel-on-steel depending on the relative velocity between the two surfaces.

(5) The time domain dynamic simulations use a generic model of the cask transporter with a track length that is the same as the length of the Diablo Canyon cask transporter tracks. The analyses considered the stability of the cask transporter when supported by a horizontal ground surface.

(6) In all stability analyses, the positioning of the cask in the cask transporter is set slightly higher than the anticipated carry height to ensure that overturning moments are conservatively computed at each time point during the dynamic simulations.

(7) All bodies are assumed to be rigid for the global analysis. The cask transporter design specification includes a requirement that the transporter be designed such that its lowest global natural frequency is in the rigid range (>33 Hz).

Key Input Data

The key input data used in the cask transporter seismic analyses are shown in Tables 8.2-2 through 8.2-4.

Input time histories used for the dynamic simulations are five sets of ILP design earthquake excitations. These seismic events are identified below with their duration:

Set 1: Lucerne Valley (48 sec)
Set 2a: Yarimca (40 sec)
Set 3: LGPC (22 sec)
Set 5: El Centro (40 sec)
Set 6: Saratoga (40 sec)

Results of Analyses

A series of nonlinear dynamic simulations were performed using the Visual Nastran 4-D computer code to assess the seismic stability of the cask transporter during the five ILP earthquakes. Table 8.2-5 lists the simulations performed for the stability evaluation. The combinations of grade, coefficient of friction, and seismic events have been chosen to be bounding for the site-specific conditions.

For each case considered, the loaded transporter was assumed to be on a flat or inclined surface with specified coefficients of friction. The simulations performed under Phase 1 serve to identify potentially bounding events from among the five candidate time histories. The choice of simulations for the remaining phases was based on the results from the simulations in Phase 1. The combination of grade and coefficient of friction were chosen to induce sliding as opposed to tipping.

Table 8.2-6 summarizes the estimates of the maximum transporter horizontal excursions in the transverse and longitudinal direction for each phase of the dynamic simulations performed for the cask being carried in a horizontal configuration. Table 8.2-6 results remain bounding for carrying a cask in a vertical orientation. The reported maximum excursions are at the top of the transporter relative to the ground.

These results are bounding for all Diablo Canyon cask transporter operational modes and for all ILP earthquakes. The maximum value of 10.7 inches reported for the transverse excursion with a coefficient of friction of 0.4 demonstrates that in the event of seismic excitation, the transporter will not leave the road while moving from the FHB/AB to the Diablo Canyon CTF or while moving from the CTF to the ISFSI. The small relative movements reported for the case with friction coefficient of 0.8 demonstrate that overturning of the loaded cask transporter is not a credible event under the ILP seismic events. For the case where the transporter is on the 8.5 percent grade when the seismic event is postulated to occur, the results demonstrate that, the maximum sliding movement along the axis of the road (30.2 inches) is less than one transporter track length. In addition, the transverse movement of the transporter during a seismic event is small, 10.7 inches, compared to the distance between the edge of the transporter and the edge of the roadway (roadway minimum width is 22 ft and the width of the transporter from outside of track to outside of track is approximately 18 ft), provides additional margin of safety.

The time domain dynamic simulations of the cask transporter demonstrate that the cask transporter, carrying either a loaded HI-TRAC transfer cask in the horizontal orientation or a loaded HI-STORM 100SA overpack in the vertical orientation, will not overturn during a seismic event and will not leave the road while moving from the FHB/AB to the CTF or from the CTF to the storage pads. When the transporter is carrying a HI-TRAC horizontally, up or down the 8.5 percent grade, the magnitude of sliding displacement along the axis of the road is less than the length of the transporter track. These results for carrying the transfer cask in the horizontal orientation are bounding for carrying the transfer cask in the vertical orientation.

Cask drop during transport (seismic)

As discussed in Section 8.2.4, the load path portions of the cask transporter and the lifting devices attached to the cask components will be designed to preclude drop events, either through redundancy or enhanced safety factors. The design includes consideration of seismic loads. Therefore, a seismic event occurring during transport would not result in a cask drop.

8.2.1.2.2 Seismic Analysis of Cask Transfer Facility Seismic Configuration 3

8.2.1.2.2.1 CTF Steel Structure

The CTF at the Diablo Canyon ISFSI is used in conjunction with the cask transporter to perform MPC transfers between the HI-TRAC transfer cask and the HI-STORM 100SA overpack. Prior to the transfer operation, the empty HI-STORM 100SA overpack is placed in the CTF. The overpack is sitting on the CTF baseplate and a mating device is installed on the top of the overpack. This mating device serves as a structural connection and an alignment device between the top of the overpack and the bottom of the HI-TRAC transfer cask. The transfer cask is positioned over the overpack by the cask transporter, which remains in position during the transfer operation. Restraints are used to secure the cask transporter to ground during the MPC transfer operation.

The cask transfer facility is shown in Figure 4.4-3 and includes the following main structural components:

Main Shell – A cylindrical shell is positioned into a larger vertical hole in the rock with concrete backfill providing an interface connection with the rock walls of the hole. The bottom of the shell is anchored to a reinforced concrete base. This cylindrical shell serves as the cavity liner into which the overpack is lowered. Wedge assemblies are installed at the top and bottom of the shell, which serve to restrain the cask under lateral loads from seismic events (References 32 and 59).

Reinforced Concrete Support Structure -The CTF steel structure is placed on a steel reinforced concrete foundation slab and surrounded by heavily reinforced concrete up to the surface. The concrete structure carries all the compressive loadings on the base and the side-walls (cylindrical in shape) to the ground rock. The structure has a gravity fed sump for drainage.

Cask Transporter Lateral Restraint System - The cask transporter lateral restraint system is designed to apply external restraint loading to the cask transporter structure. The structural components of the transporter resisting the restraint loads are designed to the applicable limits of ASME Section III, Subsection NF including Appendix F. As discussed in References 47 and 52, the restraints are steel struts or similar equipment suitable sized to restrain the transporter by transferring the restraint loading to the ground adjacent to the CTF support structure. The restraints are designed to meet the stress limits of ASME Section III, Subsection NF including Appendix F. The surface-level, in-ground portion of the restraints (seismic tie downs) are designed in accordance with ACI 349-97 (Reference 10) and Appendix B to ACI 349-01 (Reference 55), as endorsed by Regulatory Guides 1.142 (Reference 56) and 1.199 (Reference 57), respectively. These tie downs are installed in accordance with ACI 349-01. The tie-downs are supported by rock anchor installations into the ground. Holtec Licensing Drawing 4480, showing the CTF shell structure, is provided as Figure 4.4-3.

The next section discusses the seismic structural analyses and evaluations of the CTF at the Diablo Canyon ISFSI. The calculations provide the loads on the CTF base, CTF shell, and surrounding concrete under the specified ASME Section III (Reference 7), Subsection NF service (Level A and Level B) load conditions and Appendix F seismic (Level D) load conditions. A description of the analysis of the reinforced concrete support structure is also included.

<u>Methodology- Structural Analysis</u>

The analysis (Reference 32) evaluates the capacity of the CTF structural components under static loads (dead weight and factored dead load) and under static plus seismic and wind loads. Bounding values for the weights of the spent fuel casks and canisters are used to evaluate the dead loads applied on the CTF structure. Quasi-static stability analyses provide the magnitudes of the seismic loads on the CTF steel structure during the governing LTSP earthquake excitation. The natural frequencies of the cask transporter, the HI-TRAC transfer cask, and the HI-STORM 100SA overpack stack were calculated. The actual horizontal spectral acceleration value corresponding to 19.85 Hz was used in the seismic analysis. Under vertical excitation, the ground vertical zero period acceleration value was used in the seismic analysis since the stacked configuration is rigid in the vertical direction. Examination of the response spectra for the four DCPP seismic events (DE, DDE, HE and LTSP) shows that the bounding spectral accelerations for CTF structural design are those from the LTSP spectra.

The analysis considers the most critical combinations of design loads for the loading scenario wherein a loaded HI-TRAC transfer cask is stacked on top of the HI-STORM 100SA overpack sitting on the CTF baseplate (Configuration 3).

The seismic analysis considers two critical combinations of the specified design earthquake components when the CTF structure is subjected either to upward vertical inertia forces or downward vertical inertia forces. The Newmark 100-40-40 Method is used to combine the 3 specified directions of the seismic load.

Using the calculated inertia loadings together with known dead loading, strength-of-materials solutions from the theory of elasticity are used to determine the stresses in the CTF structural components and weld connections. The ratio of the allowable stresses to the calculated stresses in the components and welds defines safety factors for service (Level A) and seismic (Level B and Level D) load conditions.

In addition to the above analysis, an impact analysis on the CTF due to the lateral loads of the HI-TRAC/HI-STORM stack under an LTSP event was performed. That analysis found that all loads and stresses remained acceptable and all safety factors remained above 1.0 (Reference 59).

<u>**Acceptance Criteria**</u>

The stresses in the CTF structural components and welded connections under the service loads must be below the limits prescribed in ASME Section III, Subsection NF (Level A and Level B). The stresses in the CTF structural components and welded connections under the combination of dead plus seismic loads must be below the limits prescribed by ASME Section III, Appendix F (Level D).

The seismic connectors at the CTF (cask transporter to ground, and between the transfer cask and the overpack) must have sufficient structural capacity to prevent extensive motions of the transfer cask during MPC transfer operations that could put the contained fuel at risk. The load capacity of all necessary connectors is designed to meet the applicable limits of ASME Section III, Subsection NF and Appendix F.

Assumptions

The following conservative assumptions are employed in the linear elastic structural analyses:

- The stability analysis of the CTF shell extensions conservatively neglects any contributory stiffening from the main shell and ignores the support from the concrete fill between the shell and the rock walls.

Key Input Data

The key input data used in the CTF seismic analyses are shown in Table 8.2-7. The seismic inputs for the analyses are obtained from ground acceleration response spectra for DCPP. The ZPAs for the vertical direction were used because the stacked casks in the CTF are rigid (>33 Hz) in the vertical direction. The spectral accelerations in the horizontal directions corresponding to 19.85 Hz were used. The ZPAs and spectral accelerations used in the analysis are shown in Table 8.2-8. Where load combinations are required for the strength evaluation, the Newmark 100-40-40 Method (for LTSP seismic event) is used to combine the three specific directions of the seismic load.

Results of Analyses

The results from the CTF structural analyses demonstrate that all structural members and welds stresses satisfy the condition that safety factors are greater than 1.0. Safety factors are defined as:

SF= (Allowable stress or load)/(Calculated stress or load).

In addition to the structural analysis of the CTF components, mandated by the appropriate design codes, analyses of the connector restraints (that inhibit relative movements between the cask transporter and ground) and the mating device (between the transfer cask and the overpack) are also performed to ensure that any relative motion between the transfer cask and the overpack during the cask transfer operation will not compromise the integrity of the MPC. Load/stress limits on these ancillary items meet applicable requirements of Subsection NF and Appendix F. An additional analysis (Reference 58) was performed for the cask stack-up including the transfer cask, mating device and overpack while in the CTF. It was determined that all load/stress limits on these components and their connections continue to meet all applicable requirements. Based on that stack up analysis an impact analysis on the CTF due to the lateral loads of the HI-TRAC/HI-STORM stack under an LTSP event was performed. The LTSP accelerations bound all other accelerations for the applicable design basis earthquakes. That analysis found that all loads and stresses remained acceptable and all safety factors remained above 1.0 (Reference 59).

8.2.1.2.2.2 CTF Reinforced Concrete Support Structure

Methodology - Structural Design/Analysis

A static analysis (Reference 30) was performed to appropriately size the base slab and the side cylindrical wall to accommodate the applied forces generated by the CTF as discussed in 8.2.1.2.2.1.

Acceptance Criteria

ACI-349 97 and Appendix B to ACI 349-01, in compliance with NUREG-1536 (Reference 13), concrete stress allowables, and Regulatory Guides 1.142 and 1.199, as applicable, are used.

Assumptions

None

Key Input Data

The surrounding rock properties and the functional requirements of the CTF steel structure (as described earlier in this section) and the loads developed in the CTF analysis (Section 8.2.1.2.2) are the key input parameters.

Results

The reinforced concrete structure meets the functional requirements of the facility and stress requirements of ACI 349-97, as endorsed by Regulatory Guide 1.142.

8.2.1.2.3 Seismic Analyses of the HI-STORM 100SA Overpack Anchored to the ISFSI Storage Pad in its Long-Term Storage Configuration Seismic Configuration 5

8.2.1.2.3.1 Cask and Anchorage Seismic Analysis

The HI-STORM 100SA overpack design differs from the HI-STORM 100S only in that it includes an extended bottom flange and gussets that enhance the structural resistance of the flange/shell around the bottom periphery of the overpack (see Figure 4.2-7). This flange includes a bolt circle to permit structural “mating” of the overpack to the ISFSI storage pad steel embedment plate by 16, 2-inch diameter, SA193-B7 preloaded cask anchor studs. The preloaded cask anchor studs are threaded into compression/coupling blocks to ensure a continuous compressive state of stress at the interface between the lower surface of the HI-STORM 100SA overpack and the top surface of the embedment plate. The continued contact ensures development of interface friction forces sufficient to resist lateral movement of the overpack base relative to the embedment plate. It also ensures that the ISFSI storage pad embedment structure provides the resisting moment to stabilize the system under seismic loading. The cask anchor studs are threaded into compression/coupling blocks that bear against the lower surface of the embedment plate from the action of the preload. The embedment plate is held to the concrete by 16 longer embedment anchor rods that are threaded into the same compression/coupling blocks, but are not preloaded. The embedment anchor rods are only loaded, as the seismic event proceeds, to the extent necessary to maintain vertical force and moment equilibrium. Oscillations in the cask anchor stud load are minimized due to the presence of the initial preload. Figure 4.2-2 shows a section depicting the embedment plate, the compression block, the cask anchor studs and the embedment anchor rods. The cask is not shown in this figure.

The cask and anchorage seismic analyses are not sensitive to long period ground motion. Therefore, these analyses (Reference 38) were performed using the four DCPP seismic events (DE, DDE, HE, and LTSP). The DE, DDE, HE, and LTSP are characterized by free-field acceleration time-histories, in each of 3 orthogonal directions, with durations of 41 seconds for the DE and DDE cases and 48 seconds for the HE and LTSP cases. The HE and LTSP events have the highest, zero-period accelerations, and the largest, free-field excursions. Therefore, the results from these events are bounding and the dynamic simulations to obtain time-history behavior of the system are performed using the Visual Nastran (VN) simulation code described previously only for these two events.

An alternate analysis of the anchorage, cask and storage pad was performed to determine the performance of these SSCs assuming a partial and a complete loss of anchor stud pre-tension (Reference 65). The methodology used was similar to that of the ISFSI pad sliding dynamic analysis in that the low frequency effects of the ILP spectra were applied.

<u>Methodology</u>

The dynamic model of the HI-STORM 100SA overpack in VN consists of the following major components:

(1) The HI-STORM 100SA overpack plus the embedment plate is modeled as a six degree-of-freedom (rigid body) component.

(2) The loaded MPC is also modeled as a six degree-of-freedom (rigid body) component that is free to rattle inside the overpack shell. Gaps between the two bodies reflect the nominal dimensions from the design drawings in Reference 12.

(3) The embedment anchor rods provide the vertical connection between the embedment plate and ISFSI slab. The embedment anchor rods are modeled as individual linear springs connecting the periphery of the extended baseplate to the ISFSI storage pad section. The concrete

pad/embedment compression resistance at the interface is simulated with compression-only stiffness elements around the periphery.

(4) For the global dynamic analysis of the anchored cask, the slab section under the cask is assumed rigid and the three components of acceleration and time-history are applied simultaneously at the base of the slab. Since the HE and LTSP events provide the bounding loads to the anchorage, the importance of directional effects on the responses is evaluated for both the HE and LTSP events by repeating the simulations with the only change being the negative of the vertical seismic time history is used in conjunction with the specified horizontal time histories.

(5) The contact between the MPC and the overpack is simulated by a classical impulse-momentum equation. The coefficient of restitution (COR) is set to 0.0 reflecting the large contact areas involved and the coefficient of friction is set to 0.5, which is representative of steel-on-steel. This is a realistic assumption and allows for energy loss during contact between the two, large rigid bodies.

(6) The interface contact between the base of the overpack and the ISFSI storage pad embedment is modeled by discrete linear springs to simulate the embedment anchor rods and by compression-only elements to simulate the balancing force from the embedment. The spring rates are computed using established methodology for embedment anchor components. Damping is consistent with that specified for steel and concrete components in Table 8.2-1. These are realistic assumptions that appropriately model the expected interface behavior.

(7) Bounding (high) weights for the cask components are used for conservative results; inertia properties are computed consistent with these bounding weights.

Each VN dynamic simulation produces time-history results for the tensile loads in each of the 16 embedment anchor rods, as well as time-history results for the total interface compression load between the base of the embedment plate and the ISFSI pad concrete. The results of the VN-time-history analyses are stored in spreadsheet form and a FORTRAN computer code is used to post-process the results to determine vertical-load and overturning-moment time-histories for subsequent structural-integrity evaluation.

To ensure the capture of all energy from a seismic event, while at the same time eliminating high frequency components not pertinent to satisfying Code requirements in a structural evaluation, the filtering frequency for processing the “raw” numerical results is set as 40 Hz. The use of filtering of dynamic results in cask structural integrity analysis has been previously licensed for the HI-STORM 100 System as described in the HI-STORM 100 System FSAR.

Acceptance Criteria:

The design criteria for the HI-STORM 100 SA overpack are discussed in Chapter 2 of the HI-STORM 100 System FSAR. The anchorage system, being an integral part of the overpack structure, is subject to the same design requirements. The anchorage (cask anchor studs, sector lugs, and adjacent shell structure) is designed to meet the static stress limits of ASME Section III, Subsection NF and Appendix F.

Two conditions for analysis are defined as follows:

(1) Level A (Preload) - The anchor stud preload is established at approximately 157 kips in each stud. Under this load and the corresponding balancing load from the ISFSI storage pad, the sector lug structural components must meet the allowable stress limits for plate and shell structures given in Article NF-3200. The stress limits at 200°F for SA-516, Grade 70 material (used for the sector lugs) listed in Table 3.1.10 of the HI-STORM 100 System FSAR are used in the acceptance evaluation.

(2) Level D (Preload plus Seismic Load) - In accordance with Appendix F of ASME Section III, the tensile stress in the stud, averaged through the cross-section is limited to 70 percent of the ultimate strength of the stud material. The extreme fiber stress in the stud is limited to ultimate strength per NF-1335.1. The design criteria and stress intensity limits for the sector lug components are given in Chapter 2 and Table 3.1.12, respectively, of the HI-STORM 100 System FSAR. The stud alternating stress intensity under the dynamic loading induced by the seismic event must be sufficiently low to ensure a safety factor greater than 1.0 against fatigue failure for the number of stress intensity cycles associated with the seismic event.

In addition to the above anchorage acceptance criteria, it is required to demonstrate that the seismic events do not induce acceleration levels in the body of the cask that exceed the cask design basis (45 g) as defined in the HI-STORM 100 System FSAR (Reference 6).

Assumptions

The key assumptions used in the dynamic model are listed and explained within the methodology description given above.

Key Design Inputs

Bounding weights of 270,000 lb for an empty HI-STORM 100SA and 90,000 lb for a loaded MPC are used in the analyses (References 6 and 12, Table 3.2.1). SA193-B7 material is used for the anchor stud material. For the dynamic analyses, anchor stud minimum yield and ultimate strengths of 105 ksi and 125 ksi, respectively, are used. Dimensions for the two cask bodies are taken from Drawing 3187 in Reference 12. Mass moment of inertia properties are determined based on cylindrical body assumptions with the specified mass uniformly distributed.

The spring rate of the embedment anchor rods is equivalent to a 2-inch diameter carbon steel rod, 48 inches long.

Seismic inputs for the dynamic analyses are obtained from acceleration time histories developed from the response spectra for each of the DCPP earthquakes.

Results of Analyses

The results from the series of analyses performed for the anchored cask can be summarized as follows:

(1) The anchored HI-STORM 100SA overpacks do not exceed the generic cask design basis deceleration limit of 45 g under any of the seismic events.

(2) The state of stress in the cask anchor studs and in the overpack bottom flange, gussets, and the shell structure remain below the stress limits of ASME Section III, Subsection NF and Appendix F under all seismic events.

(3) The interface loads on the embedment structure determined for the ISFSI pad structural qualification are summarized in Table 8.2-9. The peak values are obtained from the filtered, time-history results for embedment anchor rod tension and for interface compression from the dynamic simulations.

A finite element analysis of the sector lug was performed using as input the tensile load in the cask anchor stud. Structural integrity evaluations were performed for both Level A (where the preload is balanced by compression between the extended flange and the embedment plate) and for Level D conditions (where local lift-off of the flange is assumed and the stud maximum load capacity is conservatively assumed). The results from the finite element analyses are reported in Table 8.2-10.

The maximum values obtained for the interface loads at the embedment structure are summarized in Table 8.2-9 and form the input for the structural integrity evaluation of the ISFSI pad.

The bounding cask weight is 360 kips. Using the maximum net shear force result from Table 8.2-9 and dividing by the cask weight provides the effective “g” loading on the cask as 1.43 g. This demonstrates that the cask design basis deceleration level (from the HI-STORM 100 System FSAR) of 45 g is not exceeded with a large margin of safety.

The results summarized in Table 8.2-9 provide the information needed to determine the coefficient of friction required at the cask/embedment plate interface to ensure that there is no relative sliding at that location. These results are obtained by dividing the net filtered shear force by the filtered normal force at each instant of time through the simulation. From the simulations performed, the largest required value for the coefficient of friction is 0.18. In accordance with the ASME Code (NF-3324.6, Table 3324.6(a)(4)-1), a minimum coefficient of friction of 0.25 may be assumed to exist at the interface when preload is used. Therefore, the minimum safety factor against sliding of the cask relative to the embedment plate is 1.39 and the desired benefit of the preload is assured.

To evaluate the propensity for a failure by fatigue in the sector lug, the results from the finite element stress analysis of the sector lug under the limiting tensile load was used. Using the recommended methodology for fatigue analysis as outlined in ASME Section III and determining the likely number of stress cycles by using the results from the dynamic analyses, large margins of safety against a fatigue failure during a single seismic event were obtained. Therefore, fatigue failure of the overpack anchorage is not credible at the Diablo Canyon ISFSI.

To ensure continued maintenance of the design bases assumptions for preloading of the anchorage connections, PG&E will develop an inspection program that periodically visually checks a sampling of the exposed portions of the anchor studs, washers, nuts, and storage cask baseplate surrounding the nuts to note any degradation or relaxation of these connections. This program will verify that the studs, washers, and nuts have not turned from their as-left preloaded position, are not loose to the touch, and that visually their mating surfaces remain engaged.

Alternate Cask and Anchorage Seismic Analysis (Partial Loss of Preload Condition)

Methodology

The alternate analysis model of the overpack and anchorage consists of the following major components:

(1) A simplified linear dynamic model of the cask was developed and benchmarked to the design basis Holtec VN model, and tuned to represent the fundamental frequencies of the cask both horizontally and vertically.

(2) The benchmarked model was then re-analyzed subject to the ILP motion (5 sets of TH, each developed to match the target ILP spectra) and cask reactions were extracted.

(3) A detailed non-linear model of the cask anchorage system was developed. The main non-linearity of this model allows for the fact that the individual anchors have different support stiffness in compression vs. tension (consistent with Holtec VN model). The compressive stiffness was set to “rigid” to model the fact that the ISFSI pad concrete provides the desired stiffness in compression. The tensile stiffness was set to individual stiffness of the anchorage system (anchor rods and couplers), to model the fact that under tension, only the anchorage system provides the tensile stiffness.

(4) This model was run subject to cask global reactions in order to obtain the individual embedment tensile load for the governing Hosgri (original design basis load case), to serve as benchmarking of the model with the Holtec VN model.

(5) The non-linear embedment model was re-run again subject to cask global ILP reactions to arrive at the individual anchorage tensile load for the condition of no pre-tension in cask studs and subject to ILP control motion.

(6) The cask sector lug analysis in the Holtec VN model was reviewed to evaluate the impact of higher cask reactions on the available margins.

(7) The cask stud evaluation in the Holtec VN model was reviewed to evaluate the impact of higher cask reactions on the available margins. In addition, the impact of change of design concept for studs from that of a friction (slip-critical) type connection to bearing type connection was evaluated.

(8) Every component within the individual anchorage design (anchor rods, embedment plates, couplers, nuts and bolts) were reviewed to evaluate the impact of higher cask reactions on the available margins.

Acceptance Criteria

The cask and anchorage meet the same applicable acceptance criteria as the original design basis analysis described above.

Assumptions

The key assumptions used in the dynamic model are explained within the methodology description given above.

Key Input Data

The key input data used are the same as for the original design basis analysis above.

Results

The maximum values obtained for the interface loads at the embedment structure are summarized in Table 8.2-9A and form the input for the alternate structural analysis of the ISFSI pad described in Section 8.2.1.2.3.2.

The bounding cask weight is 360 kips. Using the maximum net shear force result from Table 8.2-9A and dividing by the cask weight provides the effective “g” loading on the cask as 1.51 g. This demonstrates that the cask design basis deceleration level (from the HI-STORM 100 System FSAR) of 45 g is not exceeded with a large margin of safety.

The analysis concludes that up to three anchor studs could lose pretension and the cask, anchorage and storage pad (results in Section 8.2.1.2.3.2) remain within their original design basis load limits.

The anchor stud load in Table 8.2-9A remains bounded by the preloaded analysis in the Holtec VN model presented in Table 8.2-10. Therefore, fatigue failure of the overpack anchorage is not credible as discussed above.

8.2.1.2.3.2 Storage Pad Seismic Analyses

The objective of the seismic analyses of the concrete pad is to ensure that the steel reinforced concrete pads and the anchored casks remain functional during all seismic conditions. A static analysis was performed to determine the storage pad size and thickness required to resist the loads resulting from seismic accelerations (DE, DDE, HE, and LTSP ground zero period acceleration [ZPAs]) applied to the pad, in addition to the resultant loads from the cask dynamic analysis (Section 8.2.1.2.3.1). Also, a nonlinear time history analysis of the cask/pad set-up was performed to determine the extent of sliding that occurs at the pad/rock interface.

In order to accommodate the periodic ISFSI storage pad construction discussed in Section 3.3.2, an additional analysis of the storage pad was performed to provide alternate cask loading sequences (Section 4.2.1.1.6) that could be used to place casks as far away from adjacent construction activities as practicable.

Pad Static Analysis

Methodology

The analysis is a nonlinear static finite element analysis (FEA), using the ANSYS computer code. The storage pad size and thickness analysis is not sensitive to long period ground motion. Therefore, this analysis (Reference 29) was performed using the four DCPP seismic events (DE, DDE, HE and LTSP). The seismic inputs used for this analysis were HE and LTSP ZPAs. The HE and LTSP spectra were used since these spectra produce the largest ZPAs and the cask/pad interfaces are not sensitive to longer period ground motion. The concrete slab was allowed to lift off the rock support if the loads and geometry dictate that liftoff should occur. All material properties are linear. Compression only gap elements are used at the interface between the slab and the rock. This is the only nonlinear modeling feature in the analysis.

The FEA model consists of the pad, portion of the underlying rock, and elements representing the cask on top of the pad. The casks are modeled up to a plane, 118.5 inches above the slab. This is the location of the center of gravity of the casks and is, therefore, where the loads are applied. The pad uplift and concrete stresses are determined by the FEA analysis. The steel embedment/anchorage structure is designed to meet the ductile anchorage provisions of Appendix B to ACI 349-01, as endorsed by Regulatory Guide 1.199. Certain provisions of Appendix B to ACI 349-01 that are endorsed by Regulatory Guide 1.199 are not applicable due to the thickness of the pad and length of the rod. Specifically, design strength capacity of the embedded base plate; concrete bearing and diagonal tension shear capacity computed must be more than the required ductile design strength of the embedded rod/stud. The Newmark 100-40-40 Method is used to combine the three specified directions of the seismic load.

Acceptance Criteria

Concrete and the embedded steel structures are designed in accordance with the requirements of ACI-349-97 and Appendix B to ACI 349-01, as endorsed by Regulatory Guides 1.142 and 1.199, respectively, and in accordance with NUREG-1536 (Reference 13). The concrete and embedment steel structures are constructed in accordance with ACI 349-01.

Assumptions

Normal engineering assumptions associated with developing FEA models (for example, boundary conditions, modeling techniques). The anchorage evaluation methodology used assumes the loading imposed on the pad embedment structure is similar to an inverted column and as such diagonal shear provisions of the ACI 349-97, Section 11.3, were followed.

Key Input Data

Table 8.2-9 shows the resultant cask loading on the pads. The underlying rock material properties have an impact on the analysis. The rock's Young's modulus range of 0.2×10^{6} psi to 2.0×10^{6} psi were considered in the analysis to account for variability of the rock types.

Rock elastic properties for the analyses were obtained from References 48 and 49. These properties are stress-strain dependent; that is, they are appropriate only for the range of stresses and strains for which they are determined.

A check was made of the calculation results in Reference 29 to verify that stresses and strains calculated in the underlying rock mass were within the range for which use of the rock properties is appropriate. It was determined that the average values of shear and compressive strain calculated within a volume of rock approximately 35 ft deep beneath an ISFSI pad are comparable to the range of strains for which the rock elastic properties were determined. Therefore, the elastic properties are appropriate for use in pad load-displacement analyses.

Results

The maximum pad stresses and the embedded steel ductility requirements meet the guidance of Regulatory Guides 1.142 and 1.199, respectively. The yield strength of the embedded studs is greater than 250 percent of the computed demand load on these studs. The maximum potential uplift on an edge of the pad is less than 1/32 inch to 1/8 inch, depending on the variation in the rock properties.

Pad Analysis - Alternate Cask Loading Sequence (Loss of Stud Pre-Tension Condition)

An alternate analysis of the storage pad conservatively assuming a full loss of pretension condition was performed in order to load casks in a different sequence on the storage pad (Reference 66). The analysis concludes that while margins decreased due to the higher seismic input loading from ILP, the cask, anchorage and storage pad all meet design and licensing basis limits.

Methodology

The methodology used in this analysis is to develop a 3-D non-linear pad model, modeling the pad itself using thick shell elements, casks using rigid beam elements, and the underlying soil/rock using Winkler spring formulation. The non-linearity will be at the interface of rock and pad allowing for potential separation of the pad from supporting rock during possible uplift in a seismic simulation. The effects of potential separation of the pad from the rock under seismic loading (also called pad uplift) will be explicitly accounted for by introduction of gap elements (geometric non-linearity) between the Winkler springs modeling the rock and the shell elements modeling the pad.

In this model, the following modeling technique is used:

- Pad will be modeled by thick plate elements having out-of-plane bending stiffness for rotation about global X and Y axes, as well as in-plane shear resistance along global Z axis.
- Casks will be modeled by rigid stick elements connecting the cask CG to the cask/mat interface along the circumference of each cask.
- The circular base of the cask is modeled using a series of rigid link elements along the circumference of the cask with radial rigid link elements connecting the center of the cask to the circumference at the cask CG elevation.

- Rock will be modeled by non-linear link (NLLink) elements which will have gap characteristics at the top. While under compression, contact is maintained and rock spring constant resists the pad movement, but if the pad is separated from the rock, the gap element opens, and the spring constant of the rock will no longer hold the pad down. Rock spring constant is modeled by equivalent Winkler formulation.
- Applied reaction from the cask to the pad is applied as a point load at the location of cask CG in order to simulate both the lateral shear and overturning moment imposed by the casks onto the pad.
- Pad vertical seismic inertia load is applied by a uniform acceleration multiplied by the pad weight along the vertical direction. The applied acceleration will be the ZPA of ILP motion in the vertical direction, since the pad is considered rigid.
- Pad horizontal inertia load is ignored in these analyses as it does not contribute to pad bending moments. The horizontal pad inertia will result in axial load and this will be calculated by hand calculation and used in pad axial-moment interaction checks, if necessary.

Only the soft rock case will be analyzed, since analyses of the original pad design conclude that this is the critical load condition.

The 3-D model is analyzed for the case of a fully loaded pad, subject to cask reactions from ILP case allowing for no pre-tension in the cask studs (as determined from the Partial Loss of Pre-Tension analysis above), in order to compute new pad moment and shears as well as rock bearing pressures.

Specifically four (4) sets of analyses for the fully loaded case will be performed as follows:

- Aligning the net cask seismic loads along NS
- Aligning the net cask seismic loads along EW
- Aligning the net cask seismic loads towards the SE corner
- Aligning the net cask seismic loads towards the NW corner

Each analysis case will be performed 3 times, once applying 100% of horizontal load in conjunction with 40% of vertical load applied in the up direction (to maximize pad uplift), and the second time, the reverse (with the 40% vertical load applying in the down direction), to maximize pad shear. The 3rd analysis cases will be with the 100% vertical seismic load applied in the up direction, concurrent with 40% of horizontal seismic load. The case of 100% seismic load down is not considered critical for pad uplift scenarios and will not be analyzed.

Therefore, for the fully loaded case, 12 analysis cases will be performed. These cases will form the current design basis condition.

Acceptance Criteria

The cask and anchorage meet the same applicable acceptance criteria as the original design basis analysis described above.

Assumptions

Normal engineering assumptions associated with developing FEA models (for example, boundary conditions, modeling techniques). The assumptions used in the modeling are explained within the methodology description given above.

Key Input Data

The key input data used are the same as for the original design basis analysis above.

Results

Though the available margins decrease due to the expected larger loads from the ILP spectra input, the maximum pad stresses continue to meet the acceptance criteria of the original analysis for the two alternate cask loading sequences evaluated.

Pad Sliding Dynamic Analysis

Methodology

A nonlinear time history analysis of the cask/pad structure sliding at the rock/pad interface was performed (Reference 28). The methodology for determining sliding resistance along the base of the pads is provided in Reference 39. Analyses were performed with the five sets of ILP time histories. The ILP time histories were used since the pad sliding analysis may be sensitive to long period ground motion and the use of ILP time histories produces bounding results.

A nonlinear stick model is developed for the purposes of these analyses. A lollypop stick model representing the cask behavior represents the set of 20 casks on a pad. The pad is represented by its mass only. The interface between the rock and the pad surface is modeled using SAP2000N's NLLINK element with friction properties. This element is a biaxial friction element that has coupled friction properties for the two shear deformations, post-slip stiffness in the shear directions, gap behavior in the axial direction. The cask superstructure stick is modeled such that it represents the dynamic properties of the anchored cask. [The cask and anchorage seismic analysis described in Section 8.2.1.2.3.1 models the anchored cask (in the absence of sliding of the pad) and perform dynamic analysis to predict the cask/pad interface design shears, moments, tension, and compression forces to be used in the pad design.] The fundamental frequency of the cask superstructure in sliding analyses is based on best estimate of the rocking frequency of the anchored cask. In the absence of local nonlinearities, it is expected that the fixed base model (no pad sliding) of the cask will yield slightly more conservative results than Section 8.2.1.2.3.1 results. The same model when mounted on the friction element is called the sliding model. The relative ratio of peak response between the sliding model and the fixed base model will yield an adjustment factor, which if found to be greater than unity, would have to be applied to the design shears and moments predicted by the analysis described in Section 8.2.1.2.3.1. This approach identifies any potential increases in design responses due to sliding.

For the vertical direction, the tensile component of cask/pad reactions is studied. This component is judged to be an important parameter that controls the normal resisting force at the interface, thus affecting the sliding displacement during a seismic event.

All analyses are performed based on the nonlinear time-history analysis option using Fast Nonlinear Analysis (FNA) approach of SAP2000N computer FEA program.

Acceptance Criteria

The pad must maintain its ability to perform its functional requirements with insignificant impact on the cask design qualifications.

Assumptions

Net Vector sliding is conservatively calculated assuming simultaneous peak X and Y horizontal sliding displacements.

Key Input Data

The analysis was performed assuming two pad-to-rock interface sliding friction coefficients μ = 1.19 corresponding to a friction angle of 50 degrees, and μ = 0.73 corresponding to a friction angle of 36 degrees. This represents the range of the sliding friction coefficient expected at this interface.

Cask Weight:	W = 360 kips
No. of Casks on a pad	20

Results

Based on the results of these analyses, the following is concluded:

(1) The best estimate of maximum pad sliding for a lower bound friction coefficient of 0.73 corresponding to a rock friction angle of 36 degrees is estimated as 1.21 inches.

(2) The best estimate of maximum pad sliding for an upper bound friction coefficient of 1.19 corresponding to a rock friction angle of 50 degrees is estimated as 0.41 inches.

(3) The above pad sliding displacements are considered small and not large enough to cause any damage to the pad or the casks. The acceptance criteria for pad sliding is defined as whether pad sliding results in increased design shears and moments at the cask-to-pad interface, which is discussed further below.

(4) After pad sliding is considered, it is concluded that the cask design shear of 515 kips (load on to the pad) remains valid for design. The best estimate of the adjustment factor to account for the effects of pad sliding is calculated as 0.95 for a friction coefficient of 1.19, and 0.90 for a friction coefficient of 0.73. Both of these ratios are below unity, as such the design shear of 515 kips (and associated moments) remains valid for design.

(5) The best estimate of maximum vertical tensile load after sliding remains unchanged. Thus the design axial bolt tensions of the analysis described in Section 8.2.1.2.3.1 remain valid.

(6) The response spectra comparison plots of the rock versus pad sliding indicate that the responses at the cask-to-pad interface generally do not vary up to about 16 Hz. However, above this frequency some differences in the responses are seen as a result of sliding. An evaluation by the cask supplier determined that there were no components inside the cask sensitive to changes in input motion in this higher frequency range. The highest peak spectral ordinate associated with change in motion as a result of pad sliding is 4.1 g at approximately 26 Hz and 5 percent critical damping well below the cask qualifications.

(7) Given that the base shear (and therefore base moments) and axial tension do not change as a result of pad sliding, it is concluded that analyses described in Section 8.2.1.2.3.1 remain valid.

Effect of Alternate Analyses on Pad Sliding Dynamic Analysis

With respect to having no pre-tension in cask studs, this issue does not impact the analyses performed above as these analyses assumed a simplified fixed base model of the cask alone for validation purposes, followed by having a simple representation of 20 casks on pad, with non-linear sliding elements having a certain friction coefficient representing the interface between the pad bottom surface and rock top surface. Nowhere in the analysis is the pre-tension in cask anchorage studs a variable.

Thus, it is concluded that the issue of having no pre-tension in cask anchor studs does not invalidate the pad sliding global analyses performed. Furthermore, since the pad sliding analyses were performed subject to ILP seismic event, these analyses are already current and as such remain valid.

8.2.1.3 Earthquake Accident Dose Calculations

The HI-STORM 100SA overpack and the HI-TRAC transfer cask were explicitly analyzed for, and shown to withstand the seismic ground motion during transport to the CTF, during activities conducted at the CTF, during movement from the CTF to the storage pads, and during storage operations, as applicable. The seismic ground motion does not cause stresses above allowable limits in the MPC confinement boundary, the transfer cask, or the storage overpack during canister transport, transfer, or storage operations. The CTF and cask transporter structures are also designed to withstand the DCPP ground motion. No radioactivity would be released in the event of an earthquake and there would be no resultant dose.

8.2.2 TORNADO

A tornado is classified as a natural phenomenon Design Event IV, as defined in ANSI/ANS-57.9. This event involves the potential effects of tornado-induced wind, differential pressure, and missile impact loads on the ISFSI SSCs that are important to safety. The design basis wind and tornado evaluation is provided in Reference 27.

8.2.2.1 Cause of Accident

The cause of this event is the occurrence, at or near the ISFSI site, of meteorological conditions that are favorable to the generation of a tornado. The design-basis tornado wind speed for the ISFSI is based on a conservative estimate appropriate for DCPP (annual probability of 10^{-7}), which was developed by the NRC (Supplemental Safety Evaluation Report No. 7). The specific topography associated with the plant site indicates that the postulated tornado event is unlikely. However, it has been included in the ISFSI design basis as a potential accident event.

8.2.2.2 Accident Analysis

The accident analysis for tornado effects involves evaluation of the loaded transfer cask during transport to the CTF, MPC transfer activities at the CTF, transport of a loaded HI-STORM 100SA overpack to the ISFSI pad, and long-term storage of the loaded overpack at the ISFSI pad. As discussed in Section 3.2.1 and 4.2.3.3.2.6, tornado-wind and missile design criteria are a combination of Diablo Canyon site-specific winds and missiles and the design-basis missiles described in the HI-STORM 100 System FSAR. In the evaluation of the Diablo Canyon ISFSI for tornado effects, the missiles were categorized as large, intermediate, or small missiles and were compared with those missiles for which the HI-STORM 100 System was generically designed to withstand. The description, mass, and velocity of all missiles considered for evaluation are listed in

Table 3.2-2. As noted in Table 3.2-2, some of the additional Diablo Canyon ISFSI missiles were conservatively evaluated for the generic Region II missile velocities described in NUREG-0800, Section 3.5.1.4. The 1800 kg automobile and the 4 kg, 1-inch-diameter steel rod were determined to be the bounding large, and small missiles, respectively. For the intermediate missile category, the 500-kV insulator string was found to be bounding for penetration resistance and the 8-inch-diameter steel rod was determined to be bounding for the global stress evaluation.

The bounding large and intermediate (for penetration only) missiles were chosen by comparison of the kinetic energies of the missiles. The small missile was chosen based on the guidance of NUREG-0800, Section 3.5.1.4, for selecting a missile that can pass through an opening in a protective barrier. For the global stress evaluation of the intermediate category missile, the bounding missile was chosen based on a comparison of safety factors (SF), the missile producing the lower SF being bounding. If the generic analysis described in the HI-STORM 100 System FSAR was bounding, no additional evaluation was performed. If a DCPP site or Diablo Canyon ISFSI-specific missile was bounding, an analysis was performed for the applicable component (that is, the overpack and/or the transfer cask). The following is a summary of the evaluations performed for the four operating ISFSI configurations: transport to the CTF, MPC transfer activities at the CTF, transport to the ISFSI pad, and long-term storage at the ISFSI pad.

The missile impacts are analyzed using formulas from Bechtel Power Corporation Topical Report BC-TOP-9A (Reference 14), ORNL Report TM-1312 (Reference 33), and energy balance methods. In all cases, at all locations away from the impact locations, missile-induced stresses in the transfer cask and overpack are below ASME Level D stress intensity limits.

Another possible consequence of a tornado is to cause the collapse of a nearby 500-kV transmission tower. This event is discussed in Section 8.2.16.

8.2.2.2.1 Transport to the CTF

The transfer cask is transported between the DCPP FHB/AB and the CTF in a vertical position. Section 3.4.8.2.2 of the HI-STORM 100 System FSAR discusses the side impact from a large missile and concludes loads are below ASME Level D stress intensity limits. The small missile is bounded by the intermediate missile. The evaluations for the side, top, and bottom impact from an intermediate missile (344.7-kg insulator string traveling at 157 mph) are as follows.

- For the side impact, conservatively neglecting the water jacket and the lead shielding, the intermediate missile will penetrate the outer steel shell, but will not penetrate the 3/4-inch inner shell of the transfer cask. Using this conservative model, the minimum inner shell thickness required to withstand the missile impact is 0.266 inch. The design inner shell thickness is 0.75 inch.

- The HI-STORM 100 System FSAR contains an evaluation for the impact of the intermediate missile on the HI-TRAC pool lid bottom plate. The analysis shows that the intermediate missile would not penetrate the 1-inch, carbon-steel bottom plate of the pool lid. The minimum required steel thickness to withstand the missile impact is 0.516 inch.

- On the top of the transfer cask, the top lid has a hole for rigging, lowering, and raising the MPC during transfer of the canister between the transfer cask and the overpack. An analysis was performed for the 500-kV insulator string intermediate missile entering the transfer cask through the hole in the top lid and impacting the MPC lid. If the insulator string missile directly impacts the MPC, it will not penetrate the 9-1/2-inch-thick, stainless-steel lid. The global stress analysis of the 8-inch steel cylinder missile impacting the MPC lid yielded a safety factor against failure of the peripheral MPC lid-to-shell (LTS) weld of 1.23 versus a safety factor of 7.1 for the insulator string.

8.2.2.2.2 Transfer Operations at the CTF

During MPC transfer operations at the CTF, the transfer cask and the overpack are oriented vertically with the transfer cask stacked on top of the overpack. All but approximately the top 4 ft of the overpack are below grade and not susceptible to tornado missile strikes. The top of the overpack is shielded by the transfer cask until the transfer cask is removed to allow installation of the HI-STORM lid. As discussed in Section 8.2.3.1, cask transport and transfer operations will not be conducted during severe weather. The top of the MPC will only be exposed for a short duration (nominally less than 4 hours). Therefore, in the configuration with the lid removed, a tornado missile impact is not credible. With the top of the MPC exposed during this time, the evaluation of an intermediate missile impact on the MPC lid, described in Section 8.2.2.2.1 ensures the MPC integrity is maintained.

In the vertical orientation, the top of the transfer cask is not subject to direct impacts from these missile strikes and the bottom of the transfer cask is not exposed to tornado-missile strikes. The evaluation of the missile strike on the side of the transfer cask described in Section 8.2.2.2.1 is applicable for this configuration.

8.2.2.2.3 Overpack Transport to the ISFSI Pad

The effect of tornado missiles impacting the transporter while carrying an overpack during transport to the ISFSI pad was evaluated for a horizontal large tornado missile. The transporter with overpack will not turnover from the impact.

Tornado wind effects are enveloped by the HI-STORM 100 System FSAR analysis of a freestanding HI-STORM on a pad. The overpack is lifted only to those heights necessary to travel from the CTF to the ISFSI storage pad. Typically, this is only several inches. This small lift height eliminates tornado missiles striking the bottom of the cask as a credible event.

8.2.2.2.4 Long-Term Storage at the ISFSI Pad

The HI-STORM 100 and 100S free-standing overpack designs have been analyzed for steady state tornado wind loads with a concurrent, large-missile impact, as well as intermediate and small-sized missiles for penetration, as described in Appendices 3.C and 3.G of the HI-STORM 100 System FSAR. The anchored version of the HI-STORM 100S overpack (HI-STORM 100SA) to be deployed at the Diablo Canyon ISFSI is bounded by the free-standing analysis because the anchorage provides additional protection against overturning. The wind loading evaluated in the HI-STORM 100 System FSAR bounds the maximum wind loading at the Diablo Canyon ISFSI site (Table 3.2-1). The loads on the MPC confinement boundary due to the design-basis, 3.0-psi pressure differential are bounded by the 100-psi normal design internal pressure for the MPC described in Section 3.4.4.3.1.2 of the HI-STORM 100 System FSAR. The HI-STORM 100SA overpack is a ventilated design that includes four air inlet ducts and four air outlet ducts at the bottom and top, respectively. Therefore, no tornado-induced pressure differential analysis was performed for the overpack.

The HI-STORM 100SA overpack is generically designed to withstand 3 types of tornado missiles in accordance with Section 3.5.1.4 of NUREG-0800.
Sections 3.4.8 and 3.4.8.1, as well as Appendices 3.C and Appendix 3.G of the HI-STORM 100 System FSAR, provide discussions of the generic design criteria and the effects of the large (automobile), intermediate (rigid cylinder) and small (sphere) tornado missiles on the overpack. The Diablo Canyon ISFSI-specific intermediate missile (344.7-kg insulator string) is a more limiting design-basis missile for penetration and was evaluated for penetration after impacting the outer shell and the top lid of the overpack at design-basis velocity. The 8-inch-diameter steel cylinder was evaluated generically for global stresses induced after a strike on the top lid of the overpack. The Diablo Canyon ISFSI-specific small missile (1-inch-diameter steel rod) was evaluated for puncture and whether it will enter the overpack air ducts and impact the MPC at design-basis velocity.

The small missile, while less energetic than the intermediate missile, was analyzed specifically due to its unique ability to travel through one of the overpack air inlet ducts and directly impact the MPC pedestal. The evaluations of the effects of the large, intermediate, and small categories of missiles impacting the overpack are described below.

- The free-standing overpack is capable of withstanding the combination of tornado wind (or instantaneous pressure drop) and a large-missile-load impact with a conservative safety factor against overturning of greater than two. The anchored cask system, which provides additional resistance to overturning, is bounded by the free-standing overpack analysis. Local damage to the cask surface by a large-missile impact is bounded by the small and intermediate category missiles.

- Conservatively neglecting the concrete in the overpack, the 500-kV insulator string intermediate missile will penetrate the outer shell of the overpack, but will not penetrate the 1-inch inner shell of the overpack or result in loss of MPC retrievability. Using this conservative model, the minimum inner shell thickness required to withstand the missile impact is 0.266 inches.

- The 500-kV insulator string intermediate missile will not penetrate the 3-inch top lid of the overpack. The minimum required thickness to withstand the missile impact is 1.089 inches.

- The 8-inch steel cylinder intermediate missile will not cause an over-stress condition on the overpack lid. The factor of safety is 1.4 for this event. The factor of safety for the 500-kV insulator string for this event is 3.2.

- The 1-inch diameter steel rod (that is, small missile) is postulated to enter an overpack inlet duct and impact the pedestal shell. The analysis shows that the rod will pierce the shell and penetrate the concrete to a depth of 6.179 inches, which is significantly less than the radius of the pedestal shield. The damage to the concrete pedestal shield does not affect the confinement boundary or the ability of the MPC to remain standing on the pedestal, nor does it affect the retrievability of the MPC.

The effects of large and small missiles on the free-standing HI-STORM 100 overpack, which were determined in the generic evaluations, are applicable to and bounding for the anchored HI-STORM 100SA overpack to be deployed at the Diablo Canyon ISFSI. The Diablo Canyon ISFSI-specific intermediate missile has been evaluated for penetration and found to have acceptable consequences. The effect of the intermediate missile impact on the overpack lid has been evaluated and found to have acceptable consequences.

8.2.2.3 Conclusions

The above discussion demonstrates that the HI-STORM 100SA overpack and the HI-TRAC transfer cask provide effective missile barriers for the MPC. No missile strike will cause instability of the overpack, compromise the integrity of the confinement boundary or jeopardize retrievability of the MPC. In addition, global stress intensities arising from the missile strikes satisfy ASME Code Level D limits for an ASME Section III Subsection NF structure. For the case where the transfer cask is being transported to the CTF in the vertical position, the MPC top lid has been evaluated for an intermediate missile strike. The stress intensities from this missile strike satisfy the ASME Section III Subsection NB Level D limits. Therefore the requirements of 10 CFR 72.122(b) are met with regard to tornadoes.

The cask transporter has redundant drop protection by design (Section 3.3.3). Therefore, a loss of load due to a direct missile strike on the transporter is not credible. Since the CTF structure at DCPP is underground, it is not exposed to missile impacts (Section 3.3.4).

8.2.2.4 Accident Dose Calculations

Extreme winds in combination with tornado missiles are not capable of overturning a storage cask or of damaging an MPC within a storage cask resulting in a loss of shielding. Therefore, no radioactivity would be released due to tornado effects on the overpack in the event of a tornado. Dose rates at the controlled area boundary and onsite would not be affected by the minor damage to the transfer or storage cask from tornado-driven missile strikes.

8.2.3 FLOOD

A flood is classified as a natural phenomenon Design Event IV in accordance with ANSI/ANS 57.9.

8.2.3.1 Cause of Accident

The probable maximum flood is classified as a severe natural phenomenon. In general, floods are caused by extended periods of rainfall, tsunamis, storm surges, or structural failures, such as a dam break.

The Diablo Canyon ISFSI storage pads are located at an elevation of over 300 ft mean sea level (MSL). The Diablo Canyon ISFSI site surface hydrology is described in Section 2.4. It is concluded in Section 2.4 that there is no potential for flooding in the vicinity of the ISFSI storage pads. Therefore, flooding is not a consideration for ISFSI operations or on the capability of the dry storage cask system to safely store the spent fuel. Likewise, due to the elevation of the ISFSI site, a tsunami (about 35 ft MSL) as discussed in the DCPP FSAR Update (Reference 4), Section 2.4.7, is not a threat to the HI-STORM 100 Systems being stored on the pad. Since the CTF is located adjacent to the ISFSI pads, it is similarly concluded that there is no potential flooding and tsunami impact on the CTF.

Floods are generally predictable events. As such, administrative controls contained in ISFSI operating procedures will be used to preclude transport of the MPC in a transfer cask, CTF MPC handling activities, and transport of a loaded overpack between the CTF and storage pads during severe weather. Therefore, flooding during these configurations is also not considered credible. Also, the minimum elevation of the transport route (about 82 ft MSL) precludes a tsunami flooding the transport route while in use.

The potential for flooding at the CTF is further reduced by having the top of the CTF cylinder 1 inch above grade and a removable cover that is installed when the CTF is not in operation. As a further precautionary measure, the CTF is equipped with a sump as described in Section 4.4.5.

8.2.3.2 Accident Analysis

The HI-STORM 100 System is designed to withstand the pressure and water forces associated with a flood. The design criteria for a flood are discussed in Section 2.2.3.6 of the HI-STORM 100 System FSAR. The flood is assumed to submerge the HI-STORM 100 System to a depth of 125 ft with a water velocity of 15 ft/sec (HI-STORM 100 System FSAR, Table 2.2.8).

No additional flooding analyses have been performed for the Diablo Canyon ISFSI because flooding of the ISFSI is not considered credible.

8.2.3.3 Accident Dose Calculations

Flooding is not a credible event for the Diablo Canyon ISFSI because of the elevation of the ISFSI site. There will be no releases of radioactivity and no resultant doses.

8.2.4 DROPS AND TIP-OVER

The hypothetical drop/tip-over of a storage cask is classified as Design Event IV, as defined by ANSI/ANS-57.9. The design for the Diablo Canyon ISFSI, as explained below, eliminates the need to postulate and analyze cask drop and non-mechanistic tip-over events (Reference 40). The load path portions of the cask transporter and the lifting devices attached to the cask components (that is, the HI-TRAC lifting trunnions and the overpack lift bolt anchor blocks) are designed to preclude drop events, either through redundancy or enhanced safety factors. Overturning of the loaded HI-TRAC transfer cask attached to the LPT is not credible under all scenarios. Table 2.2.6 of the HI-STORM 100 System FSAR discusses the design codes and standards applicable to the transfer cask and the overpack. Sections 3.3.3, 4.3, and 8.2.1 discuss the design criteria, applicable codes and standards, and design features of the cask transporter that demonstrate the transporter does not leave the transport route, tip over, or drop the loaded transfer cask or overpack under all design basis conditions, including natural phenomena.

Section 8.2.1 describes the analysis of a seismic event, verifying that the cask transporter and LPT will not drop a loaded transfer cask or overpack, and the cask transporter will remain stable on the transport route for the duration of the earthquake. Therefore, transfer cask and overpack drop events are not analyzed outside the FHB/AB, nor are maximum lift heights established for handling the casks. Administrative controls in operation procedures ensure the casks are lifted only to those heights necessary to complete the required activities for cask loading and unloading.

The design of the Diablo Canyon ISFSI also includes a requirement to anchor the overpack to the concrete ISFSI pad. This design concept is necessary to accommodate a design-basis seismic event at the site without the cask sliding or tipping over. The anchored overpack concept eliminates the need to postulate a non-mechanistic tip-over of the loaded overpack when anchored to the ISFSI storage pad. Section 8.2.1 describes the analysis that verifies the anchored overpack will not slide or tip over during a seismic event. Section 8.2.2 describes the analysis that demonstrates that the overpack will not tip over as a result of tornado wind concurrent with a large tornado missile impact.

An evaluation was performed for four short-term transient lifting and lowering activities during the transport operation where the cask transporter may not maintain its complete seismic capability. These involve:

(1) lifting or lowering the loaded HI-TRAC transfer cask between its bolted configuration on the LPT and its transport configuration on the transporter;

(2) lifting or lowering the loaded HI-TRAC transfer cask between the transport configuration on the transporter and its bolted configuration on the mating device at the CTF;

(3) lifting or lowering the HI-STORM overpack between the transport configuration on the transporter and entry into the CTF shell; and

(4) lifting or lowering the HI-STORM overpack between the transport configuration on the transporter and the anchored configuration on the ISFSI pad

This evaluation shows that based on the minimal height of the lifts and the duration of these activities, the probability of a design basis event during those lifts is not credible.

8.2.4.1 Cause of Accident

Cask drop or tip-over is not a credible event outside the DCPP FHB/AB as discussed above. Cask drop events are not credible inside the DCPP FHB/AB due to the use of a single-failure-proof FHB crane.

At the Diablo Canyon ISFSI, transfer of the loaded MPC between the transfer cask and the overpack is accomplished at the CTF using the cask transporter to lift the transfer cask to the height necessary to accomplish this objective. The cask transporter used in Diablo Canyon ISFSI operations is designed, fabricated, inspected, maintained, operated, and tested in accordance with the applicable guidelines of NUREG-0612. Therefore, a drop of the loaded MPC during inter-cask transfer operations is not a credible event.

8.2.4.2 Accident Analysis

As discussed above, cask drop or tip-over or MPC drop are not credible events outside the FHB/AB.

8.2.4.3 Dose Calculation for MPC Drop Event

Cask drop or tip-over or MPC drop are not credible events. Thus, there is no breach of the MPC confinement boundary and no release of radioactivity.

8.2.5 FIRE

Fires are classified as human-induced or natural phenomena design events in accordance with ANSI/ANS 57.9 Design Events III and IV. To establish a conservative design basis, the following fire events are postulated:

(1) Onsite 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) Fire from mineral oil from the Unit 2 transformers

The potential for fire is addressed for both the overpack and the transfer cask. Locations where the potential for fire is addressed include the ISFSI storage pads, the area immediately surrounding the ISFSI storage pads, including the CTF, and along the transport route between the DCPP FHB/AB and the ISFSI storage pads. The evaluations performed for these postulated fire events (Reference 41) are discussed in the following sections.

8.2.5.1 Cause of Accident

Multiple causes, both human-induced and natural, are assumed for each of the fire events postulated above. For the purposes of this FSAR, all conservatively postulated fire events are classified as ANSI/ANS 57.9, Design Event IV, events that are postulated because they establish a conservative design basis for important-to-safety SSCs.

There are several potential mechanisms for the initiation of Events 1, 2, 3, 4, and 6, listed above, including both human-induced (electrical shorts, vehicle accidents, transmission line strikes, etc.) and natural (lightning strikes, tornado missiles, etc.)

phenomena. While the probability of occurrence of these mechanisms would be very low, the classification of these fire events as ANSI/ANS 57.9, Design Event IV, requires performing an evaluation.

The postulated fire in the vegetation surrounding the ISFSI storage pad (Event 5) could be caused by the spread of an offsite fire onto the site or as the result of natural phenomena such as a lightning strike or a transmission line strike. Unlike the other fire events, it is reasonable to expect that some type of vegetation fire will occur during the ISFSI license period. While plant personnel would quickly act to suppress or control vegetation fire, it is postulated that no fire suppression activity occurs. Thus, this fire event is conservatively classified as an ANSI/ANS 57.9, Design Event IV.

8.2.5.2 Accident Analysis

For the evaluation of the onsite transporter and other onsite vehicle-fuel-tank fires (Events 1 and 2), it is postulated that the fuel tank is ruptured, spilling all the contained fuel, and the fuel is ignited. The fuel tank capacity of the onsite transporter is limited by the Diablo Canyon ISFSI Technical Specifications (TS) to a maximum of 50 gallons of fuel. The maximum fuel tank capacity for other onsite vehicles in proximity to the transport route and the ISFSI storage pads is assumed to be 20 gallons. On the storage pad, the fuel is postulated to be burning in a pool surrounding the cask, therefore, the concrete short-term temperature limit will be exceeded and is an expected consequence of the event. Recovery from a fire event on the ISFSI pad will require a technical evaluation of the ability of the ISFSI pad, in the affected area, to perform its design function, and appropriate corrective actions taken as necessary

A potential fire in the CTF due to the release of the 50 gallons of fuel from the cask transporter has been addressed. The cask transporter will be designed with features (e.g., a limited fuel tank size and drip pan with drain) that ensure the fuel, if spilled, will not migrate into the CTF structure. The CTF opening will be located at a higher elevation than the local surrounding area such that any fuel spilled will flow away from the CTF by gravity. This ensures that any fire that may occur is bounded by the fire analysis described in Section 11.2.4 of the HI-STORM System FSAR.

Section 11.2.4 of the HI-STORM 100 System FSAR presents an evaluation of the effects of an engulfing 50-gallon fuel fire for both overpack and transfer cask. Results of these analyses indicate that neither the storage cask nor the transfer cask undergoes any structural degradation and that only a small amount of neutron shielding material (concrete, Holtite-A, and water) is damaged or lost. This analysis bounds any onsite, 20-gallon vehicle-fuel-tank fire (Event 2).

Portable generators and air compressors may be used during MPC transfer activities. If portable generators and air compressors are used, procedural controls will be established to ensure that they are bounded by the fire analysis described in Section 11.2.4 of the HI-STORM System FSAR.

The location of any transient sources of fuel in larger volumes, such as tanker trucks, will be administratively controlled to provide a sufficient distance from the ISFSI storage pads (at all times), the CTF, and the transport route during transport operations to ensure the total energy received is less than the design-basis fire event. In addition, when the tanker truck is moving on the roadway past the ISFSI, the roadbed in all cases is below the level of the ISFSI pad, which ensures that even if there were a tank rupture, the fuel would not run toward the ISFSI. An analysis was performed for a ruptured 2,000-gallon gasoline tanker truck, which determined that at a distance of more than 100 feet, it does not result in exceeding the design basis of the storage casks (Reference 34). There are fuel trucks on the DCPP site that carry up to 4,000 gallons of gasoline, however, those trucks are administratively maintained at least 1,100 ft from a cask being transported or the CTF/ISFSI facility. In addition, only trucks containing no more than 800 gallons of gasoline are allowed to pass the CTF/ISFSI facility at any time, and that movement is administratively controlled to ensure that the tanker is never at a distance that would not be bounded by the analysis performed for a ruptured 2,000-gallon gasoline tanker truck, which determined that at a distance of more than 100 feet, does not result in exceeding the design basis of the transfer cask. (Reference 34)

Administrative controls are imposed to ensure no combustible materials are stored within the security fence around the ISFSI storage pads. Prior to any cask transport, a walkdown will be performed to ensure all local combustible materials (Event 4), including transient combustibles, are controlled in accordance with ISFSI fire protection requirements. All stationary fuel tanks (Event 3) are at least 50 ft from the ISFSI storage pad security fence and at least 100 ft from the transport route and the CTF. These existing stationary tanks have been evaluated. Due to their distances to the transport route or the ISFSI pad, the total energy received by the storage cask or the transporter is insignificant compared to the design-basis fire event.

The native vegetation surrounding the ISFSI storage pad is primarily grass, with no significant brush, and no trees. Maintenance programs prevent uncontrolled growth of the surrounding vegetation. As previously stated, no combustible materials will be stored within the ISFSI protected area. A conservative fire model was established for evaluation of grass fires. Analysis has demonstrated that grass fires are bounded by the 50-gallon transporter-fuel-tank fire evaluated in the HI-STORM 100 System FSAR (Event 5). The wildfire evaluation uses predictive models called FARSITE and FLAMMAP (Reference 36) to determine the potential characteristics of wildfire in the Diablo Canyon. Both models utilize mapped data about the type of vegetation (fuel model), slope, aspect, elevation, wind, and moisture to predict wildfire characteristics such as flame length, rate of spread, heat per unit area, etc. The ISFSI site, located immediately southeast of the power plant's raw water reservoirs, is surrounded on the south, southeast, and north sides by a vegetation type of "annual grassland" (Reference 37). The main access road forms the northwest boundary of the proposed site. The annual grassland vegetation is grazed and has relatively low cover. Consequently, the fire risk of this fuel type is relatively low.

For Event 6, the physical properties of mineral oil limit the threat of a fire. The pertinent material property for this determination, the flash point, is defined as the lowest temperature at which the vapor pressure of a liquid is sufficient to produce a flammable vapor/air mixture at the lower limit of flammability. In other words, a combustible liquid cannot vaporize sufficiently to detonate if the ambient temperature is below the flash point. Such materials could conceivably burn, but would be incapable of detonation.

The flash point of mineral oil is 275°F. To be classified as flammable, the flash point of a liquid must be less than 100°F as discussed in the National Fire Protection Association Handbook (Reference 15). The highest ambient temperature predicted for the Diablo Canyon ISFSI site (5- to 10-year recurrence interval) is 104°F and would normally (99 percent of the time) be no more than 85°F; and the normal operating temperature of the 13,000 gallons of mineral oil in each of the DCPP Unit 2 main bank transformers is approximately 160°F. These temperatures are considerably less than the flash point of mineral oil. Therefore, under ambient or normal operating temperature, these materials do not represent a credible fire hazard. However, if an electrical fault were to occur in a transformer, the increase in heat within that transformer could cause it to rupture and its contents may support a local fire. The resulting fire is considered to be limited and bound by the design basis fire provided in Section 11.2.4 of the HI-STORM 100 System FSAR, and is further supported by an analysis performed for a ruptured 2000-gallon gasoline tanker truck, which determined that at a distance of more than 100 feet does not result in exceeding the design basis of the transfer cask. (Reference 34)

The probability of this event occurring while the transfer cask is in proximity and it affecting the transporter and transfer cask is extremely low. This is based on the properties of mineral oil, the minimum distance from the transformers to the transporter, the limited amount of exposure time, a dedicated transformer fire suppression system, and a significant difference in elevation between the transformers and the transporter route.

The transformers are approximately 240 ft from the transporter at its closest point during transport and the transporter is within a line of sight of the transformers for no more than 10 hours per year. Each of the transformers is surrounded by a dedicated fire suppression system that will act to control and minimize any fire that could potentially occur. There is also a 30-ft difference in elevation between the transporter route and the transformers that will not allow oil from a transformer to approach within approximately 120 ft of the transporter.

In addition, although a fire from a transformer is considered bounded by the design basis of the transfer cask and not an unacceptable hazard, in an effort to further minimize its probability, PG&E is taking prudent actions to minimize the transformer fire hazards during transport as follows:

For potential external hazards, administrative procedures will not allow any vehicle motion in the vicinity of the transformers during transport operations. In addition, administrative procedures are in place that will not allow transport of fuel when severe weather (which could result in lightning or other hazards) exists or is predicted to occur during the transport time in the vicinity of the DCPP plant site. To address the potential hazard for an internal short, PG&E administrative procedures consider offsite power conditions prior to transport operations in the vicinity of the Unit 2 transformers.

Based on the above discussion, the potential hazard from a transformer fire is considered credible; however, its potential effects are limited and considered bounded by the design basis fire analysis for the transfer cask.

In summary, the fire evaluations performed generically in the HI-STORM 100 System FSAR, the physical layout of the Diablo Canyon ISFSI, the fire analysis for the surrounding vegetation, and the administrative controls on fuel sources ensure that the general design criteria related to fire protection specified in 10 CFR 72.122(c) are met.

8.2.5.3 Accident Dose Calculations

The effects of an onsite transporter, or other onsite vehicle-fuel-tank fire postulated for the Diablo Canyon ISFSI, are enveloped by the design basis transporter fire evaluated in the HI-STORM System FSAR. Section 11.2.4 of the HI-STORM 100 System FSAR describes how the MPC confinement boundary remains intact after a design basis fire for both the overpack and the transfer cask. Therefore, there is no release of the contained radioactive material from the MPC and no dose consequences in this regard. The shielding implications of a design basis fire for each of these components are discussed below.

8.2.5.3.1 HI-STORM 100 Overpack

Section 11.2.4.2.1 of the HI-STORM 100 System FSAR discusses the fire analysis for the overpack, including radiological implications. The design-basis fire for the HI STORM 100 overpack causes a small reduction in the shielding provided by the concrete. No portions of the steel structure of the overpack experience temperatures exceeding the short-term temperature limits. While the temperature in the outer 1-inch of concrete is shown to exceed the material short-term temperature limit, there is no significant reduction in the shielding provided by the overpack. All MPC component and fuel assembly temperatures remain within their short-term temperature limits as demonstrated by the Diablo Canyon ISFSI specific thermal analyses (Reference 67).

8.2.5.3.2 HI-TRAC Transfer Cask

Section 11.2.4.2.2 of the HI-STORM 100 System FSAR discusses the fire analysis for the transfer cask. The elevated local temperatures due to the fire will cause approximately 11 percent of the water in the water jacket to boil off and relieve as steam through the relief valves on the water jacket. However, it is conservatively assumed for the dose calculations that all of the water in the water jacket is boiled off. The fire could also heat the Holtite-A shielding material in the transfer cask top lid above its temperature limit. Therefore, it is conservatively assumed in the dose calculations that all of the Holtite-A in the transfer cask is lost.

The postulated losses of all neutron shielding, due to the loss of water in the water jacket and all Holtite-A in the transfer cask top lid, will not exceed the 10 CFR 72.106 dose limits at an assumed controlled-area boundary located 100 meters from the ISFSI pad for the 30-day duration of the accident, as discussed in Section 5.1.2 of the HI-STORM 100 System FSAR. The nearest controlled area boundary at Diablo Canyon is approximately 1,400 ft from the ISFSI storage pads, which would further decrease the estimated accident dose to well below the 10 CFR 72.106 limit.

Also, as shown in Tables C.3 and C.4 of Reference 67, the increase in fuel cladding and component material temperatures due to the fire and loss of water in the water jacket do not cause the short-term fuel cladding or material temperature limits to be exceeded. The internal MPC pressure also remains below the 200-psig accident design limit, as shown in Reference 67, Table C.5. Thus, there is no effect on the integrity of the MPC confinement boundary.

The ISFSI system is not affected by the postulated combustion of local fuel tanks, combustible materials outside the ISFSI storage pad perimeter or along the transport route, or an unsuppressed vegetation fire. Therefore, there are no dose consequences beyond the 10 CFR 72.106 limits for these postulated events.

8.2.6 EXPLOSION

Explosions are classified as human-induced or natural phenomena design events in accordance with ANSI/ANS 57.9 Design Events III and IV. The following explosion event categories have been evaluated (Reference 42) for the Diablo Canyon ISFSI:

(1) Detonation of a transporter or onsite vehicle fuel tank

(2) Detonation of propane bottles transported past the ISFSI storage pad

(3) Detonation of compressed gas bottles transported past the ISFSI storage pad

(4) Detonation of large stationary fuel tanks in the vicinity of the transport route

(5) Explosive decompression of a compressed gas cylinder

(6) Detonation of the bulk hydrogen storage facility

(7) Detonation of acetylene bottles stored on the east side of the cold machine shop

Events 1, 2, 3, and 5 are assumed to occur in the vicinity of the ISFSI storage pads, CTF, or transport route; and potentially affect both the overpack and the transfer cask. Events 4, 6, and 7 occur in the vicinity of the transport route and affect only the transfer cask.

As a result of its physical properties, diesel fuel does not pose any real explosion hazard. The pertinent material property for this determination, the flash point, is defined as the lowest temperature at which the vapor pressure of a liquid is sufficient to produce a flammable vapor/air mixture at the lower limit of flammability. In other words, a combustible liquid cannot vaporize sufficiently to detonate if the ambient temperature is below the flash point. Such materials could conceivably burn, but would be incapable of detonation.

The flash point of diesel fuel is 125°F. To be classified as flammable, the flash point of a liquid must be less than 100°F as discussed in the National Fire Protection Association Handbook (Reference 15). The highest ambient temperature predicted for the Diablo Canyon ISFSI site (5- to 10-year recurrence interval) is 104°F and would normally (99 percent of the time) be no more than 85°F. These temperatures are considerably less than the flash point of diesel fuel. Therefore, under ambient or normal operating temperature, diesel fuel oil does not represent a credible explosive hazard. Therefore, Event 1 for vehicles containing diesel fuel oil is excluded from further consideration.

Since the cask transporter is powered by diesel fuel, which cannot detonate as discussed above, explosion Event 1 is reduced to the explosion of onsite, gasoline-powered vehicles. The fuel tank capacity of these vehicles is an average of 20 gallons. Administrative controls are used to keep onsite gasoline-powered vehicles and tanker trucks carrying flammable liquids either: (a) at sufficient distance from the ISFSI storage pad (at all times), the CTF (while transferring an MPC), and the transport route during cask transport to ensure the total explosion overpressure is less than 1 psi, (b) a risk assessment will be performed using Regulatory Guide 1.91 risk acceptance criteria, or (c) diesel-powered vehicles will be used. The administrative controls include, but are not limited to, speed limits, single vehicle zones, no entry zones, no stopping zones, designated parking for various types of vehicles, and limitations on the size and contents of vehicles passing the ISFSI facility or the CTF during transport operations. In addition, vehicle movement is controlled in the vicinity of the transporter when it is transporting fuel. These administrative controls are further defined in the various referenced calculations provided in support of these sections.

To meet the Regulatory Guide 1.91 overpressure criterion, for a vehicle with a maximum of a 20-gallon fuel capacity, the separation distance to the ISFSI pads has been calculated as no less than 175 ft. As a result, there is the possibility that these vehicles may pass within that separation distance momentarily on their way past the ISFSI facility. No gasoline- powered vehicles are allowed to park or stop within 175 ft of the ISFSI. A probabilistic risk assessment (Reference 35) was performed and it was determined that, based on use of administrative controls and the restrictions for movement and stopping within the separation distance, the risk of exceeding the Regulatory Guide 1.91 overpressure criterion from vehicles driving past the ISFSI is insignificant.

In addition, although not considered credible, the potential explosion of a parked vehicle along the transport route was evaluated in a probabilistic risk assessment using Regulatory Guide 1.91 criteria (Reference 35). In that evaluation, it was determined that the risk was insignificant and not a credible source. This was based on the limited time the transporter is exposed (less than 10 hours per year), the lack of any ignition source in a parked car, the lack of a single vehicle explosion in the 35-year history of the Diablo Canyon project, and the administrative controls restricting vehicle movement during transport (no vehicle movement within 175 ft of the transporter).

All tanker trucks that pass the CTF/ISFSI facility are administratively limited to a maximum fuel volume of 800 gallons. These trucks will only be in this area momentarily while passing by the CTF/ISFSI facility and will be under administrative controls for their speed and continued movement through the area on their way to and from the vehicle maintenance shop that is located approximately 2,000 ft northeast of the ISFSI pad. A probabilistic risk assessment was performed (Reference 35) and it was determined, based on the use of administrative controls and the restrictions for movement and stopping within the 600-ft separation distance calculated for the 800-gallon volume based on the 1 psi Regulatory Guide 1.91 criterion, the risk is insignificant.
For Explosion Events 2 and 3, a probabilistic risk assessment was performed (Reference 35). The transport of gas bottles past the ISFSI pads is controlled by administrative controls and maintains the same separation distance as the 800-gallon fuel truck requirements. Under these controls and proper restraint of the bottles in transport, the risk of exceeding the Regulatory Guide 1.91 overpressure criteria was determined to be insignificant.

The large fuel tanks referred to in Event 4 are located along the main plant access road from the Avila Gate, approximately 1,200 ft from the onsite transport road at the closest point. The tanks include a 250-gallon propane tank, a 2,000-gallon diesel fuel tank, and a 3,000-gallon gasoline tank. The diesel fuel cannot detonate, so Event 4 is limited to the detonation of the 250-gallon propane and 3,000-gallon gasoline tanks. As shown in Section 8.2.6.2.1, Event 4 does not exceed the Regulatory Guide 1.91 1 psi criterion. These tanks will be periodically filled by standard tanker trucks with a capacity of three to four thousand gallons. The location of any tank truck is administratively controlled to ensure the total energy potentially received by the ISFSI is less than the design basis event. In addition, during cask transport between the plant and the CTF, all filling is suspended and all of the gasoline tanker trucks, which fill these tanks, are maintained greater than 1,100 ft from the transport route. This will be administratively controlled in accordance with the Diablo Canyon ISFSI TS Cask Transportation Evaluation Program.

Although the risk of a gas bottle explosion was found to be insignificant as discussed above, an Event 5 explosive decompression event for a compressed-gas cylinder was evaluated. The cylinder is evaluated as a projectile, similar to a tornado-generated missile and is discussed in Section 8.2.6.2.2.

Event 6 includes a potential source of detonation and is discussed in Section 8.2.6.2.3.

For Event 7, the probability of an explosion that would exceed the Regulatory Guide 1.91 criteria of 1 psi for the transporter is not considered credible and the hazard is bounded by the analysis of the hydrogen facility discussed in Section 8.2.6.2.3. This is documented in a probabilistic risk assessment (Reference 35) that determined that the risk from this hazard is not credible. This is based on the seismic procedural requirements for chaining the bottles in the upright position in the facility, the lack of any ignition sources in the area, the administrative controls eliminating vehicle movement when the transporter is in the area, and the limited exposure time of the transporter, which conservatively would be less than 10 hours per year.

8.2.6.1 Cause of Accident

There are several potential mechanisms for the initiation of the postulated explosion events listed above, including both human-induced (electrical shorts, vehicle accidents, transmission line strikes, etc.) and natural (lightning strikes, tornado missiles, etc.) phenomena. While the probability of occurrence of these mechanisms is expected to be very low, the credible explosion events are classified as ANSI/ANS 57.9, Design Event IV, and are evaluated.

8.2.6.2 Accident Analysis

8.2.6.2.1 Explosive Overpressure Due to Detonation Events

During a detonation event, the overpack and/or transfer cask would be subjected to an external overpressure. Regulatory Guide 1.91 states: "…for explosions of the magnitude considered in this guide and the structures, systems, and components that must be protected, overpressure effects are controlling." The magnitude of the overpressure would be a function of the calorific energy released and the distance between the overpack/transfer cask and the explosion source. Due to the extremely short duration of explosion events, any heat input to the casks would be negligible (fires are evaluated in Section 8.2.5).

Events 1 through 4 are evaluated under the following assumptions:

(1) The fuel tanks are ruptured, releasing all contained flammable material, and all spilled flammable liquids are completely vaporized.

(2) The flammable gas or vapor is mixed with air at the lower flammability limit of the material.

(3) The flammable fuel/air mixture is detonated, releasing a portion of the total heating value as a hemispherical overpressure wave front. The fraction of the available energy that contributes to the overpressure, called the explosive yield, is between 3 percent and 6 percent for hydrocarbon/air mixtures, as discussed in the Handbook of Chemical Hazards Analysis (Reference 17).

To determine the magnitude of the explosive overpressure incident on the overpack and transfer cask, the energy released during detonation is converted to an equivalent weight of trinitrotoluene (TNT). This is accomplished by dividing the explosion energy by the detonation energy of TNT, which is 4.5 megajoules per kilogram as discussed in Perry's Chemical Engineers' Handbook (Reference 18).

Once the equivalent weight of TNT is known, the explosive overpressure can be determined as a function of the separation distance between the explosion and the cask systems using a methodology developed by the U.S. Army (Reference 19) and endorsed by the NRC through Regulatory Guide 1.91. This methodology requires the calculation of a scaled ground distance, Z_G, which is the ratio of the physical separation distance divided by the cube root of the equivalent weight of TNT and has units of ft/lb$^{1/3}$. The incident overpressure at a given scaled ground distance is then obtained directly from Figure 2-15 of Reference 19.

For Event 4, the minimum physical separation distance to the transport route or the ISFSI is 1,200 ft based on the maximum quantities of flammable material having an equivalent weight of TNT of 12,100 lb, the resultant setback distance to ensure that the 1 psi maximum overpressure acceptance criteria is met per Regulatory Guide 1.91, is 1,033 ft. Therefore, further evaluation, is unnecessary.

The site-specific explosive overpressures caused by detonation events are bounded by the 1 psi Regulatory Guide 1.91 criterion or are determined not to be risk significant in accordance with Regulatory Guide 1.91. Therefore, 10 CFR 72.122(c) is met.

8.2.6.2.2 Missiles Due to Explosive Decompression of a Compressed Gas Cylinder

Although not considered a credible event, as discussed above, the missile created by the explosive decompression of a gas cylinder (Event 5) is evaluated assuming that a compressed gas cylinder under high-pressure is damaged such that the valve assembly located at the top of the cylinder breaks off. Expansion of the high-pressure compressed gas out of the hole in the cylinder accelerates the cylinder or valve assembly toward the cask systems, resulting in an eventual impact. Cylinders filled with acetylene, air, argon, helium, nitrogen, oxygen, and propane are evaluated.

The acceleration of the cylinder is dependent on the thrust force generated by the escaping high-pressure gas, which reduces over time as the cylinder internal pressure decreases. The thrust force as a function of time is determined from principles of compressible flow, which state that the thrust force is the product of the mass flow and velocity of the gas escaping through the hole in the cylinder wall. While the internal pressure of the cylinder is sufficiently high (that is, greater than the critical pressure), the velocity of the gas is limited to the speed of sound (that is, sonic or choked flow). As the pressure falls below the critical pressure, the velocity becomes subsonic, and eventually reaches zero when the cylinder internal pressure is equal to the atmospheric pressure.

Conservatively neglecting aerodynamic drag (which would decrease the maximum velocity of the cylinder by opposing the thrust force), and assuming bounding discharge coefficients, the cylinder is determined to accelerate from rest to a maximum of approximately 109 mph as the internal pressure drops toward ambient pressure (propane gas). The detached valve assembly is determined to accelerate to a maximum of approximately 342 mph (all gases equal).

Section 8.2.2 presents evaluations of the impact of tornado missiles on both the loaded overpack and the transfer cask. Using the same energy method employed in Section 8.2.2, the effects of the impact of cylindrical missiles are evaluated. The maximum penetration into a steel target for the cylinder and valve assembly missiles is less than 1/4 inch. These penetrations are insufficient to completely penetrate either a storage overpack or a transfer cask, thereby precluding damage to the MPC confinement boundary. These missile evaluations conclude that neither the loaded overpack nor the transfer cask undergoes any significant reduction of structural integrity and no shielding material (concrete and water) is damaged or lost, such that the licensing basis acceptance criteria for the casks is met.

8.2.6.2.3 Potential Explosion Event at the Bulk Hydrogen Facility

As shown in Figure 2.2-1, a bulk hydrogen facility is located east of the FHB/AB and approximately 0.14 miles from the ISFSI pad with its elevation several hundred feet below the ISFSI facility. Therefore the hydrogen facility can only potentially affect the transport of fuel and not the ISFSI facility. This hydrogen facility contains 6 tanks for a total of about 300 cubic ft and is near the transport route from where the transfer cask enters and leaves the Unit 2 FHB/AB. These tanks are refilled approximately twice a month. They are held in a seismic-qualified rack, which is enclosed in a seismic-qualified vault. The vault is only open on the side toward the FHB/AB and is provided with a 12-inch-diameter top vent to ensure no possible buildup of gas from leakage. This facility is designed to protect against over pressurization, excessive flow, and vehicle (delivery truck) damage during filling. The transporter will only be in this area for a very short period of time, and during this time, all filling of tanks is suspended and all vehicle movement is administratively controlled in accordance with the Diablo Canyon ISFSI TS Cask Transportation Evaluation Program. A probabilistic risk assessment (Reference 35) was performed in accordance with the Regulatory Guide 1.91 methodology. Due to the noncredible nature of an explosion and the limited exposure to the transporter, the event is not risk significant using the Regulatory Guide 1.91 acceptance criteria and is considered acceptable.

8.2.6.3 Accident Dose Calculations

As discussed above, the effects of the Diablo Canyon site explosion events involving detonation (Events 1, 2, 3, 4, and 7) are enveloped by the design-basis accident conditions (explosion and transfer cask side drop) in the HI-STORM 100 System FSAR or are not considered risk significant in accordance with Regulatory Guide 1.91. The missile evaluation for Event 5 concludes that only a small amount of the shielding materials may be damaged or lost. The structural evaluations in Chapter 3 of the HI-STORM 100 System FSAR confirm that the MPC confinement boundary remains intact and the shielding effectiveness of the HI-STORM 100 System is not significantly affected by these explosion and missile events. The radiological evaluations presented in Chapter 11 of that document also conclude that the loaded overpack and transfer cask continue to meet the accident dose limits of 10 CFR 72.106 at the controlled area boundary after these events.

8.2.7 LEAKAGE THROUGH CONFINEMENT BOUNDARY

This section only applies to the initial 16 casks loaded at the Diablo Canyon ISFSI. Following construction of the first 16 casks, the testing requirement for the MPC boundary welds was changed to the leaktight criteria of ANSI N14.5-1997. The helium leak testing requirements for the vent and drain port cover plate welds had been changed to the “leaktight” criteria of ANSI N14.5-1997 in LA 1. The lid-to-shell (LTS) weld is a large, multi-pass weld which is placed and inspected in accordance with ISG-15; therefore, in accordance with ISG-18, leakage from this weld is considered non-credible. Because all the closure welds meet a leaktight criteria, the confinement boundary of the subsequently fabricated MPCs can be considered leak tight.

The hypothetical leakage of a single, loaded MPC-32 under accident conditions, where the cladding of 100 percent of the fuel rods is postulated to have ruptured, is described in this section.

8.2.7.1 Cause of Accident

The analyses presented in Chapters 3 and 11 of the HI-STORM 100 System FSAR demonstrate that the MPC confinement boundary remains intact during all hypothetical accident conditions, including the associated increased internal temperature and pressure due to the decay heat generated by the stored fuel.

This section evaluates the consequences of a non-mechanistic, 100-percent, fuel-rod rupture and confinement boundary leak (Reference 43). The breach could result in the release of gaseous fission products, fines, volatiles, and airborne crud particulates to the MPC cavity. Doses resulting from the canister leakage under hypothetical accident conditions were calculated in accordance with Interim Staff Guidance (ISG) Document 5 (Reference 20), ISG 11 (Reference 21) and NUREG/CR-6487 (Reference 22).

8.2.7.2 Accident Analysis

8.2.7.2.1 Confinement Vessel Releasable Source Term

The MPC-32, which holds 32 PWR fuel assemblies, is used in the confinement analysis because it bounds the other, lower-capacity Holtec PWR MPCs for the total quantity of radionuclides available for release from a single cask. The methodology for calculating the spent fuel isotopic inventory for an MPC-32 is detailed in Section 7.2.2. A summary of the isotopes available for release is provided in Table 7.2-8.

8.2.7.2.2 Release of Contents under Accident Conditions of Storage

In this hypothetical accident analysis, it is assumed that 100 percent of the fuel rods have developed cladding breaches, even though, as described below, the spent fuel is stored in a manner such that the spent fuel cladding is protected against degradation that could lead to fuel rod cladding ruptures. The MPC cavity is filled with helium after the MPC has been evacuated of air and moisture that might produce long-term degradation of the spent fuel cladding. Additionally, the HI-STORM 100 System is designed to provide for long-term heat removal capabilities to ensure that the fuel is maintained at a temperature below those at which cladding degradation occurs. It is, therefore, highly unlikely that a spent fuel assembly with intact fuel rod cladding will undergo cladding failure during storage, and the assumption that 100 percent of the fuel rods have ruptured is extremely conservative.
The assumption that 100 percent of the fuel rods have ruptured is incorporated into the postulated pressure increase within the MPC cavity to determine the maximum possible pressure of the MPC cavity. This pressure, combined with the maximum MPC cavity temperature under accident conditions, is used to determine a postulated leakage rate during an accident. This leakage rate is based on the leakage rate limit of $\leq 5.0 \times 10^{-6}$ atm-cm^3/sec for the helium-leak-rate test, and is adjusted for the higher temperature and pressure during the accident to result in a hypothetical accident leak rate of 1.28×10^{-5} cm^3/sec.

The radionuclide release fractions, which account for the radionuclides trapped in the fuel matrix and radionuclides that exist in a chemical or physical form that is not releasable to the MPC cavity from the fuel cladding, are based on ISG-5. Additionally, only 10 percent of the fines released to the MPC cavity are assumed to remain airborne long enough to be available for release through the confinement boundary based on SAND88-2778C (Reference 23). It is conservatively assumed that 100 percent of the volatiles, crud, and gases remain airborne and available for release. The release rate for each radionuclide was calculated by multiplying the quantity of radionuclides available for release in the MPC cavity by the leakage rate calculated above, divided by the MPC cavity volume. No credit is taken for any confinement function of the fuel cladding or the ventilated overpack.

8.2.7.3 Dose Calculations for Hypothetical Accident Conditions

Doses at the Diablo Canyon ISFSI site boundary resulting from a postulated leaking MPC-32 were calculated using an inhalation and submersion pathway. An ingestion pathway is not included because of the lack of broadleaf vegetation within 4 miles of the site boundary; the lack of fresh surface water; the lack of milk animals or a credible meat pathway within 800 meters of the ISFSI site; and the very low population within a 6-mile radius of the site. The nearest distance from the ISFSI to the DCPP is 1,400 ft. A χ/Q value of 4.50×10^{-4} s/m^3 was assumed. This χ/Q value is conservative because it is based on a 1-hour release period, whereas the hypothetical accident duration is 30 days per ISG-5. The dose conversion factors for internal doses due to inhalation and submersion in a radioactive plume were taken from EPA Federal Guidance Report No. 11 (Reference 24) and EPA Federal Guidance Report No. 12 (Reference 25), respectively. An adult breathing rate of 3.3×10^{-4} m^3/s was assumed.

Doses to an individual present continuously for 30 days were calculated assuming a release from a single cask with the wind blowing constantly in the same direction for the entire duration. The following 30-day doses were determined:

- The committed dose equivalent from inhalation and the deep dose equivalent from submersion for critical organs and tissues (gonad, breast, lung, red marrow, bone surface, thyroid)
- The committed effective dose equivalent from inhalation and the deep dose equivalent from submersion for the whole body
- The lens dose equivalent for the lens of the eye
- The shallow dose equivalent from submersion for the skin
- The resulting total effective dose equivalent and total organ dose equivalent.

The doses were calculated, as appropriate, for both inhalation and submersion in the radioactive plume. Doses due to exposure to soil with ground surface contamination and contamination to a depth of 15 cm have been evaluated generically for the HI-STORM 100 System. The dose due to ground contamination was found to be negligible compared to those resulting from submersion in the plume and are not reported here (HI-STORM 100 System FSAR, Section 7.2.8).

Table 8.2-12 summarizes the accident doses for a hypothetical confinement boundary leak. The estimated doses are a fraction of the limits specified in 10 CFR 72.106(b).

8.2.8 ELECTRICAL ACCIDENT

Electrical accidents considered include a lightning strike and a 500-kV transmission line drop. Both events are postulated to apply high voltage electrical current through the overpack or the transfer cask, the effects of which are evaluated in Reference 44. These events are classified as natural phenomena, Design Event IV, in accordance with ANSI/ANS 57.9.

8.2.8.1 Cause of Electrical Accident

Lightning strikes are natural phenomena caused by meterological conditions conducive to the discharge of large amounts of static electricity to ground. The 500-kV transmission line drop is postulated as a result of a transmission tower collapse or transmission line hardware failure near the ISFSI storage site and the CTF. The worst-case fault condition for a cask is that which places a cask in the conduction path for the largest current. This condition is the line drop of a single conductor of one phase with resulting single, line-to-ground fault current and voltage-induced arc at the point of contact.

A number of transmission line failure modes were postulated. These included the break or drop of: a single conductor of one phase, both conductors of a single phase, and all three phases. The failure modes considered are:

(1) Three-phase drop onto cask structures - The fault would be balanced, most current would return through the phase conductors and only a small amount would pass through the casks and into the earth.

(2) Both conductors of one phase fall onto one cask - The single line-to-ground fault would split evenly between the two conductors (spaced at 18 inches) and effectively reduce the energy at the point of contact by a factor of two. Therefore, it would create two points of contact, each dissipating half the energy.

(3) One conductor of one phase breaks into two and each end falls onto separate casks or onto different points of the same cask - The single, line-to-ground fault would split between the two points of contact reducing the energy at each point of contact.

(4) One conductor falling while remaining intact - The single, line-to-ground fault would be forced into one point of contact, through the cask, and into the earth/ground grid. All energy would be forced to dissipate at this one point. This would be the worst-case for the cask systems.

Protective relaying is assumed to actuate on arc initiation. The time duration from relay actuation to breaker opening is assumed to be 0.1 sec (6 cycles).

8.2.8.2 Electrical Accident Analysis

The overpack and the CTF are sited beneath a 500-kV transmission line. The transmission line connects the Unit 1 main generator to the 500-kV switchyard. The transmission line is protected from direct lightning strikes by two shield wires installed above the line. Similarly, the transmission conductors provide lightning protection for the overpack and the CTF. The transmission lines themselves act as shield wires for metal objects located below them and within their effective shield angle. Inside this effective shield angle, the distance from the lightning arc to the line will be less than from the lightning arc to the top of the cask, and all lightning within this zone will hit the transmission line instead of the cask. Outside of this effective shield angle, the lightning will be so close to the ground that it will directly hit the ground before it strikes any metal object. Thus, the overhead transmission line prevents a direct lightning strike on any overpack or the CTF. Even so, the effects of a lightning strike are evaluated.
The cask transporter provides protection for the transfer cask from direct lightning strikes and transmission line drops. The gantry and rigging metal is sufficiently above the cask material that any line drop would be effectively deflected by this metal before it is able to contact the cask surface.

For the evaluation of the lightning strike, direct atmospheric lightning strikes on the overpack and the transfer cask are postulated. The lightning strike, defined by a current versus time profile, is defined by standard industry practice as a peak current of 250 kiloamps for 260 microseconds followed by a continuing current of 2 kiloamps for 2 additional seconds.

For the evaluation of the 500-kV transmission line drops for both the overpack and the transfer cask, it is postulated that while both DCPP units are operating at full power a single overhead transmission conductor falls onto a cask. The 500-kV system is operated at a nominal voltage of 525-kV phase to phase. The line-to-ground voltage is 303-kV. The transmission line drop sequence of events is defined in three distinct time periods as follows:

- Period 1 free air arc (wire falling but not yet touching cask) - voltage drops from 303 kV to 1 kV and current rises from 0 kiloamps to 18.6 kiloamps over a 0.05 second arc duration.

- Period 2 prior to breaker trip (wire in solid contact with the cask but breaker not yet fully open) - voltage and current are constant at 1 kV and 18.6 kiloamps, respectively, over a 0.05 second breaker trip duration.

- Period 3 during generator coast-down (all breakers open, faulted generator still contributing fault current) - voltage and current are constant at 0.2 kV and 5.08 kiloamps, respectively, over a generator, coast-down duration of 3.9 seconds.

Both electrical events result in an electrical discharge that travels along the least resistive path through the cask to the ground. Both the lightning strike and the transmission line drop originate external to the casks, so the least resistive path for both the overpack and the transfer cask will be through the outermost shell (that is, overpack outer shell and transfer cask enclosure shell). The MPC contained within an overpack or transfer cask will, therefore, be protected from any electrically-induced damage.

For the postulated lightning strike, the electrical discharge deposited into the cask and conducted to ground must overcome the inherent electrical resistance of the conducting material. This resistance to current flow generates heat, called resistance or Joulean heating, and is governed by the following formula:

$$E = I^2 \times t \times R$$

where E is the resistance heat energy, I is the current, t is the current duration and R is the material resistivity. The electrical resistivity value for iron (10 $\mu\Omega$-cm) was obtained from the CRC Handbook of Chemistry and Physics and conservatively increased by 20 percent to obtain an estimated value of 12 $\mu\Omega$-cm for steel, which was used in the lightning strike analysis. Even if the resistivity were doubled from this value (to 24 $\mu\Omega$-cm), the temperature rise from the lightning event would still be less than 1°F.

The heat generated by resistance heating must be absorbed by sensible heating of the affected cask component, governed by the following equation:

$$E = m \times c_p \times \triangle T$$

where m is the mass of the cask component, c_p is the material heat capacity and ΔT is the component temperature rise. These two equations can be used to determine the cask component temperature rise for each cask, the results of which are contained in Table 8.2-13.
All of the computed, electrically-induced, temperature-rise values are less than 1°F. The HI-STORM 100 System FSAR contains evaluations of both the overpack and the transfer cask under normal temperature conditions. The increase in outer shell temperature for both structures is well below the normal temperature condition limits. Accident condition temperature limits for the outer shells of both casks are significantly higher than the normal condition limits. It is therefore concluded that the postulated lightning strike will not cause the affected cask components to exceed either normal or accident condition temperature limits and do not adversely affect the performance of either system.

For the postulated transmission line break, because of the significant influence of the time-varying voltage and the longer time periods involved, a slightly different method of calculating the energy input is used. The electrical energy is governed by the following formula:

$$E = \int_t V(t) \times I(t) dt$$

where V(t) is the time-varying voltage function, I(t) is the time-varying current function and t is the independent time variable. The electrical energy is calculated separately for each time period of the postulated electrical profile.

As the transmission line drops onto a cask, the predominant portion of arc energy is dissipated to the atmosphere, with the remaining portions heating the cask and vaporizing a portion of the steel outer shell. During the arc phase (Period 1) of the postulated accident, it is conservatively assumed that 10 percent of the total energy is dissipated in sublimating (vaporizing) steel at the point of arc, 40 percent of the total energy is dissipated in resistance heating of the affected cask component, and the balance of the arc energy is dissipated to the environment. During the breaker trip and generator coast-down periods (Periods 2 and 3) of the postulated accident, it is conservatively assumed that all energy is dissipated in resistance heating of the affected cask component. The results of these evaluations are contained in Table 8.2-14.

With respect to the computed, electrically-induced, temperature rise values, the HI-STORM 100 System FSAR contains evaluations of both the overpack and the transfer cask under normal temperature conditions. Again, the increase in the outer shell temperature of both structures is well below the normal condition temperature limits. Accident condition temperature limits for these components for both casks are significantly higher than the normal condition limits.

The sublimated hole diameters are calculated assuming that a cylindrical plug of material, with a length equal to the thickness of the component material, is vaporized. Even if a hole is sublimated in the overpack outer shell, there are no negative thermal consequences. Behind the steel outer shell is a thick concrete layer that is unlikely to be significantly affected given the rapidity of the event and the low thermal diffusivity of concrete. Experience with high-fault currents has shown that spalling and crystallization of the concrete surface would be expected at the point of contact of the fault. The maximum depth of the concrete plug affected would be less than the diameter of the surface hole. It should also be noted that the existence of a hole in the overpack outer shell was postulated and evaluated in Section 8.2.2. The cause of the hole in that section was due to a hypothesized tornado missile. Should a hole be formed in the transfer cask, the water jacket used to provide shielding and to help maintain cool conditions inside the MPC could be drained. This condition has an insignificant thermal impact, and the shielding impact is already addressed in Section 8.2.11 and was found to be acceptable. Section 8.2.11 considers a loss of water jacket without considering any specific cause.

These results are considered bounding for the design life of the ISFSI. Even if the fault current increases over the life of the facility, the results remain valid because the resulting damage increase would not be significant. The line-to-ground voltage is the predominant factor in arc ignition. An increase in fault current would have minimal consequences. A larger hole size does not change the radiological dose consequences because there is minimal damage to the concrete shielding in the overpack, no damage to the lead shielding in the transfer cask, and no damage to the inner steel liners in both the overpack and the transfer cask.

It is concluded that the postulated transmission line break will not cause the affected cask components to exceed either normal or accident condition temperature limits and that localized material damage at the point of arc is bounded by accident conditions discussed in Sections 8.2.2 and 8.2.11. As a result of these considerations, it is concluded that the postulated transmission line drop does not adversely affect the thermal performance of either system.

8.2.8.3 Electrical Accident Dose Calculations

The postulated electrical events are shown to result in a negligible increase in the temperatures of the affected components and damage to a small amount of material in the localized area of arc. The resulting temperatures would remain bounded by both the normal and accident condition temperature limits.

The small loss of material is negligible compared to the total mass of shielding materials, so there would be no significant increase in overall cask dose rates. As noted above, the concrete behind the overpack outer shell would not likely be affected. Thus, the change in shielding would be negligible. In any event, a more limiting condition is evaluated in Section 8.2.2.

In the case of the transfer cask, there would be an increase in radiation doses adjacent to the cask should the shielding water in the water jacket be lost. The loss of neutron shielding is evaluated in Section 8.2.11. The addition of a hole in the transfer cask outer shell would have a negligible impact on dose. The impact on personnel exposures is considered to be negligible.
The MPC is protected from electrical damage by the overpack. Thus, there is no release of the contained radioactive material from the MPC. Doses to persons located offsite are not affected by these events.

8.2.8.4 Conclusions

The postulated electrical events may possibly result in a small hole in either the overpack or the transfer cask. Both conditions are conservatively bounded by previously analyzed events in Sections 8.2.2 and 8.2.11.

8.2.9 LOADING OF AN UNAUTHORIZED FUEL ASSEMBLY

The Diablo Canyon ISFSI TS and Section 10.2 specify limiting values for the initial enrichment, burnup, decay heat, and cooling time after reactor discharge for the fuel assemblies to be placed into the MPCs. The possibility of storing a fuel assembly that does not meet the Diablo Canyon ISFSI TS and Section 10.2 has been considered.

8.2.9.1 Cause of Loading an Unauthorized Fuel Assembly

Procedures are used to administratively control and document the planning and loading of all DCPP fuel assemblies to be stored in each overpack. The cause of this event is postulated to be an error during spent fuel planning or loading operations (for example, a planning error occurs in selecting the fuel assembly to be stored or the wrong fuel assembly is loaded into an MPC).

8.2.9.2 Analysis of the Loading of an Unauthorized Fuel Assembly

The chance of loading of an unauthorized fuel assembly is greatly minimized because of the multiple administrative controls imposed via procedures to ensure a fuel planning or loading error does not remain undetected. These procedures prescribe how the planning is performed and verified to ensure the characteristics of selected fuel assemblies are within the applicable Diablo Canyon ISFSI TS and Section 10.2 limits. Likewise, the spent fuel loading procedures require that a final verification of the identity and location of fuel assemblies be performed prior to placing the lid on the MPC. These procedures are part of the ISFSI operational procedures described in Section 9.4.1.1.4.

The loading of an unauthorized fuel assembly has no consequence while the transfer cask/MPC assembly remains in the spent fuel pool (SFP) as explained below. The borated water in the SFP provides adequate protection against a criticality event, and also provides shielding and heat removal. Loading of an unirradiated fuel assembly will not cause a criticality event because the MPC design precludes criticality assuming all loaded fuel assemblies are unirradiated (that is, no burnup credit taken). Loading of a fuel assembly with gross cladding defects will not cause further damage to the cladding or result in the release of radioactive material. Loading of a fuel assembly with structural defects will likely be detected during placement into the MPC. These events will not go undetected because fuel condition will be verified as part of the loading process.

8.2.9.3 Conclusion

As discussed above, the use of procedures, which prescribe and verify the rigorous planning and loading activities, provides reasonable assurance that only fuel assemblies meeting Diablo Canyon ISFSI TS and Section 10.2 requirements will be loaded for storage.

8.2.10 EXTREME ENVIRONMENTAL TEMPERATURE

Extreme environmental temperature is classified as a natural phenomenon Design Event IV as defined in ANSI/ANS-57.9. The extreme environmental temperature accident involves the postulation of an unusually high ambient temperature at the Diablo Canyon ISFSI site. Unlike the off-normal high temperature evaluated in Section 8.1.2, the postulated, extreme-high temperature is beyond what can be reasonably expected to occur over the life of the ISFSI and represents a bounding, worst-case scenario.

8.2.10.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature event for the HI-STORM 100 System is analyzed at an environmental temperature of 125°F in Reference 63 and at -40°F in Section 4.4.3 of the HI-STORM 100 System FSAR. To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System, with its corresponding large thermal inertia and the limited duration for the extreme temperature, this assumption is conservative.

8.2.10.2 Extreme Environmental Temperature Analysis

8.2.10.2.1 Upper Temperature Limit

The accident condition considered in Reference 67 assumes an extreme environmental temperature of 125°F for a duration sufficient to reach thermal equilibrium. This bounds the extreme-maximum-site ambient temperature for the Diablo Canyon ISFSI site of 104°F (Section 3.4.). This condition is evaluated with respect to accident condition component design temperatures listed in Table 2.2.3 of the HI-STORM 100 System FSAR. The evaluation was performed with considering baseline conditions (steady state conditions, normal ambient temperature and the maximum design decay heat load of 28.74 kW) the temperatures of the HI-STORM 100SA system are conservatively assumed to rise by the difference between the extreme and normal ambient temperatures.

These temperatures are calculated at a normal environmental temperature of 65°F. The extreme environmental temperature is 125°F, which is an increase of 60°F. This event is simplistically evaluated by adding the 60°F difference to each of the limiting normal component temperatures. This yields conservatively bounding temperatures for all of the HI-STORM 100 System components because the thermal inertia of the HI-STORM 100 System is not credited. The resulting component temperatures under extreme environmental temperature condition are listed in Table B.5.7 of Reference 67. As illustrated by the table, all the temperatures are well below the accident-condition, design-basis component temperatures. Since the extreme environmental temperature is of a short duration (several consecutive days would be highly unlikely), the resultant temperatures are evaluated against short-term accident condition temperature limits.

Therefore, the HI-STORM 100 System component temperatures meet design requirements under the extreme environmental temperature condition.

Additionally, the effect of extreme environmental temperature on MPC internal pressure was evaluated. The resultant pressure, from Table B.5.10 of Reference 67, is calculated as 84.3 psig which is below the accident design pressure of 200 psig.

8.2.10.2.2 Lower Temperature Limit

The HI-STORM 100 System was also evaluated for a -40°F extreme low ambient temperature condition, as discussed in Section 4.4.3 of the HI-STORM 100 System FSAR. Zero decay heat generation from spent fuel and no solar insolation were conservatively assumed. All materials of construction for the MPC and overpack will perform their design function under this extreme cold condition. Since the minimum temperature at the Diablo Canyon ISFSI is greater than or equal to 24°F (Table 3.4-1), the extreme low ambient temperature evaluation in the HI-STORM 100 System FSAR bounds the conditions at the Diablo Canyon ISFSI.

8.2.10.3 Extreme Environmental Temperature Dose Calculations

The extreme environmental temperature range at the Diablo Canyon ISFSI will not cause the overpack concrete to exceed its normal design temperature. Therefore, there will be no degradation of the concrete shielding effectiveness. The extreme temperature range will not cause a breach of the confinement system and the short-term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact on the HI-STORM 100 System for the extreme environmental temperature range, and the dose rates under this accident condition are equivalent to the normal condition dose rates.

8.2.10.4 Extreme Environmental Temperature Corrective Action

There are no consequences of this accident that require corrective action.

8.2.11 HI-TRAC TRANSFER CASK LOSS-OF-NEUTRON SHIELDING

This accident event postulates the loss-of-neutron shielding provided by the transfer cask water jacket and the Holtite-A solid neutron shielding in the transfer cask top lid. A loss-of-neutron shielding is classified as a Design Event IV, as defined in ANSI/ANS-57.9.

8.2.11.1 Cause of Loss-of-Neutron Shielding

Throughout all design-basis-accident conditions, the axial location of the fuel will remain fixed within the MPC because of the upper fuel spacers. Chapter 3 of the HI-STORM 100 System FSAR shows that the fuel spacers, transfer cask inner shell, lead, and outer shell remain intact throughout all design-basis normal, off-normal, and accident loading conditions. (The 10 CFR 50 LAR and license amendments [References 53 and 54, respectively] in support of the Diablo Canyon ISFSI addresses the effect of lead slump on the transfer cask shielding after a vertical drop inside the FHB/AB.) Localized damage of the transfer cask outer shell could be experienced, but no loss of shielding results.

Two potential causes for the loss of neutron shielding provided by the transfer cask are:

(1) Elevated temperatures as a result of a fire accident could result in the temperature of the Holtite-A exceeding the design-accident temperature. The pressure of the water jacket could also increase due to a fire, to the point where the overpressure relief valve on the water jacket would vent steam and water to the atmosphere. This would result in the loss of some amount of the water used for neutron shielding.

(2) Puncture of the transfer cask outer neutron shield jacket by a small object traveling at high speed, such as a tornado-borne missile, would cause the shield water to drain out at the point of puncture.

Other shielding credited in the shielding analyses includes the steel transfer cask and overpack structures, concrete, and lead. There are no credible events that could cause a significant degradation or loss of these solid forms of shielding.

8.2.11.2 Loss-of-Neutron Shielding Analysis

In the transfer cask, which uses Holtite-A in the top lid for neutron shielding, a fire could cause the Holtite-A to exceed its design-accident-temperature limit. For the dose analysis, it is conservatively assumed that all of the Holtite-A in the transfer cask top lid is lost. The potential reduction in shielding effectiveness of the Holtite-A in the transfer cask top lid results in a dose rate that is bounded by the normal dose rates in the area of the access hole in the transfer cask top lid. Therefore, no additional evaluation of this scenario is required.

The bounding consequence that affects the shielding materials of the transfer cask is the potential for damage to the water jacket shell and the loss of all of the neutron shield (water). In the accident consequence analysis, it is conservatively assumed that the neutron shield (water) is completely lost and replaced by a void. The assumed loss of all water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding analysis results presented in Section 5.1.2 of the HI-STORM 100 System FSAR demonstrate that the dose limits of 10 CFR 72.106 are not exceeded if all of the water in the water jacket is lost. It is shown in Section 11.2.4 of the HI-STORM 100 System FSAR that the increase in fuel cladding and component material temperatures due to the loss of water in the water jacket do not cause the short-term fuel cladding or material temperature limits listed in the HI-STORM 100 System FSAR Table 2.2.3 to be exceeded. The internal MPC pressure also remains below the 200-psig-accident design limit. Therefore, there is no affect on the integrity of the MPC confinement boundary.

8.2.11.3 Loss-of-Neutron Shield Dose Calculations

The complete loss of the transfer cask neutron shield along with the water-jacket shell is assumed in the shielding analysis for the post-accident analysis of the loaded transfer cask in Section 5.1.2 of the HI-STORM 100 System FSAR. As shown therein, the complete loss of the transfer cask neutron shield significantly affects the dose rate at mid-height of the transfer cask, and the accident dose rate (calculated using the burnups and cooling times that produce the highest dose rates) is 1.47 mrem/hr at an assumed distance of 100 meters from the ISFSI storage pad. For the 30-day duration of the event, the total dose at this location is 1.058 rem, which is less than the accident dose limit in 10 CFR 72.106. The minimum distance to the controlled-area boundary at the Diablo Canyon ISFSI is approximately 1,400 ft (430 m). Therefore, the generically-calculated doses for this accident from the HI-STORM 100 System FSAR bound those for the Diablo Canyon ISFSI site.

Doses to onsite personnel will be monitored after a loss-of-neutron shielding event and temporary shielding may be employed at the discretion of the DCPP radiation protection organization.

8.2.12 ADIABATIC HEAT-UP

This noncredible accident event postulates that the loaded overpack is unable to reject heat to the environment through conduction, convection, or radiation. This is classified as a Design Event IV, as defined by ANSI/ANS 57.9.

8.2.12.1 Cause of Accident

There is no credible accident that could completely stop heat transfer from the overpack to the environment. Even if the overpack were to be completely buried, with the inlet and outlet vent ducts blocked, some heat transfer would occur via conduction through the overpack structure and the material covering the overpack, and through convection at the surface of the outer material. The Diablo Canyon ISFSI site is located where a portion of the hill has been excavated (Figure 2.1-2). The slope protection of the hill adjacent to the storage pads (Section 4.2.1.1.9) precludes a landslide that completely covers one or more casks on the ISFSI pads. Should a slide occur, minor amounts of material could be removed before excessive heat up would occur. Also, there are no sources of volcanic activity or large amounts of debris located above, and sufficiently close to, the ISFSI site that could cause a complete covering of one or more casks on the ISFSI pads. This is a non-mechanistic accident and is evaluated to yield the most conservative response of the HI-STORM 100 System.

8.2.12.2 Accident Analysis

Section 11.2.14 of the HI-STORM 100 System FSAR discusses the "Burial-Under-Debris" accident, which is modeled as an adiabatic heat-up event. The analysis of this event is summarized below.

Burial of the loaded overpack does not impose a condition that would have more severe consequences for criticality, confinement, shielding, and structural analyses than that performed for the other accidents analyzed. The debris would provide additional shielding to reduce radiation doses. The accident external pressure encountered during the flooding accident (Section 8.2.3) bounds any credible pressure loading caused by the burial under debris.

Burial under debris can affect thermal performance because the debris acts as an insulator and heat sink. The insulating effect will cause the HI-STORM 100 System and fuel cladding temperatures to increase. A thermal analysis has been performed to determine the time for the fuel cladding temperatures to reach the short-term, accident-condition temperature limit during a burial under debris accident.

To demonstrate the inherent safety of the HI-STORM 100 System, a bounding analysis that considers the debris to act as a perfect insulator is considered. Under this scenario, the contents of the HI-STORM 100 System will undergo a transient heat up under adiabatic conditions. The minimum time required for the fuel cladding to reach the short-term, design, fuel-cladding-temperature limit depends on the amount of thermal inertia of the cask, the cask initial conditions, the spent nuclear fuel decay heat generation and the margin between the initial cladding temperature and accident temperature limit.

This evaluation is performed in Section B.5.4 of Reference 67 and determined a substantial allowed burial time of 72.7 hours. In addition, Table B.5.6 of Reference 67 demonstrates that all component temperatures remain below the accident temperature limit for a 32 hour blocked air inlet duct event.

8.2.12.3 Accident Dose Calculations

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event. As discussed in burial-under-debris analysis, the shielding is enhanced while the HI-STORM 100 System is covered. The elevated temperatures will not cause the breach of the confinement system and the short-term, fuel-cladding-temperature limit is not exceeded. Therefore, there is no radiological impact.

8.2.13 PARTIAL BLOCKAGE OF MPC VENT HOLES

Each MPC basket fuel cell wall has elongated vent holes at the bottom and top. These holes facilitate the natural circulation of helium inside the MPC for convection heat transfer. The partial blockage of the MPC basket vent holes accident has been evaluated to determine the effects on the HI-STORM 100 System due to the reduction in the size of the vent openings. This accident condition is discussed in Section 11.2.5 of the HI-STORM 100 System FSAR.

8.2.13.1 Cause of Partial Blockage of MPC Vent Holes

After the MPC is loaded with spent nuclear fuel, the MPC cavity is drained, dried, and backfilled with helium. There are three possible sources of material that could block the MPC basket vent holes. These are the fuel cladding, fuel pellets, and crud. Gross fuel cladding rupture is precluded by design in accordance with 10 CFR 72.122(h)(1). Due to the maintenance of relatively low cladding temperatures during storage, it is not credible that the fuel cladding would rupture and that fuel cladding and fuel pellets would fall to block the basket vent holes. Damaged fuel and fuel debris are stored in damaged fuel containers, which have screens to minimize the dispersal of gross particulates. However, it is conceivable that a percentage of the loose crud deposited on the external surfaces of the fuel rods may fall away and deposit at the bottom of the MPC.

Helium in the MPC cavity provides an inert atmosphere for storage of the fuel. During normal storage operations, the design of the HI-STORM 100 System maintains the peak fuel rod cladding temperature below the required long-term storage limits. There are no credible, design-basis accidents that cause the fuel assembly to experience a deceleration loading greater than the limits established in the HI-STORM 100 System FSAR, Section 3.5. (As discussed in Section 8.2.4, the load portions of the transporter and the lifting devices attached to the transfer cask and overpacks are designed to preclude drop events.)

Crud can be made up of two types of layers, namely, loosely-adherent and tightly-adherent. The fuel assembly movement from the fuel racks to the MPC, and subsequent movement of the MPC during cask loading, transfer, and transport operations, may cause a portion of the loosely-adherent crud to fall away. The tightly-adherent crud remains in place during ordinary fuel handling operations.

8.2.13.2 Analysis of Partial Blockage of MPC Vent Holes

The MPC vent holes that act as the bottom plenum for the MPC internal helium circulation are of an elongated, semi-circular design to ensure that the flow passages will remain open under a hypothetical shedding of the crud on the fuel rods. For conservatism, only the minimum semi-circular hole area is credited in the thermal models (that is, the elongated portion of the hole is completely neglected).

The amount of crud on fuel assemblies varies greatly from plant to plant. The maximum crud depths calculated for each of the MPCs is listed in Table 2.2.8 of the HI-STORM 100 System FSAR. The maximum amount of crud was assumed to be present on all fuel rods within the MPC. Both the tightly- and loosely-adherent crud was conservatively assumed to fall off of the fuel rods. The assumed crud depth does not totally block any of the MPC basket vent holes as the crud accumulation depth is less than the elongation of the vent holes. Therefore, the remaining cross-sectional flow area through the vent holes area is greater than that used in the thermal models.

The partial blockage of the MPC basket vent holes has no effect on the structural, confinement, and thermal analysis of the MPC. There is no significant effect on the shielding analysis because the source term from the crud is enveloped by the source term from the fuel and the activated nonfuel hardware of the fuel assemblies. As the MPC basket vent holes are not completely blocked, preferential flooding of the MPC fuel basket is not possible during draining operations and, therefore, the criticality analyses are not affected.

8.2.13.3 Dose Calculations for Partial Blockage of MPC Vent Holes

Partial blockage of basket vent holes will not result in a compromise of the confinement boundary because the thermal model accounts for the partial blockage. Fuel decay heat, burnup, and cooling time limits in Section 10.2 are determined accordingly to ensure that the cask heat transfer remains within the limits of the licensing analysis. Therefore, there will be no loss of confinement or radioactive material release.

Any increase in dose rate through the bottom of the cask due to crud accumulation is inconsequential for several reasons. The total amount of source in the cask is not increased; it is simply relocated by the distance between where the crud particle was located on the fuel assembly and the bottom of the MPC. Any minimal dose increase at the bottom of the cask is inconsequential while the cask is on an ISFSI pad because the bottom of the cask (being flush against the pad surface) is not a source of exposure during storage operations. During vertical handling operations, the overpack and transfer cask are lifted only to those heights necessary to facilitate required cask movements. These heights are typically low enough to physically prevent personnel access. Administrative controls related to prudent, heavy-load movement will preclude personnel from access underneath the lifted cask inside the FHB/AB.

8.2.14 100 PERCENT FUEL ROD RUPTURE

This accident event postulates that all of the fuel rods in a sealed MPC rupture and that fission-product gases and fill gas are released from the fuel rods into the MPC cavity.

8.2.14.1 Cause of Accident

Through all credible accident conditions, the HI-STORM 100 System maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel-cladding temperature below the short-term temperature limits, thereby ensuring fuel-cladding integrity. Although rupture of all the fuel rods is assumed, there is no credible cause for 100 percent fuel rod rupture. This accident is postulated to evaluate the MPC confinement boundary for the maximum possible internal pressure based on the non-mechanistic failure of 100 percent of the fuel rods.

8.2.14.2 Accident Analysis

The 100 percent fuel-rod-rupture accident has no thermal, criticality, or shielding consequences. The event does not change the reactivity of the stored fuel, the magnitude of the radiation source, which is being shielded, the shielding capacity, or the criticality control features of the HI-STORM 100 System. It only has the potential for affecting the internal pressure of the MPC and the leakage from the MPC. The determination of the maximum accident pressure due to a hypothetical 100 percent fuel rod rupture accident was evaluated for the MPC-32 as a bounding case for all MPCs that are licensed for use at the Diablo Canyon ISFSI.

The MPC-32 internal cavity pressure was calculated for the 100 percent rod rupture accident using the methodology from the HI-STORM 100 System generic analysis documented in Section 4.4.4 of the HI-STORM 100 System FSAR. Limiting input values were assumed for initial fuel rod fill pressure (715 psia), fuel burnup (70,000 MWD/MTU), decay heat load (28.74 kW) and minimum MPC cavity volume. The presence of nonfuel hardware and the release of fission gases from the BPRAs was also accounted for. These assumptions bound the characteristics for fuel to be loaded in any MPC to be deployed at the Diablo Canyon ISFSI. The computed MPC internal pressure from the 100 percent rod rupture accident is 183.5 psig (Reference 67, Table B.5.9), which is less than the MPC accident design pressure of 200 psig (Reference 12, Table 2.0.2).

8.2.14.3 Accident Dose Calculations

There is no effect on the shielding performance or criticality control features of the system as a result of this event. There is no effect on the confinement function of the MPC as a result of this event. All stresses remain within allowable values, ensuring confinement boundary integrity. Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

The MPC confinement boundary maintains its integrity for this postulated event. There is no effect on the shielding effectiveness, and the magnitude of the radiation source is unchanged. However, the radiation source could redistribute within the sealed MPC cavity causing a slight change in the radiation dose rates at certain locations. In that case though, the radiation dose at the ISFSI site boundary would not be affected. There is no release of radioactive material or significant increase in radiation dose rates.

8.2.15 100 PERCENT BLOCKAGE OF AIR INLET DUCTS

This accident postulates the complete blockage of all four inlet air ducts of the overpack. Blockage of the inlet air ducts is equivalent to the condition where all four outlet air ducts are blocked because either scenario stops air flow through the overpack. While a small amount of warmed air may exit the outlet air ducts and be replaced with cooler ambient air, this mechanism is of second order compared with the heat redistribution effect of the buoyancy-driven, natural-convection circulation that is established in the annular space between the MPC and overpack. As the dominant natural convection circulation is identical for either the inlet or outlet air ducts blockage, the following evaluation is applicable to both conditions. The loss of the small, second-order, air-exchange effect should the top ducts be blocked would be a lesser magnitude than the inherent conservatisms in the analysis resulting from the assumptions of complete blockage, maximum decay heat load, high ambient temperature, conservative conductivity modeling, and conservative solar heat. The complete blockage of air inlet ducts is classified as Design Event IV as defined by ANSI/ANS-57.9.

8.2.15.1 Cause of 100 Percent Blockage of Air Inlet Ducts

In Section 11.2.13 of the HI-STORM 100 System FSAR the 100 percent blockage of all overpack air inlet ducts is postulated to occur due to an environmental event such as flooding, snowfall, tornado debris, or volcanic activity. Of these, only blockage by tornado debris is credible at the Diablo Canyon ISFSI (Chapter 2). The slope protection of the hill adjacent to the storage pads (Section 4.2.1.1.9) precludes a landslide that completely covers all air inlet ducts. Should a slide occur, minor amounts of material could be removed before excessive heatup would occur. There is no credible, design-basis event at the Diablo Canyon ISFSI that could completely block all four air inlet ducts for an extended period of time where corrective action could not be taken in a timely manner to remove the blockage.

8.2.15.2 Analysis of 100 Percent Blockage of Air Inlet Ducts

The immediate consequence of a complete blockage of the air inlet ducts is that the normal circulation of air for cooling the MPC is stopped. An amount of heat will continue to be removed by localized air circulation patterns in the overpack annulus and outlet ducts, and the MPC will continue to radiate heat to the relatively cooler storage overpack. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Under this condition, the temperatures of the overpack, the MPC, and the stored fuel assemblies will rise as a function of time.
As a result of the large mass, and correspondingly large thermal capacity, of the storage overpack (in excess of 170,000 lb), it is expected that a significant temperature rise is only possible if the completely blocked condition is allowed to persist for a number of days. This accident condition is, however, a short-duration event that will be identified

and corrected through the performance of daily surveillance inspections required by the Diablo Canyon ISFSI TS.

There is a large thermal margin between the maximum-calculated, fuel-cladding temperature with design-basis fuel decay heat (HI-STORM 100 System FSAR Tables 4.4.9, 4.4.26, and 4.4.27) and the short-term, fuel-cladding-temperature limit (1,058°F), to accommodate this transient, short-term, fuel-cladding temperature excursion. The fuel stored in a HI-STORM 100 System can heat up by over 300°F before the short-term temperature limit is reached. The concrete in the overpack has a smaller, but nevertheless significant, margin between its calculated, maximum, long-term-temperature and its short-term-temperature limit, with which to withstand the temperature rise caused by this accident.

A detailed discussion of the analysis of this accident is provided in Section 11.2.13.2 of the HI-STORM 100 System FSAR. This accident has been generically analyzed both with and without considering the effect of the thermosiphon convection heat transfer phenomenon inside the MPC. Since the limiting decay heats, burnups, and cooling times for the DCPP spent fuel authorized for loading into the HI-STORM 100 System are based on credit for thermosiphon convection in the MPC; the convection-based analysis is applicable to the Diablo Canyon ISFSI.

The results of the analysis without thermosiphon bound the Diablo Canyon ISFSI design-basis analysis with thermosiphon and show that the concrete section average (that is, through-thickness) temperature remains below its short-term-temperature limit for the 72-hour duration of the accident. Both the fuel-cladding and the MPC-confinement boundary temperatures remain below their respective short-term-temperature limits at 72 hours, the fuel cladding by over 150°F, and the confinement boundary by almost 175°F. Table 11.2.9 of the HI-STORM 100 System FSAR summarizes the temperatures at several points in the HI-STORM 100 System at 33 hours and 72 hours after complete, inlet-air-duct blockage.

The thermosiphon effect is credited in the determination of the maximum allowable fuel heat emission rates (via maximum burnup, maximum decay heat, minimum cooling time limits) in Section 10.2 and in the Diablo Canyon ISFSI TS. Incorporation of the MPC thermosiphon internal convection phenomenon, as described in Chapter 4 of the HI-STORM 100 System FSAR enables the maximum, design-basis, PWR-decay-heat load to rise to about 37 kW. The thermosiphon effect also shifts the highest temperatures in the MPC enclosure vessel toward the top of the MPC. The peak, MPC-lid, outer-surface temperature, for example, is computed to be about 600°F in the thermosiphon-enabled solution compared with about 210°F in the thermosiphon-suppressed solution, with both solutions computing approximately the same peak cladding temperature. In the 100 percent, inlet-duct-blockage condition, the heated MPC lid and MPC shell become effective heat dissipaters because of their proximity to the overpack outlet ducts and because the thermal radiation heat transfer rises at the fourth power of absolute temperature. As a result of this increased heat rejection from the upper region of the MPC, the time limits for reaching the short-term peak fuel-cladding temperature limits calculated without thermosiphon (72 hours) remains bounding.

Under the complete, air-inlet-duct-blockage condition, it must also be demonstrated that the MPC internal pressure does not exceed its design-basis accident limit. The bounding MPC internal pressure was calculated at an ambient temperature of 65°F, design-basis insolation, and maximum decay heat as part of the site specific thermal analysis (Reference 63). The analysis did not assume a simultaneous 100% rod rupture event, since the peak fuel cladding temperatures for the accident conditions never exceed the regulatory accident temperature limit, which ensures no significant cladding failures would occur. This is consistent with the latest NRC guidance on fuel cladding in dry storage casks (Reference 21), which states, “In order to assure integrity of the cladding material . . . For off-normal and accident conditions, the maximum cladding temperature should not exceed 570°C (1058°F).” The same result is confirmed for all accidents evaluated for the Diablo Canyon ISFSI. Therefore, no coincident 100% rod rupture postulations with an accident are evaluated. This is supported by the HI-STORM 100 CoC, Amendment 5 (Reference 64). The resultant MPC internal pressure is calculated to be 90.5 psig (Reference 67, Table B.5.10), which is less than the accident design pressure of 200 psig (HI-STORM 100 System FSAR Table 2.2.1).

8.2.15.3 Dose Calculations for 100 Percent Blockage of Air Inlet Ducts

As shown in the analysis of the 100 percent blockage of air inlets accident in the HI-STORM 100 System FSAR, the shielding capabilities of the HI-STORM 100 System are unchanged because the section average concrete temperature does not exceed its short-term-condition design temperature limit for the duration of the accident. The Diablo Canyon ISFSI TS require the blockage to be cleared within 8 hours of declaring the heat removal system inoperable. Assuming the blockage occurs just after the last 24-hour surveillance is performed, the 8-hour completion time provides a total of 32 hours in this condition, which is the analyzed duration of the event. The concrete, fuel cladding and MPC shell do not reach their short-term-temperature limits over the entire analyzed 32-hour duration of the event. In addition, the emergency procedures will require an inspection of the ISFSI following a tornado, which will shorten the time to complete clearing the blockage. The elevated temperatures will not cause a breach of the confinement system and the short-term, fuel-cladding-temperature limit is not exceeded. Therefore, there are no direct or airborne radiation consequences of this accident.

For complete blockage of air inlet ducts it is estimated that the removal, cleaning, and replacement of the affected perforated plates (sheets) will take two people approximately 2 hours. The radiation doses to workers who remove debris blocking the inlet ducts are estimated to be double those conservatively estimated for the analysis of the partial inlet blockage in Section 8.1.4. The dose rate at this location is estimated to be 58 mrem/hour. The total exposure for two people taking 2 hours to perform these corrective actions is 0.232 man-rem.

8.2.16 TRANSMISSION TOWER COLLAPSE

Two 500-kV transmission towers are located in the vicinity of the ISFSI storage pads and CTF. This section addresses the impact of a fallen transmission tower on a loaded overpack. During transportation to the CTF and all handling and lifting activities at the CTF, a loaded transfer cask is protected from the impact of a falling transmission tower at all times by the structure of the cask transporter. Therefore, an analysis of the transfer cask for tower collapse impact loads is not required and has not been performed. A postulated transmission tower collapse at both the ISFSI storage site and CTF was analyzed (Reference 45) to demonstrate that there is no loss of confinement from damage to an MPC during both transfer operations or while stored at the ISFSI pad in an overpack. The collapse of a transmission tower is classified as Design Event IV, as defined by ANSI/ANS-57.9.

8.2.16.1 Cause of Transmission Tower Collapse

The transmission tower collapse is postulated as a consequence of extreme wind speeds (above 84 mph) creating greater than design loads on the tower structure.

8.2.16.2 Analysis of the Transmission Tower Collapse

The location of the transmission towers with respect to the CTF and ISFSI storage pads is shown in Figure 2.1-2. A transmission tower is postulated to collapse by hinging of the legs and failure of braces without incident of leg or pile foundation pullout or lateral failure due to wind- or tornado-wind-generated loads. The transmission tower is a four-legged structure with a “T” shape at the top. Based on the location of the transmission corridor with respect to the CTF and the ISFSI storage pad and the conduct of loading operations, in the unlikely event of a collapse, a tower could impact the loaded overpack in different orientations at the CTF and the storage pad. At the CTF, the tower collapse is modeled with the pointed section of the “T” cross-bar impacting the MPC lid directly because the overpack may not have its top lid installed at the time of the event. At the ISFSI, the flat side of the “T” cross-bar impacts the overpack top lid.

A commercial computer code developed by the Livermore Software Technology Corporation and QA validated by Holtec International, LS-DYNA (Reference 26), was used to numerically model the problem and develop the impact forces of the tower structure on the target. LS-DYNA is a general purpose, explicit finite element program used to analyze the nonlinear dynamic response of two- and three-dimensional inelastic structures.

There are two towers that are close enough in proximity to the CTF and ISFSI storage site to impact a cask if a tower collapse were to occur. The applicable physical characteristics for the two transmission towers are:

(1) One tower has a height of approximately 125 ft, measured from the ground to the highest point. It is located, at its nearest foundation, approximately 100 ft west of the ISFSI pads and 60 ft south of the CTF. It has a total structural weight of approximately 25 kips.

(2) The other tower has a height of approximately 135 ft, measured from the ground to the highest point. It is located, at its nearest foundation, approximately 60 ft east of the ISFSI pads. It has a total structural weight of approximately 31 kips.

The analysis evaluates the impact forces generated by collapse of the second tower as the governing case since it is a taller and heavier tower.

8.2.16.2.1 Tower Collapse at the CTF

The LS-DYNA computer simulation of the tower collapse at the CTF models the pointed portion of the “T” bar impacting the MPC lid. The force of the tower impact on the MPC lid is 427 kips. This force is much smaller than the allowable impact force for the weld (2,789 kips) determined in the tornado-missile analysis, and thus will not cause a breach of the MPC confinement boundary. The maximum local stress of the MPC lid due to the impact is 14.6 ksi, which is smaller than the yield stress of the lid material (18.8 ksi). The potential for MPC-lid puncture due to this event is bounded by the intermediate-missile evaluation described in Section 8.2.2. The design-basis intermediate missile (a 760-lb insulator string traveling at 157 mph) is shown not to penetrate the 9-1/2-inch-thick MPC lid.

8.2.16.2.2 Tower Collapse at the ISFSI Storage Pad

The LS-DYNA computer simulation of the tower collapse at the ISFSI storage pad models the flat side of the “T” bar impacting the overpack top lid. The unfiltered impact force was computed to be 534 kips. To convert this to an equivalent g-load on the overpack, the 534 kips is divided by the weight of the loaded overpack:

$$534 / 360 = 1.48 \text{ g}$$

The overpack structure is designed to withstand a 45-g deceleration. Therefore, the impact of the force due to the transmission tower collapse is bounded with margin. The horizontal component of the impact force is less than 93 kips, which is bounded by the large tornado missile load of 122 kips described in Section 8.2.2. The overturning moments are also bounded for the effects on the anchorage to the ISFSI pad. MPC confinement boundary integrity related to tower impact discussed in Section 8.2.16.2.1 is applicable at the pad.

8.2.16.3 Dose Calculation for Transmission Tower Collapse

There are no offsite dose consequences as a result of this accident because the MPC confinement boundary remains intact. Potential damage to the overpack structure as a result of this event will vary based on the actual location and severity of the impact on the overpack. Based on the loads described above, no significant damage to the shielding effectiveness of the overpack is expected. If necessary, corrective actions will be implemented based on the nature of the damage in a time frame commensurate with safety significance.

8.2.17 SUPPLEMENTAL COOLING SYSTEM (SCS) FAILURE

The SCS system is a supplied fluid device used to provide supplemental HI-TRAC cooling during the loading operation of high burnup fuel while utilizing temporary shielding on the transfer cask, and unloading operation of any MPC loaded under Amendment 2 of this license. The SCS system maintains water in the MPC/HI-TRAC annulus to cool the MPC shell in order to maintain the fuel cladding below the ISG-11 Rev. 3 temperature limit. Although an SCS System failure is highly unlikely, for defense-in-depth an accident condition that renders it inoperable for an extended duration is postulated herein.

8.2.17.1 Cause of SCS Failure

Because the SCS is a keep full system, the only failure mode is a complete loss of annulus water from an uncontrolled leak or line break, and SCS cannot be reestablished within the required restoration time because of equipment configuration.

8.2.17.2 Analysis of Effects and Consequences of SCS Failure

In the event of an SCS failure, a rapid water loss occurs and annulus water is replaced with air. For the condition of a vertically oriented HI-TRAC with air in the annulus, the maximum steady state temperatures are below the accident temperature limits for fuel cladding and components (Reference 63).

Because none of the temperature or pressure limits are exceeded, shielding, criticality and confinement functions are unaffected. Because there is no adverse effect on the shielding or confinement functions, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, the SCS failure does not affect the safe operation of the HI-STORM 100 System.

8.2.17.3 SCS Failure Dose Calculations

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

8.2.17.4 SCS Failure Corrective Action

In the vertical orientation the HI-TRAC is designed to withstand an SCS failure without an adverse effect on its safety functions. However, actions will be taken to either restore supplemental cooling or transfer the MPC into the HI-STORM in order to return the high burnup fuel cladding temperatures to below ISG-11 Rev. 3 limits.

8.2.18 REFERENCES

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8.3 SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

The Diablo Canyon ISFSI storage site is located as shown in Figure 2.1-2. The protected area for the storage site is shown in Figure 4.1-1 and described in Section 4.1. The nearest road to the ISFSI site is a DCPP access road that is used to access various onsite facilities. Use of this road is controlled by PG&E. This access road will be relocated to the north side of the raw water storage reservoir. As concluded in Section 2.2, there are no credible accident scenarios involving any offsite industrial, transportation, or military facilities in the area around the DCPP site that will have any significant adverse impact on the ISFSI. In addition, there are no potential onsite fires, explosions, or chemical hazards that would have a significant or unacceptable impact on the ISFSI. Site characteristics that affect the safety analysis, and how they have been considered in developing suitable margins of safety for the storage of DCPP's spent fuel, are summarized in Table 8.3-1.

TABLE 8.1-1

OFF-NORMAL OPERATION ANNUAL DOSES AT THE SITE BOUNDARY AND FOR THE NEAREST RESIDENT DUE TO EFFLUENT RELEASE FROM A SINGLE HI-STORM CASK[d]

	Site Boundary Dose[a] (mrem)	Nearest Resident Dose[b] (mrem)
Whole body ADE[c]	1.27E-03	5.33E-03
Thyroid ADE	1.02E-04	4.31E-04
Critical Organ ADE (Max)	9.31E-03	3.92E-02

Notes:

(a) Occupancy at the site boundary is assumed to be 2,080 hrs/yr.

(b) Occupancy for the nearest resident is assumed to be 8,760 hrs/yr. Also, the site boundary χ/Q is used for the nearest resident; this is conservative because the nearest resident is located farther away from the release point than the site boundary.

(c) ADE is annual dose equivalent.

(d) These values are taken from Table 9-1 of Holtec Report HI-2002513, Revision 7.

TABLE 8.2-1

MODAL DAMPING VALUES FOR VARIOUS STRUCTURES

Type of Structure	Percent of Critical Damping			
	DE	DDE	HE	LTSP
Cask/Module Assembly - Mechanical Components	2	2	4	5
Welded Steel Assemblies	1	1	4	5
Bolted or Riveted Steel Assemblies	2	2	7	5
Reinforced Concrete Structures Above Ground	5	5	7	5

TABLE 8.2-2

DIABLO CANYON CASK TRANSPORTER GEOMETRY AND WEIGHT

Item	Value
Minimum Length of Tracks	294 inches
Length of Track in Contact with the Ground	197 inches
Width of Tracks	29.5 inches
Inner Distance Between Tracks	160.5 inches
Maximum Height of Center-of-Gravity Above Ground	107.53 inches
Minimum Height of Center-of-Gravity Above Ground	78 inches
Distance Between Center-of-Gravity and Rear of Tracks	115 inches
Distance Between Tower Centerline and Rear of Tracks	136.6 inches
Weight of Cask Transporter Without Payload	190,000 lb

TABLE 8.2-3

GENERIC CASK TRANSPORTER GEOMETRY AND PROPERTIES USED IN SEISMIC SIMULATIONS

Item	Value
Length of Tracks	234 inches
Width of Tracks	30 inches
Outer Dimension Between Tracks	152 inches
Height of Center-of-Gravity Above Ground	77.4 inches
Distance Between Center-of-Gravity and Rear of Tracks	122 inches
Distance Between Tower Centerline and Rear of Tracks	118 inches
Weight of Cask Transporter Without Payload	170,000 lb

TABLE 8.2-4

TRANSFER CASK AND OVERPACK INPUT DATA FOR CASK TRANSPORTER SEISMIC ANALYSIS

Transport Mode	Lifted Weight (With Loaded MPC) (lb)	Grade (percent)	Height Above Roadway In Transit (at lowest point of cask) (inches)	Center of Gravity Height (above lowest point on cask) (inches)
Vertical HI-STORM Overpack	360,000	5	10	118.5
Vertical HI-TRAC Transfer Cask	260,000	5	42	95
Horizontal HI-TRAC Transfer Cask[a]	260,000	8.5	6	65

[a] The results from the simulations with the transfer cask in the horizontal orientation remain valid for the transfer cask in the vertical orientation (Holtec Report HI-2012768, Revision 3).

TABLE 8.2-5

CASK TRANSPORTER DYNAMIC SIMULATIONS FOR STABILITY EVALUATION

Phase	Transporter Configuration	Grade and Friction Factors	Time Histories
1	Transporter with horizontal HI-TRAC rigidly connected to Transporter	Flat surface 0.4 friction factor	5 time history sets. Time history sets are designated as Sets 1, 2a, 3, 5, and 6
2	Transporter with horizontal HI-TRAC rigidly connected to Transporter	6% grade 0.4 friction factor	Sets 1P, 5N, 6N, 6P
3	Transporter with horizontal HI-TRAC rigidly connected to Transporter	8.5% grade 0.4 friction factor	Set 5N
4	Transporter with vertical HI-STORM rigidly connected to Transporter	6% grade 0.4 friction factor	Sets 5N and 6N
5	Transporter with vertical HI-STORM rigidly connected to Transporter	Flat surface 0.8 friction factor	Set 6

Note 1: For all simulations in Phase 1, and for Phase 5, the longitudinal axis of the transporter is aligned with the Fault Parallel time history. For simulations in Phases 2-4, the designator of N or P means that the component (N for Fault Normal and P for Fault Parallel) is aligned down-slope.

Note 2: The HI-TRAC will only be transported in a vertical orientation; nevertheless, the results from the simulations with the HI-TRAC in a horizontal orientation are retained as the conclusions from these simulations remain valid (Holtec Report HI-2012768, Revision 3).

TABLE 8.2-6

MAXIMUM CASK TRANSPORTER HORIZONTAL EXCURSION
DURING A SEISMIC EVENT

Simulation Phase No.	Mode Of Operation	Max. Horizontal Excursion (inches)		
		Bounding Seismic Time History Set	Transverse	Longitudinal
1	HI-TRAC in Horizontal Orientation[a]	Saratoga	8.90	8.9
2		El Centro (Longitudinal) Saratoga (Transverse)	10.7	21.5
3		El Centro	4.6	30.2
4	HI-STORM in Vertical Orientation	El Centro (Longitudinal) Saratoga (Transverse)	10.6	21.3
5		Saratoga	0.43	0.24

[a] The results from the simulations with the transfer cask in the horizontal orientation remain valid for the transfer cask in the vertical orientation (Holtec Report HI-2012768, Revision 3).

TABLE 8.2-7

KEY INPUT DATA USED FOR CTF SEISMIC/STRUCTURAL ANALYSIS

Parameter	Value
HI-STORM 100SA Overpack Weight (empty)	270,000 lb
Loaded MPC Weight	90,000 lb
HI-TRAC Transfer Cask Weight (empty)	142,000 lb
HI-STORM Mating Device Weight	20,000 lb
HI-STORM 100SA Overpack Height	217 inches
HI-STORM 100SA Overpack Outer Diameter	146-1/4 inches
HI-STORM 100SA Overpack Center-of-Gravity Height	118.5 inches
HI-TRAC Transfer Cask Height	192-1/2 inches
HI-TRAC Transfer Cask Outer Diameter	93 inches
HI-TRAC Transfer Cask Center-of-Gravity Height	95 inches
HI-STORM Mating Device Height (excluding lift lugs and alignment ring)	9.563 inches
HI-STORM Mating Device (spacer ring) Length/Width	143/117 inches
HI-TRAC-to-Mating Device Bolt Geometry	2-4 1/2 UNC
HI-STORM-to-Mating Device Bolt Geometry	3 1/4-4 UNC & 2-6 UNC
Structural Steel Material	SA-516-Gr. 70
Bolt Material	SA 193-B7

TABLE 8.2-8

GROUND SPECTRAL ACCELERATIONS

	Seismic Coefficient		
Earthquake	Horizontal #1	Horizontal #2	Vertical (See Note)
DE	0.225	0	0.1335
LTSP	1.12	1.12	0.725

Note: The vertical accelerations are the ZPA values as the stacked unit vertical frequency is 65.7 Hz. The horizontal spectral accelerations correspond to a horizontal frequency of 19.85 Hz.

TABLE 8.2-9

ISFSI STORAGE PAD SEISMIC ANALYSIS RESULTS - INTERFACE LOADS

Seismic Event at ISFSI	HE	LTSP	HE[(a)]	LTSP[(a)]
Maximum/Minimum Interface Compression Force (kip)[(b)]	674.2/127.6	684.1/105.8	773.3/130.6	632.0/55.6
Maximum Interface Shear Force Along X-Axis[(c)] (kip)	509.4	432.0	379.9	325.8
Maximum Interface Shear Force Along Y-Axis[(c)] (kip)	460.5	355.5	426.1	364.6
Maximum Net Interface Shear Force (kip)	515.0	440.0	428.0	390.0
Maximum Interface Moment About X-Axis at Interface (kip-in.)	54,564	42,139.2	50,498	43,209
Maximum Interface Moment About Y-Axis at Interface (kip-in.)	60,369	51,197.2	45,017	38,603
Maximum Interface Moment (kip-in.)	61,000	52,000	50,500	46,000
Effective COF at Cask/Embedment Interface	0.18	0.154	0.150	0.132
Maximum Tensile Load in Embedment Anchor Rods (kip)	62.13	48.85	49.73	42.34

Notes:
(a) These simulations have the vertical excitation reversed in direction over the total event time.
(b) Includes dead load = 360,000 lb.
(c) Base maximum shear forces are computed by dividing the appropriate maximum moment by the height to the centroid (118.5 inch). Y-Shear corresponds to MX, X-Shear corresponds to MY.

The moments and forces reported above act at the lower surface of the embed plate. The X, Y, Z-axes are located at a point on the cask longitudinal centerline (extended to the bottom surface of the embed plate). The X, Y directions correspond to the East-West and North-South directions, respectively, and the Z-axis is vertically upward.

TABLE 8.2-10

SUMMARY OF RESULTS FOR CASK ANCHORAGE (Flange, Shell, Gussets, and Cask Anchor Studs) FROM QUASI-STATIC STRENGTH EVALUATION

Item	Calculated Value	Allowable Value	Safety Factor[a]
Maximum Primary Membrane + Bending Stress away from Loaded Region and Discontinuity (ksi) – Case 1 - Preload	10.35	26.3	2.54
Maximum Primary Membrane + Bending Stress Intensity away from Loaded Region and Discontinuity (ksi) – Case 2 - Preload + Seismic	40.92	62.3	1.52
Maximum Weld Shear Stress (ksi)	26.34	29.4	1.116

[a] Allowable Value/Calculated Value

TABLE 8.2-12

CONFINEMENT BOUNDARY LEAKAGE DOSES AT THE SITE BOUNDARY

Dose Category	30-Day Dose (rem)	10 CFR 72.106 Limit (rem)
TEDE	8.3E-04	5
TODE=DDE+CDE (Max)	6.36E-03	50
LDE	2.2E-05	15
SDE	2.6E-05	50

TEDE: total effective dose equivalent
TODE: total organ dose equivalent
DDE: deep dose equivalent
CDE: committed dose equivalent
LDE: lens dose equivalent
SDE: shallow dose equivalent

TABLE 8.2-13

EVALUATION RESULTS DUE TO AN
ATMOSPHERIC LIGHTNING STRIKE ONTO A CASK

Cask Type	Resistance Heat Generated (watt-seconds)	Outer Shell Temperature Rise (°F)
HI-STORM Storage Cask	6,317	0.22
HI-TRAC Transfer Cask	7,489	0.44

TABLE 8.2-14

EVALUATION RESULTS DUE TO A
TRANSMISSION LINE DROP ONTO A CASK

	HI-STORM Storage Cask	HI-TRAC Transfer Cask
Period 1 Total Energy	13,132 watt-hr	
Period 2 Energy	258 watt-hr	
Period 3 Energy	1,101 watt-hr	
Weight of Material Sublimated in Period 1	1.46 lb	
Diameter of Sublimated Hole in Affected Component	2.563 in.	3.625 in.
Affected Component Temperature Rise	9.3°F	32.64°F

TABLE 8.3-1

SUMMARY OF SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

Site Characteristic	Effect on ISFSI Safety Analysis
Severe environmental conditions in summer and winter	Thermal analyses of the effects of abnormally high ambient temperatures on the storage system considered climatic conditions of the area. Design temperatures were selected to bound day/night average maximum temperatures that could occur over a period of several days. (Sections 3.2 and 8.2.10)
Tornado winds and missiles	Regional meteorology and plant conditions were considered in the determination of the design basis tornado maximum wind and missile parameters. (Sections 3.2 and 8.2.2)
Earthquakes	Regional and site geology and seismology were used to define the design basis ground motion. (Sections 3.2 and 8.2.1)
Explosions	Site-specific conditions were evaluated and bounded by the cask design. Administrative controls are used to limit the risk. (Sections 2.2, 3.3, and 8.2.6)
Fires	The evaluation of fire potential was based on the site characteristics and equipment, as well as the systems that are used to transfer canisters and storage casks. (Sections 2.2, 3.3, and 8.2.5)
Lightning	Evaluation determined cask design acceptable. (Sections 3.2 and 8.2.8)
Transmission line strike	Evaluation determined storage cask and transfer cask design acceptable. (Sections 2.2 and 8.2.8)
Transmission tower collapse	Evaluation determined cask design acceptable. (Sections 2.2 and 8.2.16)
Flooding	ISFSI pad and CTF evaluated and determined to be acceptable. (Sections 3.2 and 8.2.3)
Slope Stability	The stability of the slopes adjacent to the storage pad, CTF, and transport route have been evaluated and protective measures taken as appropriate. (Sections 2.6.5 and 4.2.1.1.9)
Site location	ISFSI site is remote, with less than 20 individuals residing within 5 miles of the site. (Section 2.1)

CHAPTER 9

CONDUCT OF OPERATIONS

CONTENTS

CHAPTER 9

CONDUCT OF OPERATIONS

FIGURES

CHAPTER 9

CONDUCT OF OPERATIONS

This chapter discusses the PG&E organization for the design, fabrication, construction, testing, operation, modification, and decommissioning of the Diablo Canyon ISFSI. Included are descriptions of organizational structure, personnel responsibilities and qualifications, and PG&E interface with contractors and other outside organizations.

Programs under 10 CFR 50 for DCPP, such as radiation protection, environmental monitoring, emergency preparedness, quality assurance, and training will be adopted as necessary to ensure the safe operation and maintenance of the Diablo Canyon ISFSI under 10 CFR 72. The following are the plans that support the conduct of ISFSI operations: an Appendix to the DCPP Physical Security Plan, a Safeguards Contingency Plan, a Security Training and Qualification Plan, an Emergency Plan, a Quality Assurance (QA) Program, and a Training Program.

As appropriate, 10 CFR 50 license requirements will be removed from ISFSI procedures upon termination of the 10 CFR 50 licenses. During this transition period, appropriate 10 CFR 72.48 reviews will be conducted to ensure continued compliance with ISFSI 10 CFR 72 license requirements. This process will result in stand-alone ISFSI programs that implement the 10 CFR 72 license. PG&E will maintain the appropriate administrative and managerial controls at the ISFSI until the Department of Energy (DOE) takes title to and assumes responsibility for the spent fuel.

9.1 ORGANIZATIONAL STRUCTURE

9.1.1 CORPORATE ORGANIZATION

The organization chart shown in Figure 9.1-1 represents the organization during the preoperations phase (historical) and Figure 9.1-2 represents the current organizational relationships for the ISFSI. Relationships between corporate personnel and Diablo Canyon ISFSI onsite personnel are depicted in the figures. While DCPP units are operating, the costs for construction and operation of the Diablo Canyon ISFSI will be funded from revenues generated from operation of the units. Upon shutdown of the operating units, the costs for construction, operation, and decommissioning of the Diablo Canyon ISFSI will be funded from the DCPP Decommissioning Trust, which has been approved by the California Public Utilities Commission (CPUC). All costs are monitored and controlled by the ISFSI Project Manager during the ISFSI preoperations phase, and by the Dry Fuel Management Program Manager during the ISFSI operations phase.

Following decommissioning of both operating units and termination of the 10 CFR 50 operating licenses, the Diablo Canyon ISFSI organization will change. The revised ISFSI organization will be dependent on the new PG&E organization that will result following the decommissioning of the operating units. PG&E will notify the NRC of the new Diablo Canyon ISFSI organization at that time. (The operating licenses for DCPP Units 1 and 2 expire in 2024 and 2025, respectively.)

9.1.2 CORPORATE FUNCTIONS, RESPONSIBILITIES, AND AUTHORITIES

The Senior Vice President, Generation and Chief Nuclear Officer is the corporate executive responsible for overall ISFSI safety and is responsible for taking measures needed to ensure acceptable performance of the staff in designing, fabricating, constructing, testing, operating, modifying, decommissioning, and providing technical support to the ISFSI. In addition, the Senior Vice President, Generation and Chief Nuclear Officer is responsible for providing engineering and design services, geotechnical services (through the Vice President, Generation Business and Technical Services), and learning services for the ISFSI. The Senior Vice President, Generation and Chief Nuclear Officer is also responsible for ISFSI operations, safety, and emergency services. This position is the corporate interface with the CPUC for all ISFSI cost matters. The Senior Vice President, Generation and Chief Nuclear Officer, reports to the Executive Vice President, Operations and Chief Operating Officer.

The Vice President, Generation Business and Technical Services is responsible for providing safety assessments for the ISFSI and reports to the Senior Vice President, Generation and Chief Nuclear Officer.

The Nuclear Safety Oversight Committee (NSOC) reports to the Senior Vice President, Generation and Chief Nuclear Officer, and implements the Independent Review Program. The Independent Review functions, meeting requirements, and responsibilities are described in Sections 17.1 and 17.2 of the DCPP Final Safety Analysis Report (FSAR) Update (Reference 1).

The Diablo Canyon ISFSI is operated under the same management organization responsible for the operation of DCPP. Throughout the ISFSI lifetime, legal support is available from PG&E corporate headquarters; technical and operational support is available from DCPP personnel and outside consultants. This support will be provided, when needed, for licensing, QA, engineering, radiation protection, maintenance, testing, emergency planning, security, and decommissioning.

As shown in Figures 9.1-1 (historical) and 9.1-2, the QA and quality control functions are performed by personnel independent of the ISFSI line organization. During the preoperational phase, the results of QA audits and recommendations for improvement were provided directly to the ISFSI Project Manager, the Dry Fuel Management Program Manager; the Senior Director, and the Senior Vice President, Generation and Chief Nuclear Officer. During the operations phase, the results of QA audits and recommendations for improvement will be provided directly to the Nuclear Fuels Manager (or equivalent) and all senior leadership positions up to and including the Senior Vice President, Generation and Chief Nuclear Officer. The frequency and scope of QA audits is described in Section 17.18 of the DCPP FSAR Update.

9.1.3 IN-HOUSE ORGANIZATION

The Diablo Canyon ISFSI is designed, constructed, tested, and operated under the same organization responsible for the design, testing, and operation of the DCPP. The only difference is that during the Diablo Canyon ISFSI preoperations phase, the ISFSI Project Manager was responsible for day-to-day management of ISFSI activities; whereas during the Diablo Canyon ISFSI operations phase, the Station Director is responsible for the day-to-day management of the ISFSI.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED.
Figure 9.1-1 shows the organization during the ISFSI preoperations phase, including design, fabrication, construction, fuel loading, testing, and initial operation of the first cask. During the preoperations phase, the Diablo Canyon ISFSI Project Manager was responsible for day-to-day management of ISFSI activities and ensuring that the design, fabrication, construction, fuel loading, testing, and initial operation of the first cask are safely conducted. Cost control for all of these activities was the responsibility of the Diablo Canyon ISFSI Project Manager. The Diablo Canyon ISFSI Project Manager reported to the Director, Strategic Projects. The Director, Strategic Projects, reported to the Senior Director, Engineering. The Senior Director, Engineering reported to the Site Vice President. The Site Vice President reported to the Senior Vice President, Energy Supply and Chief Nuclear Officer. The Senior Vice President - Energy Supply and Chief Nuclear Officer, was responsible for overall safety of ISFSI activities, and the industrial safety program, during the ISFSI preoperations phase.

Figure 9.1-2 shows the organization during the ISFSI operations phase, including design, fabrication, construction, fuel loading, and testing of all casks subsequent to the initial cask. During ISFSI operations, the Senior Vice President, Generation and Chief Nuclear Officer is responsible for emergency services and the overall safety of ISFSI activities, including fuel loading, testing, maintenance, and operation of all subsequent casks. The Senior Vice President, Generation and Chief Nuclear Officer reports directly to the Executive Vice President, Operations and Chief Operating Officer.

The Senior Director, Station Director, is responsible for administering, coordinating, planning, and scheduling all ISFSI operating activities. This position is responsible for ensuring that appropriate operating procedures are available and that operating personnel are familiar with the procedures. The Director, Operations Services exercises direct supervision over ISFSI operational conditions and Technical Specification activities. The Director, Nuclear Maintenance Services exercises direct supervision over ISFSI maintenance and ISFSI work planning. The Senior Director, Station Director has appointed the Director, Nuclear Decommissioning to manage the non-day-to-day operations of the ISFSI under the Dry Fuel Management Program.

The Director, Nuclear Decommissioning is responsible for technical and project management services for the ISFSI including the management of the Dry Fuel Management Program. The Director, Nuclear Decommissioning reports to the Vice President, Generation Business and Technical Services.

The Director, Security and Emergency Services, is responsible for providing security for the ISFSI. The Director, Security and Emergency Services, reports to the Senior Director Engineering, Technical and Emergency Services.

The Vice President, Generation Business and Technical Services , is responsible for nuclear risk and compliance programs, and regulatory services for the ISFSI. Throughout both phases, functions such as engineering, design, construction, QA, radiation protection, testing, operations, and security will be performed by DCPP personnel. The existing DCPP Plant Staff Review Committee (PSRC) reviews matters affecting the safe storage of spent nuclear fuel. The PSRC is chaired by the Station Director, or delegate. PSRC membership, functions, meeting requirements and responsibilities are described in Sections 17.1 and 17.2 of the DCPP FSAR Update.

9.1.4 RELATIONSHIPS WITH CONTRACTORS AND SUPPLIERS

All activities associated with the ISFSI are managed and approved by PG&E. The cask vendor is responsible for providing the HI-STORM 100 System. Consulting firms may be used to support the design and engineering efforts for the ISFSI, and for the construction of associated structures and components, including the ISFSI storage pad. Qualified vendors may be selected to provide other services and/or equipment as needed.

During the preoperations phase, the Diablo Canyon ISFSI Project Manager was responsible for providing oversight of work activities performed by contractors. Fewer contractors will be involved during the ISFSI operations phase, and their activities will be managed by the Dry Fuel Management Program Manager.

9.1.5 TECHNICAL STAFF

The PG&E staff that supports DCPP Units 1 and 2 operations is described in Section 13.1 of the DCPP FSAR Update. This staff will also support the Diablo Canyon ISFSI. The functions, responsibilities and authorities of certain Diablo Canyon ISFSI personnel identified in Figure 9.1-2 are described in Sections 13.1 and 17.1 of the DCPP FSAR Update. The responsibilities of the ISFSI Project Manager during the preoperations phase, were as described in Section 9.1.3. The responsibilities of the ISFSI support staff during the operations phase, are as described in Section 9.1.6. The qualifications of the PG&E technical staff meet or exceed the requirements specified in Section 9.1.7.

The design for the ISFSI storage system is primarily performed by the cask vendor. Designs, calculations, and analyses performed by the cask vendor and any other vendors will be reviewed and approved by Diablo Canyon personnel prior to construction.

9.1.6 OPERATING ORGANIZATION, MANAGEMENT, AND ADMINISTRATIVE CONTROL SYSTEM

9.1.6.1 Onsite Organization

This section describes the ISFSI operations organization that will be in place during long-term storage of spent nuclear fuel. The ISFSI operations organization is shown in Figure 9.1-2 and is the same organization currently responsible for the operation of DCPP. Approximately 11 full-time equivalent personnel will be used from the existing DCPP organization to perform the functions of support of the Dry Fuel Management Program and security. Lines of authority, responsibility, and communication will be defined and established for all ISFSI organization positions. These relationships will be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions.

9.1.6.2 Personnel Functions, Responsibilities and Authorities

The Senior Vice President, Generation and Chief Nuclear Officer is responsible for overall ISFSI safety and is responsible for taking measures needed to ensure acceptable performance of the staff in designing, fabricating, constructing, testing, operating, modifying, decommissioning, and providing technical support to the ISFSI. The Senior Vice President, Generation and Chief Nuclear Officer provides direction for personnel training and qualifications.

The Senior Vice President, Generation and Chief Nuclear Officer reports directly to the Executive Vice President, Operations and Chief Operating Officer, and will be responsible for the safe operation of the ISFSI, maintaining personnel trained and qualified in accordance with the Diablo Canyon ISFSI operations training program, and operation of ISFSI equipment that is important to safety. The Senior Vice President, Generation and Chief Nuclear Officer provides direction for the safe operation, maintenance, radiation protection, emergency services, and security of the ISFSI and personnel.

DCPP Operations, Maintenance, and Security staff are responsible for the day-to-day operation of the ISFSI as prescribed by the Dry Fuel Management Program. They perform their activities in accordance with the requirements of the Diablo Canyon ISFSI license, TS, physical security plan, plant procedures, and applicable state and federal regulations. Security staff personnel are responsible for ISFSI site security during routine, emergency, and contingency operations.

In order to ensure continuity of operation and organizational responsiveness to off-normal situations, a formal order of succession and delegation of authority will be established. The Senior Vice President, Generation and Chief Nuclear Officer will designate in writing personnel who are qualified to act as the Senior Vice President, Generation and Chief Nuclear Officer in his absence.

9.1.6.3 Administrative Control

Planned and scheduled internal and external QA audits in accordance with the DCPP QA Program will be performed to evaluate the application and effectiveness of management controls, procedures, and other activities affecting safety. The audit program will describe audit frequency, methods for documenting and communicating audit findings, resolution of issues, and implementation of corrective actions.

The existing DCPP change control program will be revised to incorporate 10 CFR 72.48 and other ISFSI regulatory requirements. The DCPP change control program will be used to manage Diablo Canyon ISFSI change control.

9.1.7 PERSONNEL QUALIFICATION REQUIREMENTS

Each member of the DCPP staff performing work on the Diablo Canyon ISFSI will meet or exceed the qualifications of Regulatory Guide 1.8 (Reference 2), with the exceptions as noted in the DCPP FSAR Update, Table 17.1-1. In addition, the Senior Vice President, Generation and Chief Nuclear Officer and the ISFSI support and security staff are qualified as described below:

The Senior Vice President, Generation and Chief Nuclear Officer, at the time of assuming the responsibilities for ISFSI operations, shall have a minimum of 8 years of power plant experience, of which a minimum of 3 years shall be nuclear power plant experience. A maximum of 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 ISFSI support and security staff, at the time of appointment to their positions, shall have a high school diploma or successfully completed the General Education Development test. ISFSI support staff shall have 2 years of power plant experience of which a minimum of 1 year shall be nuclear power plant experience. Consistent with the assigned duties, ISFSI support staff will be trained and qualified in accordance with the Diablo Canyon ISFSI Operations Training Program training and qualification requirements. Security staff that supports the ISFSI will be trained and qualified in accordance with the DCPP Security Training and Qualifications Plan requirements.

During loading of the ISFSI, fuel handling operations will either be performed by, or supervised by, DCPP personnel trained and qualified by the Diablo Canyon ISFSI Operations Training Program. During ISFSI operations, operation of equipment and controls that are identified as important to safety for the ISFSI will be limited to personnel who are trained and qualified in accordance with the Diablo Canyon ISFSI Operations Training Program, or personnel who are under the direct visual supervision of a person who is trained and qualified in accordance with the Diablo Canyon ISFSI Operations Training Program.

9.1.8 LIAISON WITH OUTSIDE ORGANIZATIONS

All activities associated with ISFSI operations are managed and approved by PG&E. These activities will be performed in accordance with approved procedures. The cask vendor provides engineering, technical support, and other services for the ISFSI relating primarily to the design and construction of cask structures and components. Other qualified vendors may be selected to provide specialty services and/or equipment. Interface with DOE, cask vendor, and other outside organizations is performed in accordance with contractual agreements.

9.1.9 REFERENCES

1. Diablo Canyon Power Plant Units 1 & 2 Final Safety Analysis Report Update.

2. Regulatory Guide 1.8, Personnel Selection and Training, USNRC, Proposed Rev. 2. February 1979.

9.2 PREOPERATIONAL AND STARTUP TESTING

This section describes the preoperational and startup testing plans for the storage system, including necessary equipment and facility testing. Prior to the initial movement of any spent fuel for placement on the ISFSI storage pad, preoperational and startup tests will be performed and satisfactorily completed to verify that individual components and the storage system function properly.

9.2.1 ADMINISTRATIVE PROCEDURES FOR CONDUCTING TEST PROGRAM

Preoperational and startup test procedures will be prepared, reviewed, approved, performed, and revised in accordance with existing DCPP administrative procedures, which meet the requirements of the DCPP FSAR Update, Sections 17.5 and 17.11. Test procedures will be reviewed to determine if there is any negative impact on existing DCPP structures, systems, and components.

Preoperational test procedures prepared and performed by outside vendors at their facilities will meet the requirements of a PG&E-approved QA Program. PG&E will review and approve vendor test procedures prior to use in accordance with established DCPP procedures. PG&E personnel will witness the performance of preoperational tests performed by vendors.

9.2.2 TEST PROGRAM DESCRIPTION

The test program is divided into two parts: (a) preoperational testing, and (b) startup testing.

The objective of preoperational testing is to verify that the individual components of the storage system, facilities, and equipment meet respective functional requirements. Successful preoperational testing must be completed before commencing with startup testing. Section 9.2.3 discusses the preoperational test plan.

The objective of startup testing is to verify that the complete loading and unloading sequence, using the storage system components, facilities, and equipment work together as a complete system in accordance with requirements. Successful startup testing must be completed prior to handling spent nuclear fuel. Section 9.2.4 discusses the startup test plan.

Section 9.4 addresses testing during normal operation of the ISFSI.

Discrepancies between requirements and the results from the preoperational and startup tests will be resolved in accordance with existing DCPP problem resolution procedures.

9.2.3 PREOPERATIONAL TEST PLAN

Preoperational tests are performed on the cask transfer facility (CTF), the transporter, and all storage system ancillaries, such as the welder and forced helium dehydration to verify the components operate in compliance with the requirements of the FSAR and respective functional specifications. For example, the transporter preoperational tests will verify the controls, hydraulic system, brakes, instruments, dead-man switches, locking pins or wedges, and other components operate in compliance with the requirements of this FSAR and the transporter functional specification.

Other items to be tested are described below.

9.2.3.1 Security System

The ISFSI security system will be tested to ensure proper operation prior to startup testing.

9.2.3.2 Construction Tests

Tests associated with construction will be completed as required by construction specifications.

9.2.3.3 Calibration of Measuring and Test Equipment

Measuring and test equipment with an important-to-safety or security function will be controlled in accordance with DCPP FSAR Update, Section 17.12.

9.2.4 STARTUP TEST PLAN

An overall startup testing program procedure will control the startup tests. Individual startup test procedures will be used to supplement the approved ISFSI operation procedures as required. The startup test procedures will verify the performance of the storage system and ensure that plant equipment complies with requirements.

Actual storage system components with a MPC handling simulator will be utilized for startup testing. An MPC handling simulator will be substituted for the MPC. The MPC handling simulator will mimic the external diameter, length, and center of gravity of a loaded MPC and will be equipped with attachment locations for lift cleats. One or more MPC mock-ups will be used to test the automated welding machine, including performance of the MPC-lid-closure weld, MPC-lid-weld removal, moisture removal, helium filling, and MPC cool down.

Personnel performing and managing the physical work during startup testing will have completed applicable ISFSI training program requirements. Refer to Section 9.3.

The following operations will be included in the startup tests for the Diablo Canyon ISFSI:

(1) Preparing the transfer cask and MPC for movement into the spent fuel pool (SFP).

(2) Moving the transfer cask into the fuel handling building/auxiliary building (FHB/AB), and placement in the Unit 2 seismic restraint structure.

(3) Placing the transfer cask into the SFP and simulating movement of fuel, using a dummy fuel assembly, into the transfer cask.

(4) Removing the transfer cask from the SFP and moving it to the Unit 2 cask washdown area and into the seismic restraint structure.

(5) Decontaminating the transfer cask.

(6) Welding the MPC lid, moisture removal, filling the MPC with helium, MPC cooldown, and lid weld removal. These functions may be performed outside of the FHB/AB for ALARA reasons.

(7) Installing the transfer cask top lid.

(8) Loading the transfer cask onto the Low Profile Transporter using the FHB/AB crane and removal from the FHB/AB.

(9) Transporting the loaded transfer cask from the FHB/AB to the CTF using the transporter.

(10) Movement of 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 out of the CTF using the transporter.

(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 off a loaded overpack.

(17) Transfer of the MPC simulator from the overpack back into the transfer cask.

(18) Transporting the loaded transfer cask to the FHB/AB using the onsite transporter.

(19) Installation and operation of the supplemental cooling system.

Discrepancies between the FSAR requirements and the results from startup tests will be resolved in accordance with existing DCPP problem resolution procedures.

9.2.5 OPERATIONAL STARTUP TESTING

Additional startup testing may be performed during the initial loading of an MPC. These tests will be limited to gathering information that is available only when nuclear fuel is included in the MPC or final verification of data obtained in previous startup testing.

9.2.6 OPERATIONAL READINESS REVIEW PLAN

PG&E will perform an operational readiness review prior to the commencement of ISFSI operations for the initial set of casks placed on the ISFSI pad. The readiness review will verify that all appropriate actions have been completed prior to initial MPC loading. As a minimum, the operational readiness review plan will ensure that:

- Results of preoperational and startup testing are satisfactory and that all corrective actions and lessons learned have been incorporated into the approved ISFSI operational procedures.

- Radiological procedures and controls are in place.

- Operations procedures including surveillance, operating, and emergency response procedures are approved and in place.

- All engineering issues relating to the storage system initial use are resolved.

- Fire protection procedures are approved and in place.

- Maintenance procedures are approved and in place, and all storage system and related plant components are ready for use.

- Cask Transportation Evaluation Program is in place.

- Procedures are approved and in place that prescribe how planning is performed and verified to ensure the characteristics of selected fuel assemblies are within applicable Diablo Canyon ISFSI Technical Specifications and Section 10.2 requirements.

9.3 TRAINING PROGRAM

Pursuant to 10 CFR 72.190 and 10 CFR 72.192, personnel (including supervisory personnel who personally direct the operation of important-to-safety equipment and controls) working at the Diablo Canyon ISFSI receive training and indoctrination designed to provide and maintain a well-qualified work force for safe and effective operation of the ISFSI. The existing DCPP training programs are INPO accredited and the General Employee Training portions are directly applicable to the Diablo Canyon ISFSI. Supplemental training specific to the ISFSI is provided to operations, maintenance, security, and emergency planning personnel who are assigned duties associated with the ISFSI.

This supplemental training includes training modules developed under PG&E's training program using the Systematic Approach to Training (SAT) process to require a comprehensive, site-specific training, assessment, and qualification (including periodic requalification) program for the operation and maintenance of the ISFSI. Additional details regarding training program content; required "dry run" training; retraining requirements; records; and medical requirements are provided in the ISFSI Training Program.

9.4 NORMAL OPERATIONS

This section describes the administrative controls and conduct of operations associated with activities considered important to safety. Also described in this section is the management system for maintaining records related to the operations of the ISFSI.

9.4.1 PROCEDURES

ISFSI activities that are important to safety are conducted in accordance with detailed written approved procedures. The activities include, but are not limited to, operations identified in the Diablo Canyon ISFSI Technical Specifications (TS) and Chapter 10. Preoperational, normal operating, maintenance, and surveillance testing will be in effect prior to commencing loading operations. These procedures are briefly described in Section 9.4.1.1. These procedures, and any subsequent revisions, will be prepared, reviewed, and approved in accordance with the DCPP administrative program for procedure preparation, review, and approval, as described in DCPP FSAR Update, Section 17.5. Procedures will contain sufficient detail to allow qualified and trained personnel to properly perform the actions without incident.

9.4.1.1 Categories of Procedures

9.4.1.1.1 Administrative Procedures

Administrative procedures provide directions and instructions to Diablo Canyon personnel to provide a clear understanding of operating philosophy and management policies. These procedures include instructions pertaining to personnel conduct and procedures to prepare, review, approve, and revise procedures. Administrative procedures include actions and activities to ensure that personnel safety, the working environment, procurement, and other general Diablo Canyon ISFSI activities are carried on at a high degree of readiness, quality, and success.

9.4.1.1.2 Radiation Protection Procedures

Radiation protection procedures are used to implement the radiation protection program. These procedures ensure compliance with 10 CFR 20 and ALARA principles. The procedures describe the acquisition of data, use of equipment, and qualifications and training of personnel to perform radiation surveys, measurements, and evaluations for the assessment and control of radiation hazards associated with the Diablo Canyon ISFSI.

Under the existing DCPP radiation protection program, procedures have been developed and implemented for monitoring exposures of employees, using accepted techniques, radiation surveys of work areas, radiation monitoring of maintenance activities, and for maintaining records demonstrating the adequacy of measures taken to control radiation exposures of employees and others within prescribed limits and ALARA. These procedures are revised as necessary to address ISFSI operations prior to operation of the ISFSI. The revised procedures ensure the safety of personnel performing loading operations, transport, unloading operations, surveillance testing, and maintenance of the ISFSI. Entrance to, and work performed inside, the ISFSI protected area requires a radiation work permit and is controlled by radiation protection and security personnel.

The operation and use of radiation monitoring instrumentation at the Diablo Canyon ISFSI, including personnel monitoring equipment, along with measurement and sampling techniques, are described in written procedures. There is no need for airborne radiation monitoring since no airborne radioactivity is anticipated to be released from the casks at an ISFSI.

9.4.1.1.3 Maintenance and Surveillance Testing Procedures

Maintenance procedures control performance of preventative and corrective maintenance and for surveillance testing on Diablo Canyon ISFSI equipment and instrumentation. Preventative maintenance and surveillance testing, including calibrations and full load tests, are performed on a periodic basis to verify operability and to preclude the degradation of ISFSI systems, equipment, and components. Corrective maintenance is performed to rectify any unexpected system, equipment, or component malfunction, as the need arises.

Important-to-safety structures, systems, and components (SSCs) that are purchased commercial grade are qualified by test prior to use. Testing verifies functionality and, for structural SSCs, the ability to carry full-rated load without degradation. Subsequent to the qualification testing, preventative maintenance, surveillance testing, and corrective maintenance are as described above.

9.4.1.1.4 Operating Procedures

The operating procedures provide the instructions for routine and projected contingency (off-normal) operations, including handling, loading, sealing, transporting, storing and unloading the SSCs and for other operations important to safety. Operating procedures include off-normal occurrences and operations identified in the Diablo Canyon ISFSI TS and Chapter 10.

9.4.1.1.5 Procedures Implementing the QA Program

Procedures for important-to-safety activities ensure that the operation and maintenance of the ISFSI is performed in accordance with DCPP FSAR Update, Chapter 17and applicable regulations, the Diablo Canyon ISFSI TS, the radiation protection program, and approved procedures. The requirements for qualification of personnel operating important-to-safety equipment and controls will be specified in written and approved procedures. The quality assurance procedures will clearly communicate that the responsibility for quality rests with each individual employee or visitor who enters the facility.

9.4.2 RECORDS

ISFSI records will be maintained in accordance with established PG&E practices. The records management program is described in FSAR Update, Section 17.17.

PG&E requested an exemption from 10 CFR 72.72(d), which requires that spent fuel and high-level radioactive waste records in storage be kept in duplicate. As specified in License Condition 16 of the Diablo Canyon ISFSI License SNM-2511, the exemption allows PG&E to maintain records of spent fuel and high level radioactive waste in storage either in duplicate, as required by 10 CFR 72.72(d), or, alternatively, a single set of records may be maintained at a records storage facility that satisfies the standards of ANSI N45.2.9-1974. All other requirements of 10 CFR 72.72(d) must be met.

9.5 EMERGENCY PLANNING

The DCPP Emergency Plan for 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 DCPP. This Emergency Plan also is used for any radiological emergencies that may arise at the Diablo Canyon ISFSI. As such, the Emergency Plan complies with the provisions of 10 CFR 72.32(c).

Section 4 of the DCPP Emergency Plan and the Emergency Plan Implementing Procedures reflect the conditions and indications that require entry into the Emergency Plan. Response actions and notifications are contained in the Emergency Plan. The Emergency Action Level classification for ISFSI events is the Notification of Unusual Event.

9.6 PHYSICAL SECURITY PLAN

The purpose of the security program for the Diablo Canyon ISFSI is to establish and maintain a physical capability for the protection of the stored spent fuel. The physical security program for the Diablo Canyon ISFSI is provided in the DCPP Physical Security Plan, the Safeguards Contingency Plan, and the Security Training and Qualification Plan. This program meets the requirements contained in 10 CFR 72, Subpart H, "Physical Protection," and the applicable portions of 10 CFR 73.55.

Because the ISFSI security program contains information that is to be withheld from the public in accordance with 10 CFR 2.390(d) and 10 CFR 73.21, it was submitted as a separate document to the NRC. The program as described therein was prepared and implemented as necessary to support the ISFSI operation schedule discussed in Chapter 1 of this FSAR. A summary of physical protection features that does not include safeguards information follows.

The DCPP security force controls access to the ISFSI protected area. Access is limited to individuals who require access to perform work-related activities. The DCPP security force maintains a list of approved individuals authorized for access. Individuals granted access to the ISFSI protected area are required to display badges indicating authorization and identification. Personnel, hand-carried articles, and vehicles are searched prior to entry to the ISFSI protected area to detect the presence of explosives.

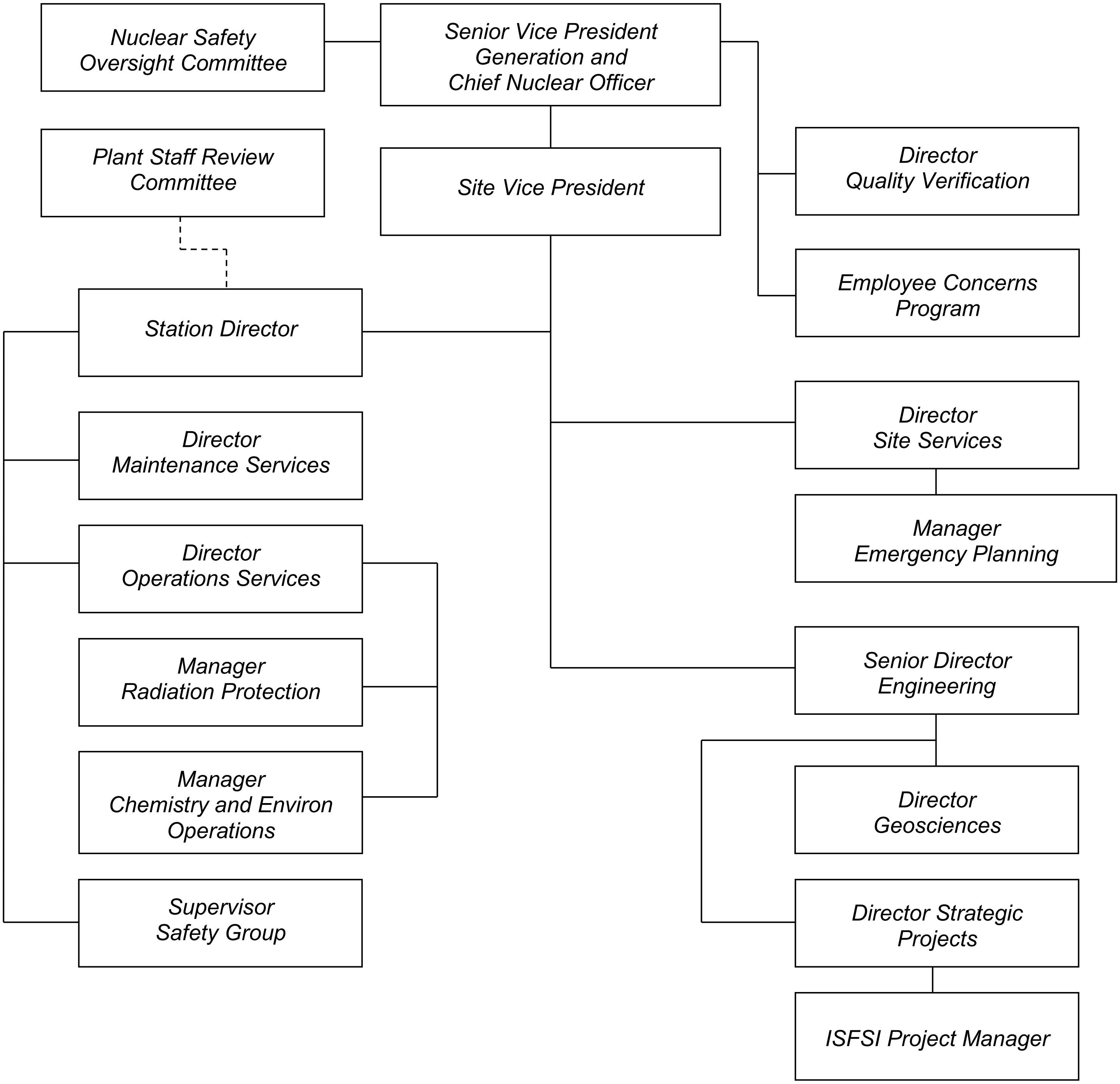
The ISFSI protected area has an intrusion detection system to detect attempted unauthorized entry. Manned alarm stations support the security program by monitoring intrusion detector system alarms, coordinating security communications, and performing closed circuit television surveillance and alarm assessment.

In accordance with 10 CFR 72.184, the DCPP Safeguards Contingency Plan addresses responses to potential threats. The Plan contains a responsibility matrix that provides guidance for corresponding security force actions. Contingency planning involves detailed response procedures and assistance from local law enforcement agencies when requested.

As stipulated in Appendix B to 10 CFR 73.55, provisions for training and qualifying security force members are contained in the DCPP Security Training and Qualification Plan. This Plan identifies crucial security tasks and the associated positions that must be trained in these tasks. The Plan also describes initial and recurring training requirements and a screening program used to determine that security force members meet prescribed background, physical, and mental qualification criteria.

Each commitment made in the DCPP Physical Security Plan, the Safeguards Contingency Plan, and the Security Training and Qualification Plan is implemented via written procedures in accordance with 10 CFR 73.55(b)(3)(i). These implementing procedures, which are developed, approved, and maintained by security management, ensure accurate and organized day-to-day security operations.

HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE UPDATED

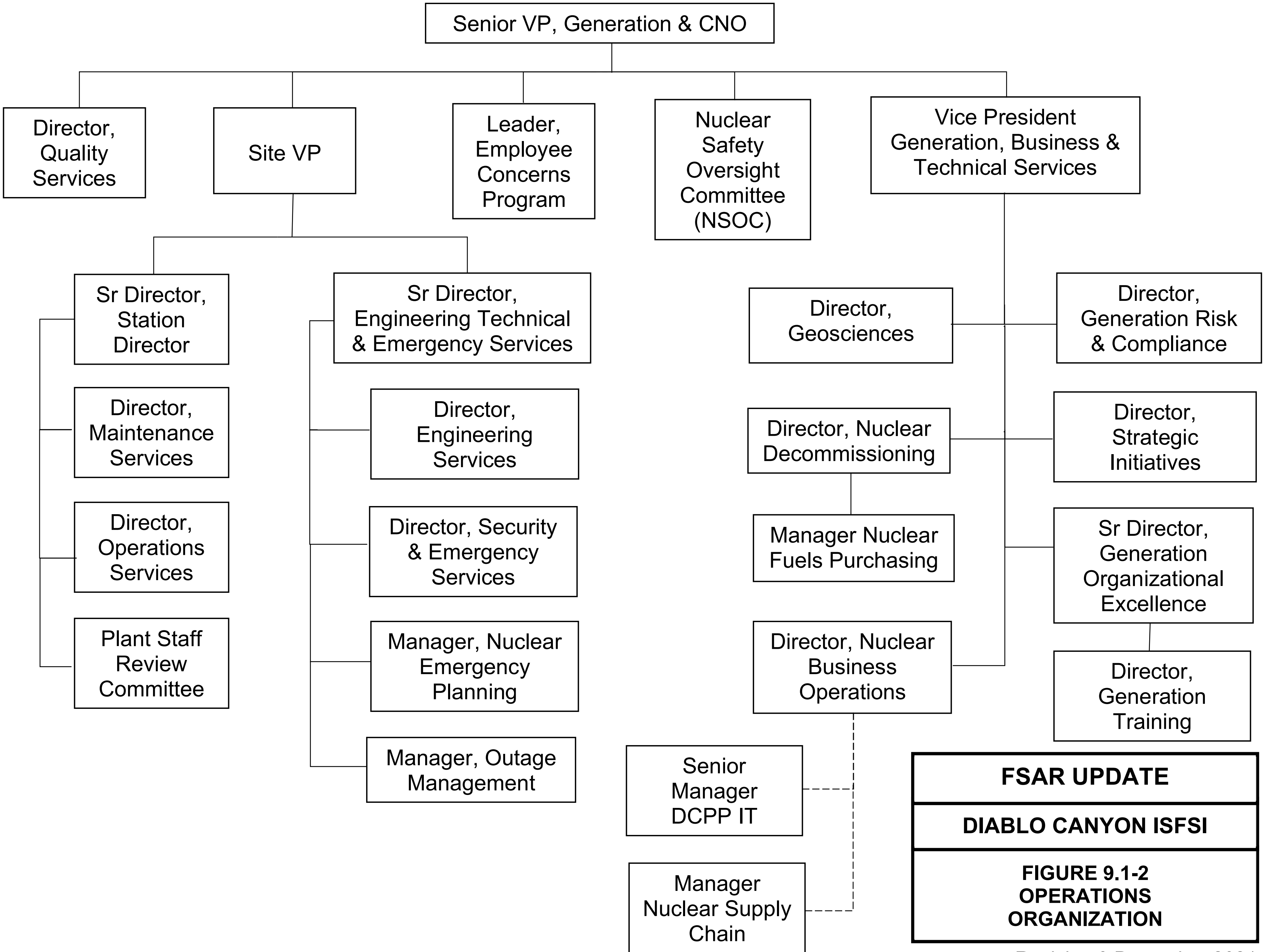


FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 9.1-1
PREOPERATIONS ORGANIZATION (HISTORICAL)

Revision 9 December 2021



FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 9.1-2
OPERATIONS ORGANIZATION

Revision 9 December 2021

CHAPTER 10

OPERATING CONTROLS AND LIMITS

CONTENTS

CHAPTER 10

OPERATING CONTROLS AND LIMITS

TABLES

CHAPTER 10

OPERATING CONTROLS AND LIMITS

FIGURES

CHAPTER 10

OPERATING CONTROLS AND LIMITS

10.1 PROPOSED OPERATING CONTROLS AND LIMITS

The Diablo Canyon ISFSI storage system is totally passive and requires minimal operating controls. The Diablo Canyon ISFSI employs a proven technology, stringent codes of construction, and comprehensive quality assurance measures. As a result, it has substantial design and safety margins. The areas where controls and limits are necessary to ensure safe operation of the Diablo Canyon ISFSI are provided in Table 10.1-1.

The items in this chapter that are to be controlled are selected based on the design criteria and safety analyses for normal, off-normal, and accident conditions.

10.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides an overview of, and the general bases for, operating controls and limits specified for the Diablo Canyon ISFSI.

10.2.1 FUNCTIONAL AND OPERATING LIMITS, MONITORING INSTRUMENTS, AND LIMITING CONTROL SETTINGS

To be consistent with the guidance contained in Interim Staff Guidance Document 11 (ISG-11), Revision 3, issued on November 17, 2003 (Reference 3), fuel assemblies to be stored initially at the Diablo Canyon ISFSI were limited to a nominal maximum average burnup of ≤45,000 MWD/MTU (defined in ISG-11 as low burnup fuel) (see PG&E Letter DIL-04-002, dated January 16, 2004).

Because the HI-STORM 100 System licensing and design basis incorporated by reference in this FSAR was originally taken from Revision 1A of the HI-STORM 100 System FSAR, many of the design and safety evaluations discussed in this FSAR were for bounding burnups exceeding those initially authorized for loading at the Diablo Canyon ISFSI (see, for example, the ISFSI thermal design discussed in Section 4.2, the radiological analyses in Chapter 7, and selected accident analyses in Chapter 8). Based on the fuel burnup limit of ≤45,000 MWD/MTU, these generic design and safety evaluations were conservative and bounded the allowed cask contents.

The fuel burnup limit is specified in the Diablo Canyon ISFSI Technical Specifications. A review of Materials License SNM-2511 and it associated Safety Evaluation Report; PG&E Letter DIL-04-002; and ISG-11, Revision 3, shows there is no regulatory requirement to include burnup uncertainty when evaluating compliance with TS burnup limits. Therefore, burnup uncertainty will not be applied to calculated fuel assembly burnup values when evaluating the eligibility of fuel assemblies for storage at the Diablo Canyon ISFSI. However, PG&E will conservatively apply a 5 percent burnup uncertainty allowance when calculating the decay heat for each loaded MPC.

The NRC reviewed and accepted a generic HI-STORM System design that would allow higher fuel burnups for loading, consistent with the guidance of ISG-11, Revision 3. This approval has been included in CoC License Amendment 1014-2. License Amendment 2, issued by the NRC on January 19, 2012, updated the authorized contents for the Diablo Canyon ISFSI to store fuel with higher burnups consistent with HI-STORM CoC Amendment 3 in the MPC-32.

The NRC issued License Amendment 3 on February 11, 2014, updated the allowed content to a 28.74 kilowatt heat load for uniform loading and 25.572 kilowatt heat load for regionalized loading. This is supported by the Holtec International Document No. HI-2125191, “Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System with up to 28.74 kW Decay Heat,” Revision 6 (Reference 6). That analysis also demonstrates the requirement for a supplemental cooling system

(SCS) during the cask processing while utilizing temporary shielding on the transfer cask.

This section provides requirements for the controls or limits that apply to operating variables classified as important to safety and are observable and measurable. The operating variables required for the safe operation of the Diablo Canyon ISFSI are:

- Spent fuel characteristics
- Spent fuel storage cask (SFSC) heat removal capability
- Multi-purpose canister (MPC) dissolved boron concentration level
- Annulus gap water requirement during moisture removal for loading and reflooding for unloading
- Water temperature of a flooded MPC
- MPC vacuum pressures
- MPC recirculation gas exit temperature
- Helium purity
- MPC helium backfill pressures
- Gas exit temperature of a MPC prior to reflooding
- Supplemental Cooling System (SCS)

Each of the specifications for these characteristics is provided below with the exception of the MPC dissolved boron concentration, and heat removal parameters, which are provided in the Diablo Canyon ISFSI TS and their bases. Although provided in the sections below, the TS and bases also provide Limiting Conditions for Operation and bases for maintaining the integrity of the MPC during loading and unloading. These include vacuum pressure, recirculation gas temperature, backfill pressure, and leak rate during loading, and exit gas temperature during unloading.

Limitations on nonfuel hardware to be stored with their associated fuel assemblies are provided in Table 10.2-10.

10.2.1.1 Fuel Characteristics

The Diablo Canyon ISFSI is designed to provide interim storage for up to 4,400 fuel assemblies, which accommodates the number of assemblies predicted to be used during the licensed operating life of the plant. The Diablo Canyon ISFSI storage system uses four MPC types for the storage of fuel assemblies, fuel debris and associated nonfuel hardware. The DCPP fuel is normally stored as nonconsolidated fuel assemblies both with and without control components. The intact fuel assemblies are stored in either the MPC-24, MPC-24E, MPC-24EF, or MPC-32 canisters. The damaged fuel assemblies can only be stored in MPC-24E or MPC-24EF canisters, and the fuel debris can only be stored in MPC-24EF canisters. Damaged fuel or fuel debris will be placed in a damaged fuel container before loading into an MPC. The fuel debris can be consolidated; however, the amount of debris is limited to the equivalent of a single intact fuel assembly.

Fuel qualification is based on the requirements for criticality safety, decay heat removal, radiological protection, and structural integrity. The analysis presented in Chapters 4, 7, and 8 documents the qualification of DCPP inventory of spent fuel assemblies and associated nonfuel hardware for storage in the Diablo Canyon ISFSI storage system design.

During the operation of DCPP, fuel integrity has been, and continues to be, monitored. Through the detection of radiochemistry changes in the reactor coolant system, most fuel damage is assessed. When damaged rods are suspected, assemblies are inspected as they are removed from the core. All assemblies with positive indication of damage are again inspected in the spent fuel pool (SFP) to determine numbers and location of rods in the assembly that have failed cladding. If the fuel assembly is to be placed back in the reactor core, any failed rods are removed and replaced with nonfuel rods of equivalent dimensional properties. If the suspected damaged fuel assemblies are at the end of their cycle, the assemblies may be stored in the SFP without repair. During this process, all known rod failures are noted and their assemblies are tracked. If the failure is visible from the exterior of the assembly, the damage may be video taped. For assemblies that are removed from the reactor core and were not inspected at that time, inspections will be performed prior to loading these assemblies into an MPC for storage. This will ensure that there are no undetected failed rods in any assembly that is placed in an MPC.

Under this failure detection process, inspections to date have found limited failures. Where single failed rods have been identified and removed, they are being stored in the SFP and will ultimately be stored in an MPC that can contain fuel debris. This detection process, along with the past history of plant operations and SFP fuel storage, provide a high level of confidence that the current spent fuel and associated nonfuel hardware will meet the criteria for storage in the appropriate MPC. In addition, based on the condition of the current spent fuel, the continued maintenance of the reactor coolant and SFP water chemistry requirements, and proper handling of the fuel, there is a high level of confidence that future spent fuel assemblies will meet the criteria for storage in the appropriate MPC.

A cask-loading plan ensures that no damaged fuel assemblies are loaded into an MPC-24 or MPC-32 canister. Damaged fuel is only stored in either an MPC-24E or MPC-24EF canister. Fuel debris is only stored in an MPC-24EF canister. If the structural integrity criterion is met, then approval for dry storage for a given assembly is made. This qualification is documented and subsequently referenced in Diablo Canyon ISFSI operating procedures prior to loading spent fuel assemblies into the MPC.

The cask-loading plan provides a loading sequence based on the various characteristics of the fuel assemblies being loaded. There are two main fuel-loading strategies used: uniform fuel loading and regionalized fuel loading. Both of these loading strategies are designed to ensure that the design bases of the fuel, MPCs, and overpacks are maintained.

Uniform fuel loading is used when the fuel assemblies being loaded are all of similar burnup rates, decay heat levels, and post-irradiation cooling times. In this case the actual location of each assembly is less critical and assemblies can be placed at any location in the MPC.

Regionalized fuel loading is used when high heat emitting fuel assemblies are to be stored in an MPC. This loading strategy allows these specific assemblies to be stored in locations in the center of the MPC basket provided lower heat emitting fuel assemblies are stored in the peripheral storage locations. Use of regionalized fuel loading must consider other restrictions on loading such as those for nonfuel hardware and damaged fuel containers.

The following controls ensure that each fuel assembly is loaded into a known cell location within a qualified MPC:

- A cask-loading plan is independently verified and approved.

- A fuel movement sequence is based upon the written loading plan. All fuel movements from any rack location are performed under controls that ensure strict, verbatim compliance with the fuel movement sequence.

- Prior to placement of the MPC lid, all fuel assemblies and associated nonfuel hardware, if included, is either video taped or visually documented by other means, and independently verified, by ID number, to match the fuel movement sequence.

A cognizant engineer is responsible for performing a third independent verification to ensure that the fuel in the MPCs is placed in accordance with the original cask-loading plan.

Based on the qualification process of the spent fuel and the administrative controls used to ensure that each fuel assembly is loaded into the correct location within an MPC, incorrect loading of an MPC is not considered to be a credible event.

10.2.1.2 Fuel Characteristics (Allowable Content)

The characteristics of the fuel that are allowable for storage in the MPCs are as follows:

- Intact fuel assemblies, damaged fuel assemblies, fuel debris, and nonfuel hardware meeting the limits specified in Tables 10.2-1, 10.2-2, 10.2-3, and 10.2-4 and other referenced tables may be stored in the SFSC system. These FSAR tables and specifications are duplicated in Tables 2.1-1 through 2.1-10 of the Diablo Canyon ISFSI TS.

- For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining intact fuel assemblies in the MPC shall meet the decay heat generation limits for the damaged fuel assemblies. This requirement applies only to uniform fuel loading.

10.2.1.2.1 Alternate MPC-32 Fuel Selection Criteria

To allow loading of high burnup fuel assemblies in the Diablo Canyon ISFSI site specific MPC-32, without changing the allowed heat load or helium fill pressure, the fuel loading selection criteria of HI-STORM CoC Amendment 3 (Reference 4) were added:

The maximum allowable fuel assembly average burnup for a given MINIMUM ENRICHMENT is calculated as described below for a minimum cooling time of 5 years using the maximum permissible decay heat determined in Table 10.2-7 or 10.2-9 as appropriate for uniform and regionalized loadings. Different fuel assembly average burnup limits may be calculated for different minimum enrichments (by individual fuel assembly) for use in choosing the fuel assemblies to be loaded into a given MPC.

1. Choose a fuel assembly minimum enrichment E_{235}.

2. Calculate the maximum allowable fuel assembly average burnup for a minimum cooling time of 5 years using the following equation:

$$Bu = (A\times q) + (B\times q^2) + (C\times q^3) + [D\times(E_{235})^2] + (E\times q\times E_{235}) + (F\times q^2\times E_{235}) + G$$

Where:

Bu = Maximum allowable average burnup per fuel assembly (MWD/MTU)

q = Maximum allowable decay heat per storage location, in kilowatts (e.g. for 898 watts, use 0.898), determined from Table 10.2-7 or 10.2-9.

E_{235} = Minimum fuel assembly average enrichment (wt% ^{235}U). For example, for 4.05 wt%, use 4.05.

A through G = Coefficients from Table 10.2-11.

3. Calculated burnup limits shall be rounded down to the nearest integer.

4. Calculated burnup limits greater than 68,200 MWD/MTU must be reduced to be equal to this value.

5. Linear interpolation of calculated burnups between cooling times for a given fuel assembly maximum decay heat and minimum enrichment is permitted. For example, the allowable burnup for a cooling time of 5.5 years may be interpolated between those burnups calculated for 5 years and 6 years.

6. Each ZR-clad fuel assembly to be stored must have a MINIMUM ENRICHMENT greater than or equal to the value used in Step 1.

7. When complying with the maximum fuel storage location decay heat limits, users must account for the decay heat from both the fuel assembly and any NON-FUEL HARDWARE, as applicable for the particular fuel storage location, to ensure the decay heat emitted by all contents in a storage location does not exceed the limit.

10.2.1.3 Uniform Fuel Loading

Fuel assemblies used in uniform fuel loading shall meet all applicable limits specified in Tables 10.2-1, 10.2-2, 10.2-3, 10.2-4, and 10.2-5. Fuel assembly burnup, decay heat, and cooling time limits for uniform loading are specified in Tables 10.2-6 and 10.2-7 and Section 10.2.1.2.1.

10.2.1.4 Regionalized Fuel Loading

Fuel may be stored using regionalized loading in lieu of uniform loading to allow higher heat emitting fuel assemblies to be stored than would otherwise be able to be stored using uniform loading. Figures 10.2-1 through 10.2-3 (these figures are duplicated in the Diablo Canyon ISFSI TS as Figures 2.1-1 through 2.1-3), define the regions for the MPC-24; MPC-24E/MPC-24EF; and MPC-32 models, respectively. Fuel assembly burnup, decay heat, and cooling time limits for regionalized loading are specified in Tables 10.2-8 and 10.2-9, or Section 10.2.1.2.1. In addition, fuel assemblies used in regionalized loading shall meet all other applicable limits specified in Tables 10.2-1, 10.2-2, 10.2-3, 10.2-4, and 10.2-5.

10.2.1.5 For Allowable Content - Functional and Operating Limits Violations

If any fuel specifications or loading conditions above are violated, the following Diablo Canyon ISFSI TS actions shall be completed:

- The affected fuel assemblies shall be placed in a safe condition.

- Within 24 hours, notify the NRC Operations Center.
- Within 30 days, submit a special report that describes the cause of the violation, and actions taken to restore compliance and prevent recurrence.

10.2.2 MPC LOADING CHARACTERISTICS

The confinement of radioactivity during the storage of spent fuel and associated nonfuel hardware in the MPC is ensured by the structural integrity of the strength-welded MPC. However, long-term integrity of the fuel and cladding depends on storage in an inert heat removal environment inside the MPC. This environment is established by removing water from the MPC and backfilling the cavity with an inert gas.

The loading process of an MPC involves placing a transfer cask with an empty MPC in the SFP and loading it with fuel assemblies (intact or damaged that meet the specifications for allowable content discussed above), fuel debris, and/or nonfuel hardware allowed per the type of MPC. Once this is complete a lid is then placed on the MPC. The transfer cask and MPC are raised to the SFP surface. The transfer cask and MPC are then moved into the cask washdown area where dose rates are measured and the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and moisture removal is performed. The MPC cavity is backfilled with helium. Additional dose rates are measured and the MPC vent and drain cover plates and closure ring are installed and welded. Nondestructive examination (NDE) inspections are performed on the welds.

As a part of the loading process there are several characteristics that must be maintained to ensure that the allowable contents placed in any MPC remains stable and intact. These characteristics involve maintaining the MPC cavity temperature. During the loading process there are times when the loaded MPC is water filled and times when it is empty of water. As a result, there are characteristics that must address each of these two conditions. One of these characteristics is MPC water temperature. The other characteristic is maintaining the water level and recirculation in the annular gap between the transfer cask and the MPC, which only applies to vacuum drying.

Also during the loading process there are several characteristics vital to ensuring that the resulting MPC internal environment is conducive to long-term heat removal and maintaining the integrity of the fuel cladding. These characteristics are: limiting the moisture in the MPC; backfilling the MPC with high quality inert gas; and limiting the leakage of this inert environment over time. The dry, inert and sealed MPC atmosphere is required to be in place during transport and storage operations.

10.2.2.1 Annulus Gap Water Requirement

During an unloading process the annular gap shall be filled with water prior to removal of the inert environment in the MPC cavity. The annulus gap must be kept free of water if the forced helium dehydration (FHD) system is used for MPC moisture removal.

Additionally, the annulus gap is filled when the SCS is required. See Section 10.2.2.7 for details.

10.2.2.2 MPC Water Temperature

During the loading and unloading processes, maintaining the integrity of the fuel in the MPC is the critical activity. As a result of decay heat produced by the spent fuel assemblies, providing a coolant source is imperative to maintaining control of cladding temperature and the fuel integrity. During these processes when there is water in the MPC, the water is considered the coolant source. As long as there is water in the MPC it will continue to perform the coolant function. This water should continue to perform its function as long as it does not reach the boiling temperature. As a result, the parameter that will best indicate the potential reduction of water would be the temperature of the water in the MPC. However, since monitoring the water temperature in the MPC directly may not always be possible, an analysis of the potential for the water to reach the boil-off temperature is performed to ensure that the boil-off temperature cannot be reached. This analysis is based on the decay heat levels of the contents and the various volumes of water in the MPC as it is loaded. The results of this analysis provide any time limitation or any requirement for compensatory measures.

While there is water in the MPC, there is adequate assurance through analysis that the temperature of that water in the MPC will not reach the boil-off level and that the volume of water in the MPC is not allowed to decrease significantly.

10.2.2.3 MPC Drying Characteristics

The cavity moisture removal is performed by the FHD system after the MPC has been drained of water. See Figure 10.2-4 for a schematic diagram of the FHD system. The NUREG-1567 acceptance criterion for dryness is ≤ 1 gram-mole per cask of oxidizing gases. This has been translated by the industry to be 3 torr for vacuum drying. For the recirculation drying process using the FHD system, measuring the temperature of the gas exiting the demoisturizer of the FHD system provides an indication of the amount of water vapor entrained in the helium gas in the MPC. Maintaining a demoisturizer exit temperature of less than or equal to 21°F for 30 minutes or more during the recirculation drying process ensures that the partial pressure of the entrained water vapor in the MPC is less than 3 torr.

When the FHD system is used, the remaining moisture in the MPC cavity is removed after all of the water that can practically be removed through the drain line using a hydraulic pump has been expelled in the water blowdown operation. The recirculation process using the FHD involves introducing dry gas into the MPC cavity that absorbs the residual moisture in the MPC. This humidified gas exits the MPC and the absorbed water is removed through condensation and/or mechanical drying. The dried gas is then forced back through the MPC until the gas exit temperature from the FHD demoisturizer is ≤ 21°F for at least 30 minutes. Meeting these temperature and time criteria ensures that the cavity is dry and the moisture level in the MPC is acceptable. The FHD system shall be designed to ensure that during normal operation (that is, excluding startup and shutdown ramps) the following criteria are met:

(1) The temperature of helium gas in the MPC shall be at least 15°F higher than the saturation temperature at coincident pressure.

(2) The pressure in the MPC cavity space shall be less than or equal to 60.3 psig (75 psia).

(3) The recirculation rate of helium shall be sufficiently high (minimum hourly throughput equal to ten times the nominal helium mass backfilled into the MPC for fuel storage operations) so as to produce a turbulent flow regime in the MPC cavity.

(4) The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr if the helium temperature at the demoisturizer outlet is ≤ 21°F for a period of 30 minutes.

In addition to the above system design criteria, the individual modules shall be designed in accordance with the following criteria:

(1) The condensing module shall be designed to devaporize the recirculating helium gas to a dew point of 120°F or less.

(2) The demoisturizer module shall be configured to be introduced into its helium conditioning function after the condensing module has been operated for the required length of time to ensure that the bulk moisture vaporization in the MPC has been completed.

(3) The helium circulator shall be sized to effect the minimum flow rate of circulation required by the system design criteria described above.

(4) The preheater module shall be engineered to ensure that the temperature of the helium gas in the MPC meets the system design criteria described above.

The design of the FHD system is subject to the confirmatory analyses listed below to ensure that the system will accomplish the performance objectives set forth in this FSAR.

(1) System thermal analysis in Phase 1: Characterize the rate of condensation in the condensing module and helium temperature variation under Phase 1 operation (i.e., the scenario where there is some unevaporated water in the MPC) using a classical thermal-hydraulic model wherein the incoming helium is assumed to fully mix with the moist helium inside the MPC.

(2) System thermal analysis in Phase 2: Characterize the thermal performance of the closed loop system in Phase 2 (no unvaporized moisture in the MPC) to predict the rate of condensation and temperature of the helium gas exiting the condensing and the demoisturizer modules. Establish that the system design is capable to ensure that partial pressure of water vapor in the MPC will reach less than or equal to 3 torr if the temperature of the helium gas exiting the demoisturizer is predicted to be at a maximum of 21°F for 30 minutes.

(3) Fuel Cladding Temperature Analysis: A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in HI-STORM 100 FSAR Subsections 4.4.1.1.1 through 4.4.1.1.4 with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation (design maximum heat load, no moisture, and maximum helium inlet temperature), is below the fuel cladding temperature limit for normal conditions of 400 °C.

If Diablo Canyon is the first user of the FHD system designed and built for the MPC drying function, the system will be subject to confirmatory testing as follows:

(1) A representative quantity of water will be placed in a manufactured MPC (or equivalent mock-up) and the closure lid and RVOAs installed and secured to create a hermetically sealed container.

(2) The MPC cavity drying test will be conducted for the worst case scenario (no heat generation within the MPC available to vaporize water).

(3) The drain and vent line RVOAs on the MPC lid will be connected to the terminals located in the preheater and condensing modules of the FHD system, respectively.

(4) The FHD system will be operated through the moisture vaporization (Phase 1) and subsequent dehydration (Phase 2). The FHD system operation will be stopped after the temperature of helium exiting the demoisturizer module has been at or below 21°F for 30 minutes (nominal). Thereafter, a sample of the helium gas from the MPC will be extracted and tested to determine the partial pressure of the residual water vapor in it. The FHD system will be deemed to have passed the acceptance testing if the partial pressure in the extracted helium sample is less than or equal to 3 torr.

At completion of the drying operation using the FHD system, the partial pressure of the helium/water vapor will be at 3 torr or less, however, the total pressure in the MPC will be approximately 2000 torr or 3 atm. To complete the process, when the FHD system is used, the FHD system is adjusted to provide a stable temperature in the MPC, and the pressure is adjusted to establish the helium fill conditions adjusted to the current MPC temperature. Maintaining positive pressure and helium flow through the MPC during the drying process ensures that the fuel cladding short-term temperature limit is not exceeded.

If the cavity moisture removal limits are not met, an engineering evaluation will be necessary to determine the potential quantity of moisture left within the MPC cavity. Once the quantity of moisture potentially left in the MPC cavity is determined, a corrective action plan shall be developed and actions initiated to the extent necessary to return the MPC to an analyzed condition. As the quantity of moisture estimated can range over a broad scale, different recovery strategies may be necessary.

Since moisture remaining in the cavity may represent a potential long-term degradation concern, immediate action is not necessary. The actions to develop and initiate the corrective actions should be undertaken as soon as possible commensurate with the safety significance of the condition. Completion times for the determined corrective actions will be controlled by the DCPP corrective actions program and will be determined and controlled based on the safety significance of the condition.

10.2.2.4 MPC Helium Backfill Characteristics and Purity

Having the proper helium backfill pressure or density ensures adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC. During the loading operation, once the dryness limits are met, the MPC cavity is backfilled with helium to provide the inert environment required for long-term storage. To ensure the proper environment is established the helium used in the backfill process shall have a purity of ≥ 99.995 percent. In addition, the helium backfill pressure shall be verified during loading to be ≥ 34 psig and ≤ 40 psig corrected to a baseline temperature of 70°F. For MPCs loaded to Amendment 2 and earlier of this license, the helium backfill pressure was verified during loading to be ≥ 29.3 psig and ≤ 33.3 psig corrected to a baseline temperature of 70°F.

If it has been determined that the helium backfill pressure limit has not been met, an engineering evaluation shall be undertaken to determine the actual helium pressure within the MPC cavity. Since too much or too little helium in the MPC cavity represents a potential overpressure or heat removal degradation concern, the engineering evaluation shall be performed in a timely manner commensurate with the safety significance of the condition (that is, if it is not addressed there is a possibility of a failure to adequately cool the contained fuel resulting in cladding damage).

Once the helium pressure in the MPC cavity is determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the helium pressure estimated can range over a broad scale, different recovery strategies may be necessary. Completion times for the determined corrective actions will be controlled by the DCPP corrective action program and will be determined and controlled based on the safety significance of the condition.

10.2.2.5 MPC Leakage Characteristics

The MPC helium leak rate limit ensures there is adequate helium in the MPC for long-term storage and proper heat removal. Because the lid to shell weld is relieved from leak testing per ISG-18, "The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Confinement Boundary for Spent Fuel Transportation," leak rate acceptance limit is limited to the vent and drain port closure welds which are verified to meet the mass-like leaktight criteria of ANSI N14.5 (1997). This is defined as the rate of change of the pressure-volume product of the leaking fluid at test conditions. This allows the leakage rate as measured by a mass spectrometer leak detector (MSLD) to be compared directly to the acceptance limit without the need for unit conversion from test conditions to standard, or reference conditions.

During transport operations or storage operations if the vent and drain port closure weld helium leak rate limit is determined not to be met, an engineering evaluation shall be performed to determine the impact of increased helium leak rate on heat removal and offsite dose. Since the SFSC is a ventilated system, any leakage from the MPC is transported directly to the environment. An increased helium leak rate represents a potential challenge to MPC heat removal and the offsite doses calculated in this FSAR confinement analyses, reasonably rapid action is warranted.

Once the cause and consequences of the elevated leak rate from the MPC are determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the recovery mechanisms can range over a broad scale based on the evaluation performed, different recovery strategies may be necessary. An elevated helium leak rate represents a challenge to heat removal rates and offsite doses, reasonably rapid action and completion of the corrective actions shall be commensurate with the safety significance of the condition. Completion times for the determined corrective actions are controlled by the DCPP corrective action program and will be determined based on the safety significance of the condition

10.2.2.6 Returning MPC to Safe Condition

If for a loaded MPC the fuel cavity dryness, backfill pressure, or helium leakage rate cannot be successfully met or maintained for any reason, the MPC must be returned to a safe analyzed condition, which may ultimately require the fuel to be placed back in the SFP. The completion time for this effort shall be based on the safety significance of the condition. The completion time shall consider the time required to perform fuel cool-down operations, reflood the MPC, cut the MPC lid welds, move the transfer cask into the SFP, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

10.2.2.7 Supplemental Cooling System Requirements

The Diablo Canyon ISFSI system thermal analysis (HI-2104625, Reference 5) demonstrates that the temperature of the MPC surface will be at the boiling temperature of water (~232°F), and the fuel cladding temperatures will be lower than the fuel cladding temperature limit of 752°F, if standing water is maintained in the MPC/HI-TRAC annulus space. To ensure standing water is maintained in the annulus, an annulus keep-full system is used for loading operations while utilizing temporary shielding on the transfer cask, and for unloading operations of MPCs loaded under Amendment 2 of this license.

When the SCS is required, SCS operability is verified every two hours. The accident condition for a loss of SCS is discussed in Section 8.2.17.

10.2.3 MPC UNLOADING CHARACTERISTICS

In the event that an MPC must be unloaded, the transfer cask with its enclosed MPC is returned to the auxiliary building/fuel handling building to begin the process of fuel unloading. The MPC closure ring, and vent and drain port cover plates are then removed. The MPC gas is sampled to determine the integrity of the spent fuel cladding. The MPC is attached to the cool-down system. The cool-down system is a closed-loop forced ventilation gas cooling system that cools the fuel assemblies by cooling the surrounding helium gas inside the MPC.

During fuel cool-down, the MPC/transfer cask annular gap is reflooded with water to ensure adequate cooling capability is maintained. Once the fuel cool-down process is complete the MPC is reflooded with borated water and the MPC lid weld is removed leaving the MPC lid in place. The transfer cask and MPC are placed in the SFP and the MPC lid is removed. The contents are removed from the MPC and the MPC and transfer cask are removed from the SFP and decontaminated.

10.2.3.1 Gas Exit Temperature Of An MPC Prior To Reflooding

The integrity of the MPC depends on maintaining the internal cavity pressures within design limits. During the unloading process, reducing the fuel cladding temperatures significantly reduces the temperature gradients across the cladding, thus minimizing thermally-induced stresses on the cladding during MPC reflooding. In addition, reducing the MPC internal temperatures eliminates the risk of high MPC pressure due to sudden generation of steam during reflooding. This is accomplished by using the cool-down system that reduces the MPC internal temperatures such that there is no sudden formation of steam during MPC reflooding. Monitoring the circulating MPC gas exit temperature from the cool-down system ensures that there will be no large thermal gradient across the fuel assembly cladding during reflooding, which could be potentially harmful to the cladding. The exit gas temperature limit of ≤ 200°F ensures that the MPC gas exit temperature will closely match the desired fuel cladding temperature prior to reflooding the MPC. This temperature was selected to be lower than the boiling temperature of water with additional margin to eliminate the possibility of flashing to steam during reflooding.

During the fuel cool-down process, if the MPC helium gas exit temperature limit is not met, proceeding with reflooding shall be prohibited and actions must be taken to restore the parameters to within the limits before reflooding. In addition, while this parameter is being restored within limits, the proper conditions must be verified to exist for the transfer of heat from the MPC to the surrounding environs to ensure the fuel cladding remains below the short-term temperature limit. Maintaining the annular gap water level between the MPC and the transfer cask ensures that adequate cooling capability exits.

10.2.4 OTHER OPERATING CONTROLS AND LIMITS

None

10.2.5 LIMITING CONDITIONS FOR OPERATION

10.2.5.1 Equipment

All Diablo Canyon ISFSI equipment important to safety is passive in nature, therefore, there are no limiting conditions regarding minimum available equipment or operating characteristics. The MPC, transfer cask, CTF, and overpack have been analyzed for all credible equipment failure modes and extreme environmental conditions. No credible postulated event results in damage to fuel, release of radioactivity above acceptable limits, or danger to the public health and safety. All operational equipment is to be maintained, tested, and operated according to the implementing procedures developed for the ISFSI. The failure or unavailability of any operational equipment can delay the transfer of an MPC to the transfer cask or to the SFSC, but would not result in an unsafe condition.

10.2.5.2 Technical Conditions and Characteristics

The following technical conditions and characteristics are required for the Diablo Canyon ISFSI:

- Spent fuel characteristics
- SFSC heat removal capability
- MPC dissolved boron concentration level
- Annulus gap water requirement during reflooding for unloading, and operation of the SCS
- Water temperature of a flooded MPC

- MPC recirculation gas exit temperature
- Helium purity
- MPC helium backfill pressures
- Gas exit temperature of an MPC prior to reflooding

The spent fuel specifications for allowable content for storage in the ISFSI and their bases are detailed in Section 10.2.1. In addition, the spent fuel specifications are also contained in Diablo Canyon ISFSI TS Section 2.0. A description of bases for selecting the above remaining conditions and characteristics are detailed in Sections 10.2.2 through 10.2.4, with the exception of the heat removal capability, and dissolved boron concentration. These are provided in the Diablo Canyon ISFSI TS bases. Although provided in the above sections, the Diablo Canyon ISFSI TS and TS Bases also provide Limiting Conditions for Operations and bases for maintaining the integrity of the MPC during loading and unloading. These include recirculation gas temperature, backfill pressure and leak rate during loading, exit gas temperature during unloading, and SCS operation during loading and unloading operations of MPCs loaded under Amendment 2 of the license.

The technical and operational considerations are to:

- Ensure proper internal MPC atmosphere to promote heat transfer, minimize oxidation, and prevent an uncontrolled release of radioactive material.
- Ensure that dose rates in areas where operators must work are ALARA and that all relevant dose limits are met.
- Ensure that the fuel cladding is maintained at a temperature sufficiently low to preclude cladding degradation during normal storage conditions.

Through the analyses and evaluations provided in Chapters 4, 7, and 8, this FSAR demonstrates that the above technical conditions and characteristics are adequate and that no significant public or occupational health and safety hazards exist.

10.2.6 SURVEILLANCE REQUIREMENTS

The analyses provided in this FSAR show that the Diablo Canyon ISFSI and the storage system fulfill its safety functions during all accident conditions as described in Chapter 8. Surveillance requirements are provided in the Diablo Canyon ISFSI TS. No continuous surveillance of the MPC is required during long-term storage. Surveillance of the SFSC duct screens is in the Diablo Canyon ISFSI TS and ensures freedom of air movement and adequate heat dissipation during long-term storage.

10.2.7 DESIGN FEATURES

The following storage system design features are important to the safe operation of the Diablo Canyon ISFSI and require design controls and limits:

- Material mechanical properties for structural integrity confinement and shielding

- Material composition and dimensional control for subcriticality

- Decay heat removal

Component dimensions are not specified here since the combination of materials, dose rates, criticality safety, and component fit-up define the operable limits for dimensions (that is, thickness of shielding materials, thickness of concrete, MPC plate thicknesses, etc.) The values for these design parameters are specified in the HI-STORM 100 System FSAR (Reference 1). Changes to any of these design features will be implemented only after conducting a safety evaluation in accordance with 10 CFR 72.48.

The combination of the above controls and limits and those discussed previously in Section 10.2 define requirements for the Diablo Canyon ISFSI storage system components that provide radiological protection and structural integrity during normal storage and postulated accident conditions.

10.2.8 ADMINISTRATIVE CONTROLS

Use of the existing DCPP organizational and administrative systems and procedures, record keeping, review, audit, and reporting requirements coupled with the requirements of this FSAR ensure that the operations involved in the storage of spent fuel at the ISFSI are performed in a safe manner. This includes both the selection of assemblies qualified for ISFSI storage and the verification of assembly identification numbers prior to and after placement into individual MPCs. The spent fuel qualification, identification, and control are discussed in Sections 10.2.1 through 10.2.4 above. Other administrative programs will control revisions to the Diablo Canyon ISFSI TS Bases; radioactive effluents; fuel-cladding-oxide thickness; MPC loading and unloading processes; ISFSI operations, and transportation route conditions. These other programs are defined in the Diablo Canyon ISFSI TS.

10.2.9 OPERATING CONTROL AND LIMIT SPECIFICATIONS

The operating controls and limits applicable to the Diablo Canyon ISFSI, as documented in this FSAR, are delineated in the Diablo Canyon ISFSI TS and the TS Bases. These include:

- MPC dryness, backfill pressure and leak rate limitations

- SFSC heat removal capability
- Fuel Cool-Down exit gas temperature limitation
- Dissolved boron concentration

10.2.10 REFERENCES

Detailed information describing the HI-STORM 100 System is provided in the following references, which must be used together:

1. Final Safety Analysis Report for HI-STORM 100 System, Revision 1A, January 2003.

2. Deleted in Revision 2.

3. Interim Staff Guidance Document 11 (ISG-11), Revision 3, Cladding Considerations for the Transportation and Storage of Spent Fuel, NRC, November 17, 2003.

4. 10 CFR 72 Certificate of Compliance No. 1014 for the HI-STORM 100 System, Holtec International, Amendment 3, May 29, 2007.

5. Holtec International Document No. HI-2104625, “Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System Design,” Revision 10.

6. Holtec International Document No. HI-2125191, “Three Dimensional Thermal-Hydraulic Analyses for Diablo Canyon Site-Specific HI-STORM System with up to 28.74 kW Decay Heat,” Revision 6.

Reference 1 contains information related to MPC-32, MPC-24, MPC-24E, MPC-24EF, and the HI-STORM 100SA. General references to these documents are made in Chapter 10 as needed to supplement FSAR information.

TABLE 10.1-1

OPERATING CONTROLS AND LIMITS

Areas For Operating Controls and Limits	Conditions Or Other Items To Be Controlled
Fuel characteristics	Physical condition
Multi-Purpose Canister	Drying temperature Helium backfill pressure
Spent Fuel Storage Cask	Heat removal capability
Administrative Controls	Fuel loading verification including assembly location

TABLE 10.2-1

MPC-24 FUEL ASSEMBLY LIMITS

A. Allowable Contents

1. Uranium oxide, intact fuel assemblies listed in Table 10.2-5, with or without nonfuel hardware and meeting the following specifications (Note 1):

Cladding type	Zr (Note 2)
Initial enrichment	As specified in Table 10.2-5 for the applicable fuel assembly.
Post-irradiation cooling time and average burnup per assembly:	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware)

B. Quantity per MPC: Up to 24 fuel assemblies.

C. Damaged fuel assemblies and fuel debris are not authorized for loading into the MPC-24.

D. One NSA is authorized for loading in an MPC-24.

Note 1: Fuel assemblies containing BPRAs, WABAs, or TPDs may be stored in any fuel cell location. Fuel assemblies containing RCCAs or NSAs may only be loaded in fuel storage locations 9, 10, 15, and/or 16 of Figure 10.2-1. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Note 2: Zr designates fuel-cladding material, which is made of Zircaloy-2, Zircaloy-4 and ZIRLO.

TABLE 10.2-2

MPC-24E FUEL ASSEMBLY LIMITS

A. Allowable Contents

1. Uranium oxide, intact fuel assemblies listed in Table 10.2-5, with or without nonfuel hardware and meeting the following specifications (Note 1):

Cladding type	Zr (Note 2)
Initial enrichment	As specified in Table 10.2-5 for the applicable fuel assembly.
Post-irradiation cooling time and average burnup per assembly	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware)

2. Uranium oxide, damaged fuel assemblies, with or without nonfuel hardware, placed in damaged fuel containers. Uranium oxide damaged fuel assemblies shall meet the criteria specified in Table 10.2-5 and meet the following specifications (Note 1):

Cladding type	Zr (Note 2)
Initial enrichment	≤ 4.0 wt% ^{235}U.
Post-irradiation cooling time and average burnup per assembly:	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware and DFC)

TABLE 10.2-2 Sheet 2 of 2

B. Quantity per MPC: Up to four (4) damaged fuel assemblies in damaged fuel containers, stored in fuel storage locations 3, 6, 19 and/or 22 of Figure 10.2-2. The remaining MPC-24E fuel storage locations may be filled with intact fuel assemblies meeting the applicable specifications.

C. Fuel debris is not authorized for loading in the MPC-24E.

D. One NSA is authorized for loading in an MPC-24E.

Note 1: Fuel assemblies containing BPRAs, WABAs, or TPDs may be stored in any fuel storage location. Fuel assemblies containing RCCAs or NSAs must be loaded in fuel storage locations 9, 10, 15 and/or 16 of Figure 10.2-2. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Note 2: Zr designates fuel-cladding material, which is made of Zircaloy-2, Zircaloy-4 and ZIRLO.

TABLE 10.2-3 Sheet 1 of 2

MPC-24EF FUEL ASSEMBLY LIMITS

A. Allowable Contents

1. Uranium oxide, intact fuel assemblies listed in Table 10.2-5, with or without nonfuel hardware and meeting the following specifications (Note 1):

Cladding type	Zr (Note 3)
Initial enrichment	As specified in Table 10.2-5 for the applicable fuel assembly.
Post-irradiation cooling time and average burnup per assembly:	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware)

2. Uranium oxide, damaged fuel assemblies and fuel debris, with or without nonfuel hardware, placed in damaged fuel containers. Uranium oxide damaged fuel assemblies shall meet the criteria specified in Table 10.2-5 and meet the following specifications (Note 1 and 2):

Cladding type	Zr (Note 3)
Initial enrichment	≤ 4.0 wt% ^{235}U.
Post-irradiation cooling time and average burnup per assembly	
Fuel	As specified in Tables 10.2-6 or 10.2-8.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware and DFC)

B. Quantity per MPC: Up to four (4) damaged fuel assemblies and/or fuel debris in damaged fuel containers, stored in fuel storage locations 3, 6, 19 and/or 22 of Figure 10.2-2. The remaining MPC-24EF fuel storage locations may be filled with intact fuel assemblies meeting the applicable specifications.

C. One NSA is authorized for loading in an MPC-24EF.

Note 1: Fuel assemblies containing BPRAs, WABAs, or TPDs may be stored in any fuel storage location. Fuel assemblies containing RCCAs or NSAs must be loaded in fuel storage locations 9, 10, 15 and/or 16 of Figure 10.2-2. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Note 2: The total quantity of fuel debris permitted in a single damaged fuel container is limited to the equivalent weight and special nuclear material quantity of one intact fuel assembly.

Note 3: Zr designates fuel-cladding material, which is made of Zircaloy-2, Zircaloy-4, and ZIRLO.

TABLE 10.2-4

MPC-32 FUEL ASSEMBLY LIMITS

A. Allowable Contents

1. Uranium oxide, intact fuel assemblies listed in Table 10.2-5, with or without nonfuel hardware and meeting the following specifications (Note 1):

Cladding type	Zr (Note 2)
Initial enrichment	As specified in Table 10.2-5 for the applicable fuel assembly.
Post-irradiation cooling time and average burnup per assembly:	
Fuel	As specified in Tables 10.2-6 or 10.2-8, or Section 10.2.1.2.1.
Nonfuel hardware	As specified in Table 10.2-10.
Decay heat per assembly	As specified in Tables 10.2-7 or 10.2-9.
Fuel assembly length	≤ 176.8 inches (nominal design)
Fuel assembly width	≤ 8.54 inches (nominal design)
Fuel assembly weight	≤ 1,680 lb (including nonfuel hardware)

B. Quantity per MPC: Up to 32 intact fuel assemblies.

C. Damaged fuel assemblies and fuel debris are not authorized for loading in the MPC-32.

D. One NSA is authorized for loading in an MPC-32.

Note 1: Fuel assemblies containing BPRAs, WABAs, or TPDs with or without ITTRs, may be stored in any fuel storage location. Fuel assemblies, with or without ITTRs, containing RCCAs or NSAs must be loaded in fuel storage locations 13, 14, 19 and/or 20 of Figure 10.2-3. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Note 2: Zr designates fuel-cladding material, which is made of Zircaloy-2, Zircaloy-4 and ZIRLO.

TABLE 10.2-5

FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Type (Note 6)	Vantage 5	Standard or LOPAR
Cladding Material	Zr (Note 5)	Zr (Note 5)
Design Initial U (kg/assy.) (Note 2)	≤ 467	≤ 467
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt% ^{235}U) (Note 4)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, or 32 with soluble boron credit) (wt% ^{235}U) (Notes 3 and 4)	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	264	264
Fuel Rod Cladding O.D. (in.)	≥ 0.360	≥ 0.372
Fuel Rod Cladding I.D. (in.)	≤ 0.3150	≤ 0.3310
Fuel Pellet Dia. (in.)	≤ 0.3088	≤ 0.3232
Fuel Rod Pitch (in.)	≤ 0.496	≤ 0.496
Active Fuel Length (in.)	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.016	≥ 0.014

Note 1: All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies.

Note 2: Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or DCPP. For each fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with DCPP fuel records to account for manufacturers tolerances.

Note 3: Soluble boron concentration per Technical Specification LCO 3.2.1.

Note 4: For those MPCs loaded with both intact fuel assemblies and damaged fuel assemblies or fuel debris, the maximum initial enrichment of the intact fuel assemblies is limited to the maximum initial enrichment of the damaged fuel assemblies and fuel debris (i.e., 4.0 wt.% ^{235}U).

Note 5: Zr designates fuel-cladding material, which is made of Zircaloy-2, Zircaloy-4 and ZIRLO.

NOTE 6: Fuel assemblies meeting the characteristics may be loaded under the requirements for the listed Fuel Assembly Type, even if the name is different.

TABLE 10.2-6

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP
(UNIFORM FUEL LOADING)

Post-Irradiation Cooling Time (years)	MPC-24 Assembly Burnup (Intact Fuel Assemblies) (MWD/MTU)	MPC-24E/24EF Assembly Burnup (Intact Fuel Assemblies) (MWD/MTU)	MPC-24E/24EF Assembly Burnup (Damaged Fuel Assemblies and Fuel Debris) (MWD/MTU)	MPC-32 Assembly Burnup (Intact Fuel Assemblies) (MWD/MTU) (Note 2)
≥ 5	40,600	41,100	39,200	32,200
≥ 6	45,000	45,000	43,700	36,500
≥ 7	-	-	44,500	37,500
≥ 8	-	-	-	39,900
≥ 9	-	-	-	41,500
≥ 10	-	-	-	42,900
≥ 11	-	-	-	44,100
≥ 12	-	-	-	45,000

Note 1: Linear interpolation between points is permitted.

Note 2: Burnup limits for fuel assemblies in an MPC-32 may alternatively be calculated using 10.2.1.2.1.

TABLE 10.2-7

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT
(UNIFORM FUEL LOADING)

Post-Irradiation Cooling Time (years)	MPC-24 Assembly Decay Heat (Intact Fuel Assemblies) (Watts)	MPC-24E/24EF Assembly Decay Heat (Intact Fuel Assemblies) (Watts)	MPC-24E/24EF Assembly Decay Heat (Damaged Fuel Assemblies and Fuel Debris) (Watts)	MPC-32 Assembly Decay Heat (Intact Fuel Assemblies) (BU ≤ 45,000 MWd/MTU) (Watts)	MPC-32 Assembly Decay Heat (Intact Fuel Assemblies) (BU > 45,000 MWd/MTU) (Watts)
≥ 5	1157	1173	1115	898	898
≥ 6	1123	1138	1081	898	898
≥ 7	1030	1043	991	898	898
≥ 8	1020	1033	981	898	898
≥ 9	1010	1023	972	898	898
≥ 10	1000	1012	962	898	898
≥ 11	996	1008	958	898	898
≥ 12	992	1004	954	898	898
≥ 13	987	999	949	898	898
≥ 14	983	995	945	898	898
≥ 15	979	991	941	898	898

Note 1: Linear interpolation between points is permitted.

Note 2: Includes all sources of heat (i.e., fuel and nonfuel hardware).

TABLE 10.2-8

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP
(REGIONALIZED FUEL LOADING)

Post-Irradiation Cooling Time (years)	MPC-24 Assembly Burnup for Region 1 (MWD/MTU)	MPC-24 Assembly Burnup for Region 2 (MWD/MTU)	MPC-24E/24EF Assembly Burnup for Region 1 (MWD/MTU)	MPC-24E/24EF Assembly Burnup for Region 2 (MWD/MTU)	MPC-32 Assembly Burnup for Region 1 (MWD/MTU) (Note3)	MPC-32 Assembly Burnup for Region 2 (MWD/MTU) (Note 3)
≥ 5	45,000	32,200	45,000	32,200	39,800	22,100
≥ 6	-	37,400	-	37,400	43,400	26,200
≥ 7	-	41,100	-	41,100	44,500	29,100
≥ 8	-	43,800	-	43,800	45,000	31,200
≥ 9	-	45,000	-	45,000	-	32,700
≥ 10	-	-	-	-	-	34,100
≥ 11	-	-	-	-	-	35,200
≥ 12	-	-	-	-	-	36,200
≥ 13	-	-	-	-	-	37,000
≥ 14	-	-	-	-	-	37,800
≥ 15	-	-	-	-	-	38,600
≥ 16	-	-	-	-	-	39,400
≥ 17	-	-	-	-	-	40,200
≥ 18	-	-	-	-	-	40,800
≥ 19	-	-	-	-	-	41,500
≥ 20	-	-	-	-	-	42,200

Note 1: Linear interpolation between points is permitted.
Note 2: These limits apply to intact fuel assemblies, damaged fuel assemblies, and fuel debris.
Note 3: Burnup limits for fuel assemblies in an MPC-32 may alternatively be calculated per 10.2.1.2.1.

TABLE 10.2-9

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT
(REGIONALIZED FUEL LOADING)

Post-Irradiation Cooling Time (years)	MPC-24 Assembly Decay Heat for Region 1 (Watts)	MPC-24 Assembly Decay Heat for Region 2 (Watts)	MPC-24E/24EF Assembly Decay Heat for Region 1 (Watts)	MPC-24E/24EF Assembly Decay Heat for Region 2 (Watts)	MPC-32 Assembly Decay Heat for Region 1 (Watts)	MPC-32 Assembly Decay Heat for Region 2 (Watts)
≥ 5	1470	900	1540	900	1131	600
≥ 6	1470	900	1540	900	1131	600
≥ 7	1335	900	1395	900	1131	600
≥ 8	1301	900	1360	900	1131	600
≥ 9	1268	900	1325	900	1131	600
≥ 10	1235	900	1290	900	1131	600
≥ 11	1221	900	1275	900	1131	600
≥ 12	1207	900	1260	900	1131	600
≥ 13	1193	900	1245	900	1131	600
≥ 14	1179	900	1230	900	1131	600
≥ 15	1165	900	1215	900	1131	600

Note 1: Linear interpolation between points is permitted.
Note 2: Includes all sources of decay heat (i.e., fuel and nonfuel hardware).
Note 3: These limits apply to intact fuel assemblies, damaged fuel assemblies, and fuel debris.

TABLE 10.2-10

NONFUEL HARDWARE COOLING AND AVERAGE ACTIVATION

Post-Irradiation Cooling Time (years)	BPRA and WABA Burnup (MWD/MTU)	TPD and NSA Burnup (MWD/MTU)	RCCA Burnup (MWD/MTU)
≥3	≤20,000	Not Authorized	Not Authorized
≥4	≤25,000	≤20,000	Not Authorized
≥5	≤30,000	≤25,000	≤630,000
≥6	≤40,000	≤30,000	
≥7	≤45,000	≤40,000	
≥8	≤50,000	≤45,000	
≥9	≤60,000	≤50,000	
≥10		≤60,000	
≥11		≤75,000	
≥12		≤90,000	
≥13		≤180,000	
≥14		≤630,000	

Note 1: Linear interpolation between points is permitted, except that TPD and NSA burnups > 180,000 MWD/MTU and ≤630,000 MWD/MTU must be cooled ≥ 14 years.

Note 2: Applicable to uniform loading and regionalized loading.

Note 3: Deleted.

Note 4: Non-fuel hardware burnup and cooling times are not applicable to ITTRs because they are installed post-irradiation.

Note 5: Only one NSA is authorized for loading in any MPC.

TABLE 10.2-11 Sheet 1 of 2

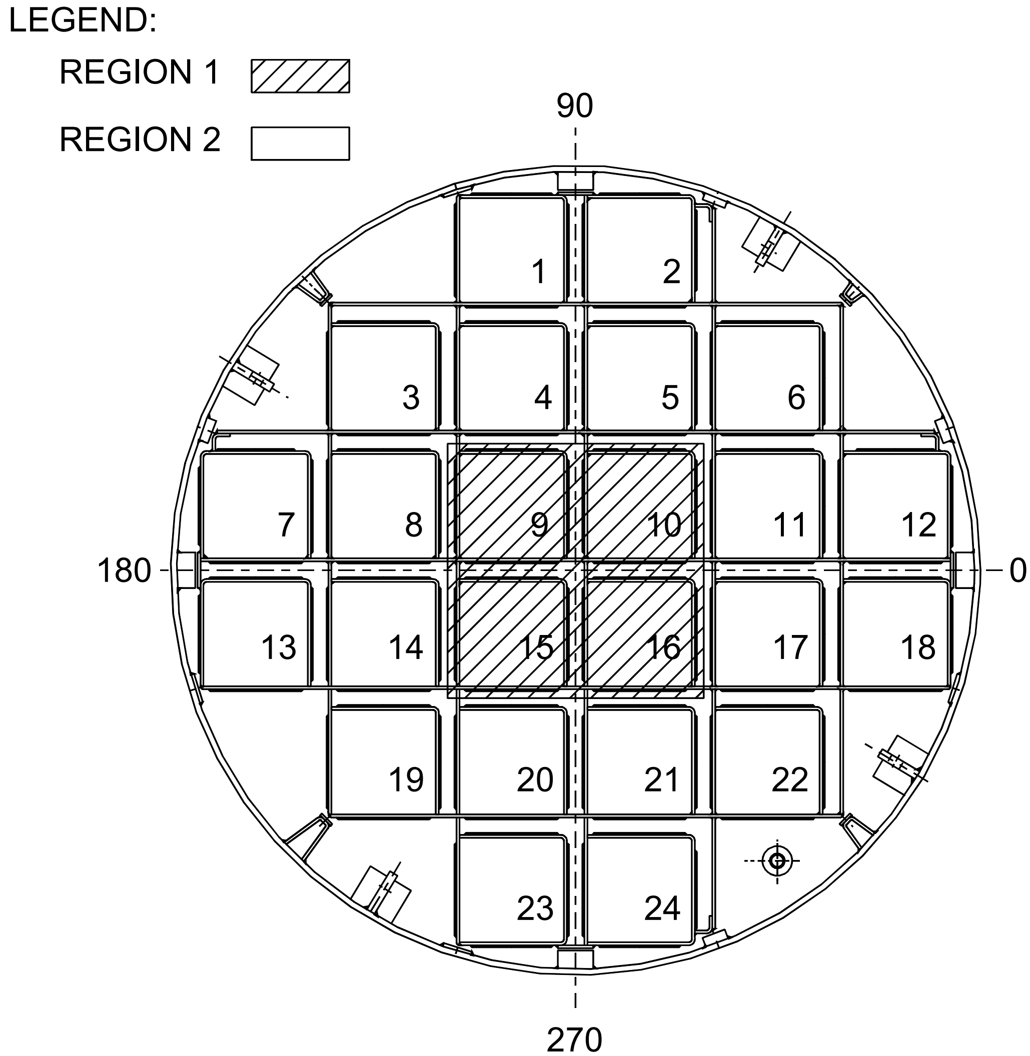
FUEL ASSEMBLY TIME-DEPENDENT COEFFICIENTS

Cooling Time (years)	Vantage 5 fuel						
	A	B	C	D	E	F	G
≥ 5	40315.9	-9724	1622.89	-140.459	3170.28	-547.749	425.136
≥ 6	49378.5	-15653.1	3029.25	-164.712	3532.55	-628.93	842.73
≥ 7	56759.5	-21320.4	4598.78	-190.58	3873.21	-698.143	975.46
≥ 8	63153.4	-26463.8	6102.47	-201.262	4021.84	-685.431	848.497
≥ 9	67874.9	-30519.2	7442.84	-218.184	4287.23	-754.597	723.305
≥ 10	72676.8	-34855.2	8928.27	-222.423	4382.07	-741.243	387.877
≥ 11	75623	-37457.1	9927.65	-232.962	4564.55	-792.051	388.402
≥ 12	80141.8	-41736.5	11509.8	-232.944	4624.72	-787.134	-164.727
≥ 13	83587.5	-45016.4	12800.9	-230.643	4623.2	-745.177	-428.635
≥ 14	86311.3	-47443.4	13815.2	-228.162	4638.89	-729.425	-561.758
≥ 15	87839.2	-48704.1	14500.3	-231.979	4747.67	-775.801	-441.959
≥ 16	91190.5	-51877.4	15813.2	-225.768	4692.45	-719.311	-756.537
≥ 17	94512	-55201.2	17306.1	-224.328	4740.86	-747.11	-1129.15
≥ 18	96959	-57459.9	18403.8	-220.038	4721.02	-726.928	-1272.47
≥ 19	99061.1	-59172.1	19253.1	-214.045	4663.37	-679.362	-1309.88
≥ 20	100305	-59997.5	19841.1	-216.112	4721.71	-705.463	-1148.45

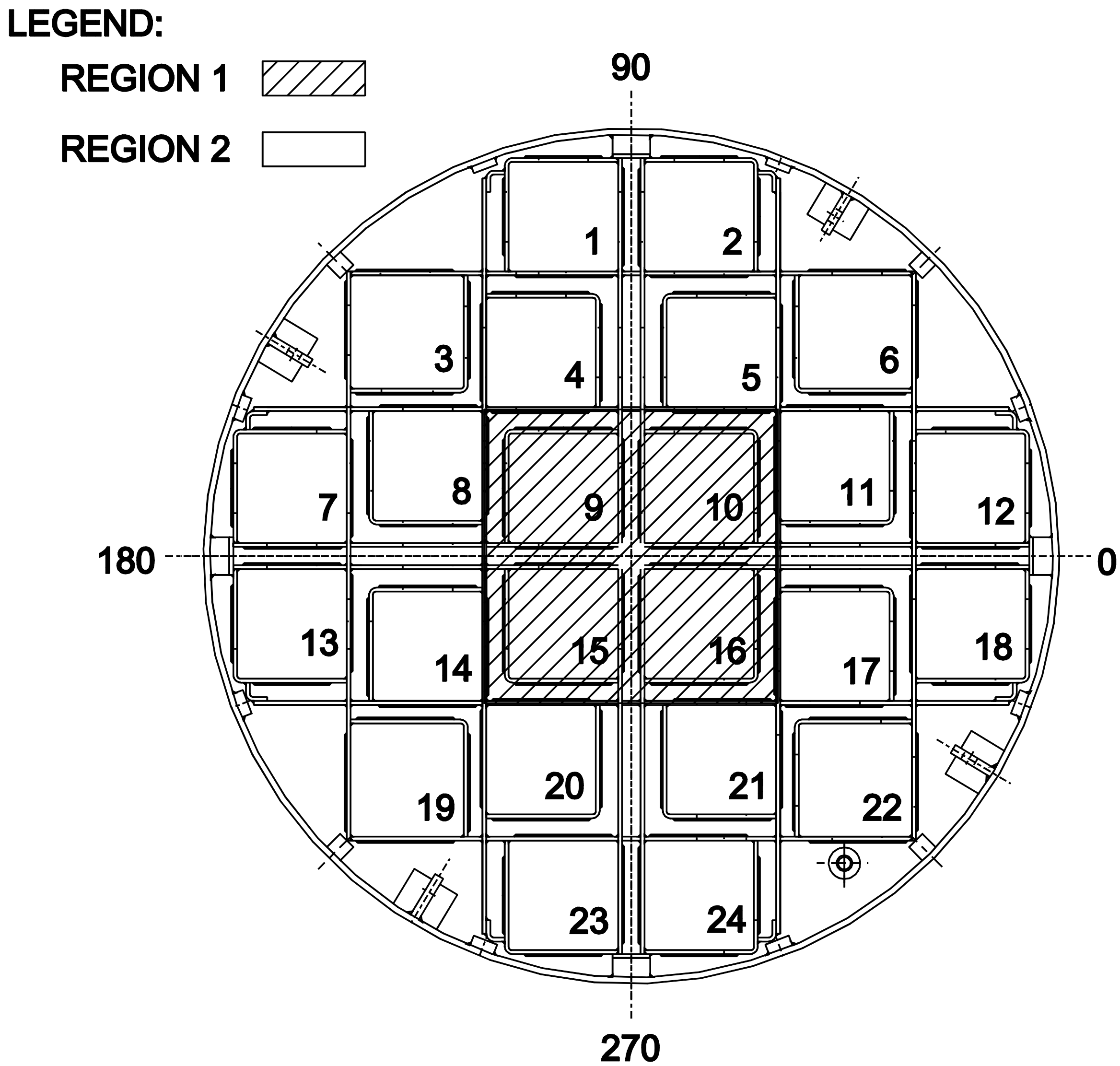
TABLE 10.2-11 Sheet 2 of 2

FUEL ASSEMBLY TIME-DEPENDENT COEFFICIENTS

Cooling Time (years)	Standard or LOPAR fuel						
	A	B	C	D	E	F	G
≥ 5	36190.4	-7783.2	1186.37	-130.008	2769.53	-438.716	519.95
≥ 6	44159	-12517.5	2209.54	-150.234	3042.25	-489.858	924.151
≥ 7	50399.6	-16780.6	3277.26	-173.223	3336.58	-555.743	1129.66
≥ 8	55453.9	-20420	4259.68	-189.355	3531.65	-581.917	1105.62
≥ 9	59469.3	-23459.8	5176.62	-199.63	3709.99	-626.667	1028.74
≥ 10	63200.5	-26319.6	6047.8	-203.233	3783.02	-619.949	805.311
≥ 11	65636.3	-28258.3	6757.23	-214.247	3972.8	-688.56	843.457
≥ 12	68989.7	-30904.4	7626.53	-212.539	3995.62	-678.037	495.032
≥ 13	71616.6	-32962.2	8360.45	-210.386	4009.11	-666.542	317.009
≥ 14	73923.9	-34748	9037.75	-207.668	4020.13	-662.692	183.086
≥ 15	76131.8	-36422.3	9692.32	-203.428	4014.55	-655.981	47.5234
≥ 16	77376.5	-37224.7	10111.4	-207.581	4110.76	-703.37	161.128
≥ 17	80294.9	-39675.9	11065.9	-201.194	4079.24	-691.636	-173.782
≥ 18	82219.8	-41064.8	11672.1	-195.431	4043.83	-675.432	-286.059
≥ 19	84168.9	-42503.6	12309.4	-190.602	4008.19	-656.192	-372.411
≥ 20	86074.2	-43854.4	12935.9	-185.767	3985.57	-656.72	-475.953



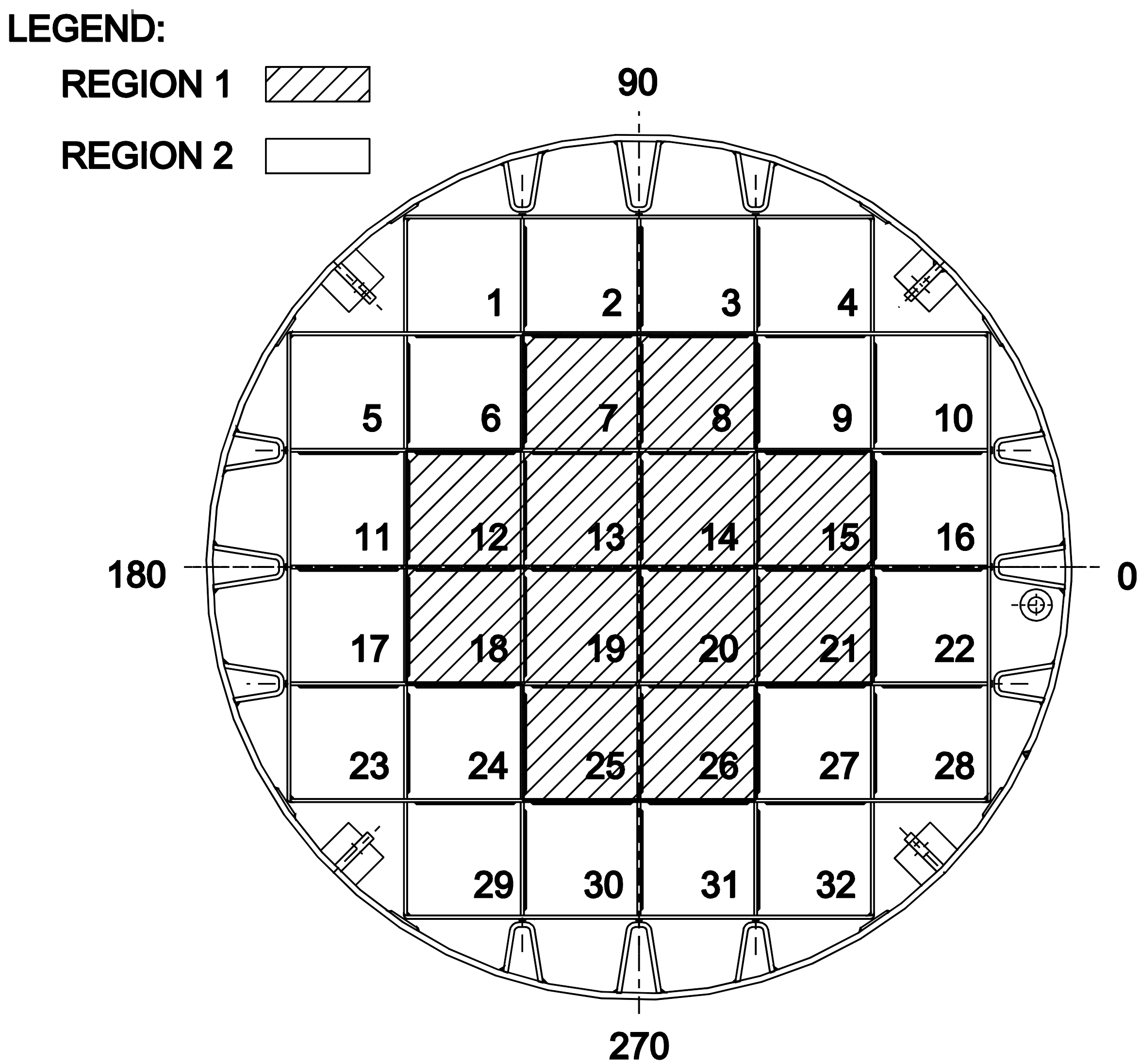
FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 10.2-1 FUEL LOADING REGIONS MPC-24



FSAR UPDATE

DIABLO CANYON ISFSI

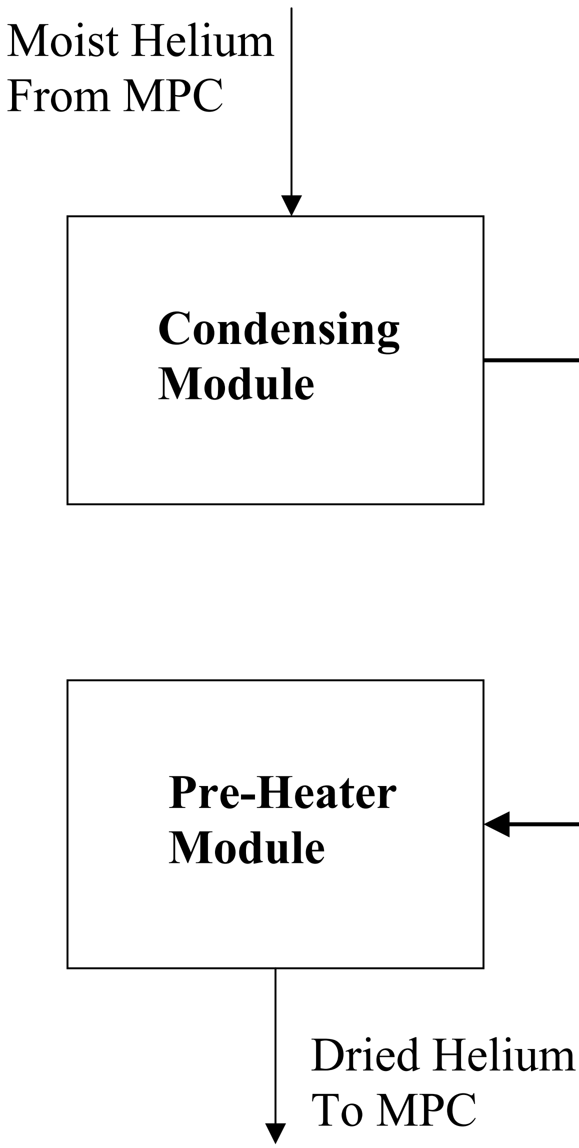
FIGURE 10.2-2
FUEL LOADING REGIONS
MPC-24E/24EF



FSAR UPDATE

DIABLO CANYON ISFSI

FIGURE 10.2-3
FUEL LOADING REGIONS
MPC-32



FSAR UPDATE
DIABLO CANYON ISFSI
FIGURE 10.2-4 SCHEMATIC DIAGRAM OF THE FORCED HELIUM DEHYDRATION SYSTEM

CHAPTER 11

QUALITY ASSURANCE

CONTENTS

CHAPTER 11

QUALITY ASSURANCE

10 CFR 72.140(b) states that each licensee shall establish, maintain, and execute a quality assurance (QA) program satisfying each of the applicable criteria of Subpart G. Paragraph (d) of 10 CFR 72.140 states that a Commission-approved QA program that satisfies the applicable criteria of Appendix B of Part 50 and which is established, maintained, and executed with regard to an ISFSI will be accepted as satisfying the requirements of 10 CFR 72.140(b).

Since PG&E is currently licensed under 10 CFR 50 to operate the Diablo Canyon Power Plant (DCPP), Units 1 and 2, a Commission-approved QA program meeting the requirements of 10 CFR 50, Appendix B, is already in place. The governing document for this program is the DCPP QA Program as described in the DCPP Final Safety Analysis Report (FSAR) Update, Chapter 17 (Reference Docket No. 50-275, OL-DPR-80 and Docket No. 50-323, OL-DPR-82). This QA Program was first submitted as part of the original DCPP FSAR in 1973; was approved by the Commission for use in NRC Supplemental Safety Evaluation Report No. 9, issued in June 1980; and is updated in accordance with 10 CFR 50.54(a). The NRC is periodically notified of changes to the document as required by 10 CFR 50.71.

This QA Program applies to the design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, modification, and decommissioning of ISFSI structures, systems, and components that are important to safety. Section 4.5 identifies systems and components that are important to safety. The program also applies to managerial and administrative controls used to ensure safe ISFSI operation.

QA Program implementation is accomplished through separately issued instructions, procedures, and drawings. The objective of the QA Program for the ISFSI is to comply with the criteria established in 10 CFR 50, Appendix B, as amended, and with applicable QA Program requirements for nuclear power plants as referenced in regulatory guides and ANSI standards. The applicable guides and standards are identified in the DCPP FSAR Update, Table 17.1-1.

The procurement documents are reviewed prior to approval to ensure that the proper criteria have been specified. During the ISFSI design phase, vendor information (drawings, specifications, procedures, etc.) is reviewed to ensure compliance with DCPP's technical and quality requirements. During design, licensing, and fabrication of the cask storage system, PG&E's vendor surveillance representative visits the suppliers' and fabricators' facilities to ensure compliance with PG&E's requirements.

Vendors and contractors that provide important-to-safety items and services work to a PG&E-approved QA program that meets the requirements of 10 CFR 72.140.

APPENDIX A

MATERIALS

NRC Interim Staff Guidance (ISG-15), dated January 10, 2001, provides specific guidance for the review of materials selected for dry cask storage systems. Regulatory requirements and review acceptance criteria are presented in Sections X.3 and X.4, respectively, of ISG-15. ISG-11, Revision 3, has subsequently been implemented by PG&E, which superseded ISG-15 guidance on cladding integrity, Sections X.4.4 and X.5.4. While there are a large number of requirements and criteria presented, they can be grouped into ten major categories, as follows:

(1) Adequate Description – Structures, systems and components (SSCs) that are important to safety and the materials from which they are constructed must be described in sufficient detail to permit adequate review (ISG-15, Sections X.3.1.a, X.3.2.d, and X.4.1).

(2) Quality Standards – SSCs important to safety must be designed, built and tested to quality standards adequate for the safety function performed by the SSC (ISG-15, Section X.3.2.a).

(3) Design Life – The cask design and the materials from which it is constructed must be designed to safely store spent fuel and permit required maintenance for the entire 20-year license period (ISG-15, Sections X.3.2.e and X.4.2).

(4) Environmental Capability – The cask design and materials from which it is constructed (including coatings) must be compatible with all expected environmental conditions, including wet and dry loading and unloading facilities. Adverse chemical or corrosion reactions that would impact safe operation must be avoided (ISG-15, Sections X.3.1.b, X.3.2.c, X.3.3, and X.4.1 through X.4.3).

(5) Cladding Integrity – Spent fuel cladding must be protected, under both normal, off-normal, and accident conditions, from temperatures and environments that could cause degradation leading to cladding rupture (ISG-15, Section X.3.4.a).

(6) Fire Protection – Noncombustible and heat resistant materials shall be used wherever possible (ISG-15, Sections X.3.2.f and X.4.3).

(7) Nuclear Control – Materials used for shielding and criticality functions must be appropriately selected to perform the function adequately and without susceptibility to slumping or other loss of effectiveness (ISG-15, Sections X.3.2.b and X.4.2).

(8) Confinement Boundary – Confinement of radioactive materials must be maintained under all normal, off-normal, and accident conditions (ISG-15, Section X.3.2.g).

(9) Offsite Shipment – The cask system must be designed to allow spent fuel to be transported off-site for eventual delivery to a DOE repository (ISG-15, Section X.3.1.a).

(10) Operating Conditions – Materials used to construct the cask system must maintain acceptable physical and mechanical properties over all operating conditions, including temperature extremes (ISG-15, Section X.4.2).

Each of these ten categories derived from ISG-15 has been evaluated for the dry cask storage system and is discussed below.

Adequate Description

This category requires that those components of the cask system that are important to safety are identified appropriately and that complete and accurate descriptions of those components be provided. Section 4.5 of this FSAR identifies equipment and components that are designated as important to safety. FSAR Chapters 3 and 4 provide descriptions of the identified important to safety components and equipment.

Quality Standards

This category requires ensuring that appropriate governing codes be selected for SSCs important to safety. FSAR Tables 3.4-1 through 3.4-5 provide the principal design criteria for the SSCs important to safety.

Design Life

This category requires that the design life of the cask system be specified and be at least 20 years in duration. The design life of the cask system is 40 years, as specified in FSAR Table 3.4-2.

Environmental Capability

This category requires that reactions between cask system materials and the environment be avoided, including reactions with the spent fuel pool water and corrosion reactions. The MPC is made entirely of stainless steel, except for the neutron absorbers, and an aluminum seal washer or port plug with thread protector in both the vent and drain port assemblies. An alternative vent and drain port plug configuration may be used, which does not contain aluminum washers. The Boral is passivated prior to use, and any continuing passivation reactions will not result in significant hydrogen production. There are no coatings of any kind in the MPC. The transfer cask is constructed from the following materials: carbon steels; elemental lead; Holtite-A neutron shield material; paint; and brass, bronze or stainless-steel appurtenances (pressure relief valves, drain tube, etc.). Exposed surfaces of the transfer cask are coated with an epoxy-based coating material that has been demonstrated not to react with the borated spent fuel pool water. The storage cask, its anchorage, and the cask transfer facility are constructed of carbon steels and concrete, with exposed surfaces coated for exterior service. The dry cask storage system is designed for marine environment service, including current-induced electromagnetic fields.

Cladding Integrity

This category requires that appropriate fuel cladding temperature limits be determined and met and that the fuel cladding be protected from exposure to reacting environments. Section 10.2 of this FSAR describes the determination of allowable fuel cladding types and temperature limits and provides values for the limits. The normal condition limits ensure a probability of cladding breach of less than 0.5 percent over the 40-year design life and the short-term accident cladding temperature limit is in accordance with NRC guidance. Section 10.2 of this FSAR describes that the MPC cavity is backfilled with helium, an inert gas, eliminating any reacting environment within the canister.

Fire Protection

This category requires using only materials that will not ignite when exposed to heat or flame. The MPC is made entirely of stainless steel, except for the neutron absorbers, and an aluminum seal washer or port plug with thread protector in both the vent and drain port assemblies. An alternative vent and drain port plug configuration may be used, which does not contain aluminum washers. The transfer cask is constructed from the following materials: carbon steels; elemental lead; Holtite-A neutron shield material; paint; and brass, bronze or stainless-steel appurtenances (pressure relief valves, drain tube, etc.). The storage cask, its anchorage, and the cask transfer facility are constructed of carbon steels and concrete, with exposed surfaces coated for exterior service. None of these materials are known to ignite when exposed to heat or flame.

Nuclear Control

This category requires the use of materials with known radiation shielding and criticality control performance. Materials used for criticality control in the MPC are the Boral panels affixed to the walls of the fuel cells. Boral has been used successfully for many years in wet storage applications and, more recently, in dry storage service in the nuclear industry. Shielding in the transfer cask is provided primarily by lead, steel and water; also commonly used in nuclear applications. A small amount of Holtite-A neutron shield material is used in the lids of the transfer cask. A detailed description of Holtite-A may be found in Section 1.2.1.3.2 of the HI-STORM 100 System FSAR.

Confinement Boundary

This category requires demonstrating that the MPC confinement boundary stresses and temperatures are not exceeded. The structural and thermal analyses discussed elsewhere in this FSAR provide this information.

Offsite Shipment

This category requires that the cask system or, in the case of canister-based systems, the MPC be designed for transportation. The MPC is designed for transportation under 10 CFR 71 in the Holtec HI-STAR 100 transport cask.

Operating Conditions

This category requires that all materials must be evaluated under all conditions that are reasonably expected to occur during the design life of the cask system. The structural, thermal, criticality and shielding calculations presented in this FSAR have evaluated the performance of the cask system materials under bounding conditions of storage and onsite handling including temperature extremes, drops and tipover, tornados, floods, seismic events, lightning, and explosions. All such evaluations have demonstrated the continued performance of the cask system materials.