ATTACHMENT C

Comparison between NRC Standard Guidance for Cask Storage Facilities, Holtec Technical Specification, and Diablo Canyon ISFSI Technical Specification

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Proposed Technical Specifications for Diablo Canyon Independent Spent Fuel Storage Installation

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Technical Specification Bases for Diablo Canyon Independent Spent Fuel Storage Installation

Comparison Between NRC Standard Guidance for Cask Storage Facilities, Holtec Technical Specification, and Diablo Canyon ISFSI Technical Specification

NRC Standard ISFSI Technical Specifications[1]	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
1.0 Use and Application	1.0 Use and Application	1.0 Use and Application	There is no difference between the NRC Std and the DCPP TS.
1.1 Definitions	1.1 Definitions	1.1 Definitions	There is no difference between the NRC Std and the DCPP TS.
Actions		Actions	There is no differences with NRC Std, however Holtec does not provide this definition.
	Cask Transfer Facility (CTF)	Cask Transfer Facility (CTF)	DCPP has a CTF and is providing a definition to bound that facility. This is similar to Holtec CofC. NRC Std allows for other design system definitions.
Canister	Multi-Purpose Canister (MPC)	Multi-Purpose Canister (MPC)	There is no technical difference between DCPP and NRC Std, however, DCPP uses MPC in place of the word canister. This is derived from the Holtec system terms.
	Damaged Fuel Assembly	Damaged Fuel Assembly	This is an additional definition describing the limitation for what is considered damaged fuel. NRC Std allows for other design system definitions.
	Damaged Fuel Container (DFC)	Damaged Fuel Container (DFC)	This is a definition of a type of container that is allowed to be used for storage of damaged fuel assemblies and debris in the DCPP storage system. NRC Std allows for other design system definitions.
	Fuel Debris	Fuel Debris	This definition provides the limitation for what is considered fuel debris. NRC Std allows for other design system definitions.

(1) NRC Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance dated 5/31/2001

Comparison Between NRC Standard Guidance for Cask Storage Facilities, Holtec Technical Specification, and Diablo Canyon ISFSI Technical Specification

NRC Standard ISFSI Technical Specifications[1]	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
Intact Fuel Assembly	Intact Fuel Assembly	Intact Fuel Assembly	There is no difference between the NRC Std and the DCPP TS definition.
Loading Operations	Loading Operations	Loading Operations	The NRC Std includes activities on the Overpack in this definition to cover those facilities that load canisters directly into the Overpacks. DCPP system loads into a Transfer Cask, then after transportation to the ISFSI facility, the MPC is transferred to an Overpack as part of the Transport Operations. In addition, the DCPP TS replaces the words "fuel assemblies" with "its approved contents." This provides for the loading of other entities such as damaged fuel rods, fuel debris and non-fuel hardware that have been authorized for loading in the DCPP cask system.
	NonFuel Hardware	NonFuel Hardware	This definition provides the limitation for what is considered nonfuel hardware. NRC Std allows for other design system definitions.
Operable/Operability		Operable/Operabilit y	There is no difference between the NRC Std and the DCPP TS definition.
Overpack	Overpack	Overpack	There is no technical difference between the definitions; however, DCPP uses MPC in place of canister.
	Planar-Average Initial Enrichment		Specific definition to Holtec submittal. Not used in the DCPP TS
Spent Fuel Storage Casks (SFSCs)	Spent Fuel Storage Casks (SFSCs)	Spent Fuel Storage Casks (SFSCs)	The DCPP SFSCs definition has been enhanced to include not only the storage of spent fuel assemblies, but fuel debris and nonfuel hardware, which is included in the approved contents for this facility. In

NRC Standard ISFSI Technical Specifications[1]	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
			addition, MPC is used in place of canister and it specifically call out the Holtec HI-STORM 100 SFSC System.
Spent Nuclear Fuel		Spent Nuclear Fuel	There is no difference between the NRC Std and the DCPP TS definition.
Storage Operations		Storage Operations	There is no technical difference., however, the DCPP definition is specific to define that this operation begins after the SFSC is in place on the storage pad and bolted down. Up until that moment the movement of the SFSC is part of the Transport Operation. This reflexes the seismic requirements of the DCPP facility. Also DCPP uses MPC in place of Canister.
Transfer Cask	Transfer Cask	Transfer Cask	This definition has been revised to use the DCPP term "MPC" in place of "canister". In addition, the definition has been revised to reflect the ability to store other entities such as fuel debris and nonfuel hardware in their cask system. As such, "spent fuel assemblies" has been replaced by "its approved contents".
Transport operations	Transport operations	Transport operations	This definition is similar, however, it is revised to reflect the DCPP cask system's ability to store other then just spent fuel assemblies. In addition, the word "canister" has been replaced by "MPC".
Unloading Operations	Unloading Operations	Unloading Operations	Technically there is no difference between these definitions, however, the definition has been made more specific where as the NRC definition is more general. In specific, the MPC that is to be unloaded

Comparison Between NRC Standard Guidance for Cask Storage Facilities, Holtec Technical Specification, and Diablo Canyon ISFSI Technical Specification			
NRC Standard ISFSI Technical Specifications[1]	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
			is transferred out of the Overpack (SFSC) and into a Transfer Cask at the Cask Transfer Facility. Then the Transfer Cask is transported to the DCPP Fuel Building. All of these activities are defined as Transport Operations by the TS. Once in the Fuel Building the Transfer Cask is taken off the transporter and then the enclosed MPC is unloaded of contents. At the point that the Transfer Cask is taken off of the transporter the Unloading Operations begin as defined in the TS and it ends when the approved contents of the MPC are completely removed. The DCPP definition reflects the use of the terms "MPC" in place of "canister" and "approved contents" in place of "fuel assemblies".
1.2 Logic Connectors	1.2 Logic Connectors	1.2 Logic Connectors	There is no difference between the NRC Std and the DCPP TS.
1.3 Completion Times	1.3 Completion Times	1.3 Completion Times	There is no difference between the NRC Std and the DCPP TS.
1.4 Frequency	1.4 Frequency	1.4 Frequency	There is no difference between the NRC Std and the DCPP TS.
2.0 Approved Contents		2.0 Approved Contents	There is no difference between the NRC Std and the DCPP TS.
2.1		2.1	The statement has been modified to reflect the other entities that are approved for storage in the DCPP ISFSI., such as fuel debris and nonfuel hardware. In addition, the contents list is specified in FSAR Section 10.2 not an Appendix as in the NRC Std.

Comparison Between NRC Standard Guidance for Cask Storage Facilities, Holtec Technical Specification, and Diablo Canyon ISFSI Technical Specification

NRC Standard ISFSI Technical Specifications[1]	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
2.2		2.2	There is no technical difference, however, the list of contents is in FSAR Section 10.2 not an Appendix as in the NRC Std.
3.0 Limiting Condition for Operation (LCO) Applicability	3.0 Limiting Condition for Operation (LCO) Applicability	3.0 Limiting Condition for Operation (LCO) Applicability	There is no difference between the NRC Std and the DCPP TS.
LCO 3.0.1		LCO 3.0.1	There is no difference between the NRC Std and the DCPP TS.
LCO 3.0.2		LCO 3.0.2	There is no difference between the NRC Std and the DCPP TS.
LCO 3.0.3		LCO 3.0.3	There is no difference between the NRC Std and the DCPP TS.
LCO 3.0.4		LCO 3.0.4	There is no difference between the NRC Std and the DCPP TS.
LCO 3.0.5		LCO 3.0.5	There is no difference between the NRC Std and the DCPP TS.
SR 3.0.1		SR 3.0.1	There is no difference between the NRC Std and the DCPP TS.
SR 3.0.2		SR 3.0.2	There is no difference between the NRC Std and the DCPP TS.
SR 3.0.3		SR 3.0.3	There is no difference between the NRC Std and the DCPP TS.
SR 3.0.4		SR 3.0.4	There is no difference between the NRC Std and the DCPP TS.
3.1 Fuel Integrity	3.1 SFSC Integrity	3.1 Fuel Integrity	There is no difference between the NRC Std and the DCPP TS.

Comparison Between NRC Standard Guidance for Cask Storage Facilities, Holtec Technical Specification, and Diablo Canyon ISFSI Technical Specification

NRC Standard ISFSI Technical Specifications(1)	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
3.1.1 Fuel Integrity During Drying			This LCO is not included in the DCPP TS because there is not a time-based concern with the Holtec system. The NRC Std provides an LCO that is based on limiting time that a cask can be with out water or an inert atmosphere. This is not an issue for DCPP cask system and as long as water is maintained in the annular gap of the Transfer Cask there is no time limitation for the allowable contents in the MPC during the drying process. The DCPP ISFSI Cask Loading, Unloading and Preparation Program provided in Section 5.1.4 controls the critical parameters including the water in the annular gap, and ensure that the design bases conditions are continually met.
LCO 3.1.1			N/A
SR 3.1.1			N/A
	3.1.1 Multi-purpose Canister (MPC)		This LCO was provided as part of the Holtec submittal. It deals with the specific vacuum, dryness and helium backfill requirements for the MPCs. This is not included as an LCO in the NRC Std and is controlled as a part of TS program 5.1.2 Cask Loading, Unloading, and Preparation Program. DCPP is also handling this under a similar program.
	LCO 3.1.1		N/A
	SR 3.1.1.1		N/A
	SR 3.1.1.2		N/A
	SR 3.1.1.3		N/A

NRC Standard ISFSI Technical Specifications[1]	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
3.1.2 Fuel Integrity During Backfill and Transfer			This LCO is not included in the DCPP TS because there is not a concern with the Holtec system. The NRC Std provides an LCO that is based on limiting time that a cask can be with out water or an inert atmosphere. This is not an issue for DCPP cask system and as long as water is maintained in the annular gap of the Transfer Cask there is no time limitation for the allowable contents. The DCPP ISFSI Cask Loading, Unloading and Preparation Program provided in Section 5.1.4 controls the critical parameters including the water in the annular gap, and ensure that the design bases conditions are continually met.
LCO 3.1.2			N/A
SR 3.1.2			N/A
	3.2 SFSC Radiation Protection		The Holtec submittal includes this, however, it is provided for in both the NRC Std and the DCPP TS as part of administrative programs for radioactive effluent control and radiation protection programs.
	3.2.1 Transfer Cask Average Surface Dose Rates		The Holtec submittal includes this, however, it is provided for in both the NRC Std and the DCPP TS as part of administrative programs for radioactive effluent control and radiation protection programs.
	LCO 3.2.1		N/A
	SR 3.2.1.1		N/A

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NRC Standard ISFSI Technical Specifications[1]	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
	3.2.2 Transfer Cask Surface Contamination		The Holtec submittal includes this, however, it is provided for in both the NRC Std and the DCPP TS as part of administrative programs for radioactive effluent control and radiation protection programs.
	LCO 3.2.2		N/A
	SR 3.2.2.1		N/A
	3.2.3 Overpack Average Surface Dose Rates		The Holtec submittal includes this, however, it is provided for in both the NRC Std and the DCPP TS as part of administrative programs for radioactive effluent control and radiation protection programs.
	LCO 3.2.3		N/A
	SR 3.2.3.1		N/A
3.1.3 SFSC Heat Removal System	3.3.1 SFSC Heat Removal System	3.1.1 SFSC Heat Removal System	There is no difference between the NRC Std and the DCPP TS.
LCO 3.1.3	LCO 3.3.1	LCO 3.1.1	There is no technical difference between the NRC Std and the DCPP TS. However in Action B.1, DCPP 's TS has been revised to specifically provide direction to transfer the MPC into a Transfer Cask. This immediately puts the MPC in a safe condition as referred to in the NRC Std., and allows either further action of placing the MPC is another operable Overpack or unloading the MPC contents back into the fuel pool.
SR 3.1.3	SR 3.3.1.1	SR 3.1.1	The NRC Std allows two surveillance possibilities. It implies physical verification of no blockage or verification of temperature differentials on Overpacks that have temperature monitors. DCPP does not have temperature monitors and will physically verify no

NRC Standard ISFSI Technical Specifications[1]	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
			blockage as the surveillance. DCPP provides 24 hours between surveillances similar to the NRC Std.
		3.1.2 SFSC Time Limitation in Cask Transfer Facility	This is a site specific TS that addresses the limited amount of time a loaded SFSC can be in the CTF to ensure not exceeding any short-term temperature limits because of reduced airflow around the SFSC.
		LCO 3.1.2	The actions provided require if the SFSC cannot be moved that adequate cooling capability be verified within the 22 hours of being placed in the CTF, and once this is verified removal of the loaded SFSC within 30 days. As an alternative to the above actions, it allows the loaded MPC to be removed from the SFSC into the Transfer Cask within the 22 hours of being placed in its lowered position in the CTF.
		SR 3.1.2.1	Verifies that the SFSC is not in the CTF for more then 22 hours.
3.2 Cask Integrity			This section is not applicable to DCPP because the MPCs are welded and not bolted.
3.2.1 Cask Interseal Pressure			This section is not applicable to DCPP because the MPCs are welded and not bolted.
LCO 3.2.1			N/A
SR 3.2.1			N/A
SR 3.2.2			N/A
3.3 Cask Criticality Control Program		3.2 Cask Criticality Control Program	There is no difference between the NRC Std and the DCPP TS.

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NRC Standard ISFSI Technical Specifications(1)	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
3.3.1 Dissolved Boron Concentration		3.2.1 Dissolved Boron Concentration	There is no difference between the NRC Std and the DCPP TS.
LCO 3.3.1		LCO 3.2.1	The DCPP LCO is similar to the NRC Std except that "MPC" is used instead of "cask". Also there are three levels of boron concentrations dependent on type of MPC and initial enrichment of the approved content. In addition the DCPP Action A.1 uses "Loading Operations and Unloading Operations" in place of "...loading of fuel assemblies into cask...". DCPP believes that the intent of this LCO is for both loading and unloading as stated in the Applicability. Therefore the action should address both conditions.
SR 3.3.1.1		SR 3.2.1.1	There is no difference in the surveillance except the use of "MPC" in place of "cask". However, DCPP has modified the frequency from 4 hours to 8 hours. This change will give more operational flexibility. The bases for this modification is that once the boron concentration has been verified changing it will take a specific action such as diluting it by adding or subtracting water. Per the TS bases, during the period between verification and loading all activities that could affect the concentration will be administratively controlled. If any active of this type takes place then the concentration will be re-verified prior to loading of any approved contents.

NRC Standard ISFSI Technical Specifications(1)	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
SR 3.3.1.2		SR 3.2.1.2	There is no difference except the use of "MPC" in place of "cask". However, DCPP has modified the frequency from 4 hours to 8 hours. This change will give more operational flexibility. The bases for this modification is that once the boron concentration of the source water has been verified, changing it will take a specific action such as, diluting it by adding water or changing the actual source. Per the TS bases, during the period between verification and introducing water into the MPC for unloading, all activities that could affect the boron concentration will be administratively controlled. If any activity of this type takes place then the boron concentration of the source water will be re-verified prior to introducing that water into the MPC to be unloaded.
3.3.2 Canister/Cask Water Temperature			In the DCPP TS water temperature in the MPC is important and the water must not be allowed to boil off. However, actually monitoring the temperature in the MPC during some activities may not be possible. As a result, DCPP requires that this parameter be verified by analysis to ensure that it cannot reach boil off. The NRC Std provides a LCO to control this parameter and it allows either a temperature limit or a time limit based on decay heat. DCPP believes that the concern here is that there is

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NRC Standard ISFSI Technical Specifications(1)	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
			adequate control of the water temperature in the MPC at all times. However, the time limit for a specific MPC will be variable depending on content and that providing a fixed time limit may not be reasonable. As a result, DCPP believes that this parameter is better controlled similar to the other parameters that are of concern during loading and unloading operations. As a result, it has been placed in the TS Section 5.1.3 "MPC and SFSC Loading, Unloading and Preparation Program". This TS program will provide the direction and methodology for the verification that the water in the MPC cannot reach a temperature of 212°F. This program will provide adequate controls, required actions, compensatory measures, and completion times for all conditions.
LCO 3.3.2			N/A
SR 3.3.2			N/A
4.0 Design Features	3.0 Design Features	4.0 Design Features	There is no difference between the NRC Std and the DCPP TS.
4.1 Design Features Significant to Safety		4.1 Design Features Significant to Safety	There is no difference between the NRC Std and the DCPP TS.
	3.1 Site		Not included in NRC Std.
	3.1.1 Site Location		Not included in NRC Std.
4.1.1 Criticality Control	3.2 Design Features Important for Criticality Control	4.1.1 Criticality Control	There is no technical difference. NRC Std lists Flux Traps and ^{10}B Loading as does DCPP TS.
	3.2.1 MPC-24	4.1.1.1 Multi-	DCPP's section is broken in to a sub-section that

NRC Standard ISFSI Technical Specifications(1)	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
		purpose Canister (MPC) MPC-24	provides a detailed list of Flux Traps and ^{10}B Loading for a specific MPC.
	3.2.2 MPC-68 and MPC-68FF	N/A	Does not apply to DCPP
	3.2.3 MPC-68F	N/A	Does not apply to DCPP
	3.2.4 MPC-24E and MPC-24EF	4.1.1.2 MPC-24E and MPC-24 EF	DCPP's section is broken in to a sub-section that provides a detailed list of Flux Traps and ^{10}B Loading for a specific MPC.
	MPC-32	4.1.1.3 MPC-32	DCPP's section is broken in to a sub-section that provides a detailed list of Flux Traps and ^{10}B Loading for a specific MPC.
4.1.2 Materials			NRC Std allows incorporation of this information in to Condition 1.b of the CofC. This is where DCPP has placed it.
4.2 Codes and Standards	3.3 Codes and Standards	4.2 Codes and Standards	NRC Std and Holtec list standards. DCPP provides codes for the confinement boundary only, and refers to others listed in SAR, which is believed to be a better place to control this information.
4.2.1 Alternative to Codes, Standards, and Criteria	3.3.1 Exceptions to Codes, Standards, and Criteria		NRC Std and Holtec list standards. However DCPP refers to list in SAR, which is believed to be a better place to control this information.
4.2.2 Construction/ Fabrication Alternatives to Codes, Standards, and Criteria	3.3.2 Construction/Fabrication Alternatives to Codes, Standards, and Criteria		NRC Std and Holtec list standards. However DCPP refers to list in SAR, which is believed to be a better place to control this information.
4.3 Structural Performance			NRC Std and Holtec list these parameters. However, DCPP does not consider that this information is prone to change or that it is necessary to be controlled in

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			TS. DCPP believes it is better controlled in the SER and has not included it.
4.3.1 Earthquake Loads			NRC Std and Holtec list these parameters. However, DCPP does not consider that this information is prone to change or that it is necessary to be controlled in TS. DCPP believes it is better controlled in the SER and has not included it.
4.3.2 Design G-Loads			NRC Std and Holtec list these parameters. However, DCPP does not consider that this information is prone to change or that it is necessary to be controlled in TS. DCPP believes it is better controlled in the SER and has not included it.
	3.4 Site-Specific Parameters and Analyses		Holtec Discusses earthquake and design loads, soils and other site-specific parameters for site. DCPP discusses some site-specific parameters below.
4.4 Cask Handling/Canister Transfer Facility	3.5 Cask Transfer Facility (CTF)	4.3 Cask Handling/Cask Transfer Facility	The title is the same. NRC Std provides a place for discussion of operations outside of the Part 50 facility.
		4.3.1 Cask Transporter	DCPP provides design features for the specific equipment here.
		4.3.2 Storage Capacity	DCPP provides design features for the specific facility here.
	3.5.1 Transfer Cask and MPC Lifters	4.3.3 Overpack Load Handling Equipment	DCPP provides design features for the specific CTF equipment being used.
	3.5.2 CTF Structural Requirements	4.3.4 CTF Structural Requirements	DCPP provides design features for the specific CTF structure being used.

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NRC Standard ISFSI Technical Specifications[1]	Holtec Certificate of Compliance No. 1014	Diablo Canyon ISFSI Technical Specifications	Comments and Discussion of Differences Between DC ISFSI TS and NRC Std TS
5.0 Administrative Controls		5.0 Administrative Controls	There is no difference between the NRC Std and the DCPP TS.
5.1 Administrative Programs		5.1 Administrative Programs	There is no difference between the NRC Std and the DCPP TS.
		5.1.1 Technical Specification (TS) Bases Control Program	This program has been recently added to the DCPP operating TS and adding it is believed to be prudent to ensure proper control of the ISFSI TS bases.
5.1.1 Radioactive Effluent Control Programs		5.1.2 Radioactive Effluent Control Program (RECP)	There is no technical difference. This program will provide overall monitoring and control of effluent and dose.
5.1.2 Cask Loading, Unloading, and Preparation Program		5.1.3 MPC and SFSC Loading, Unloading and Preparation Program	This program monitors and controls the same parameters that the NRC Std does to ensure the fuel and cask system integrity at all times during loading, unloading and preparation for loading. In addition, the DCPP program monitors annular gap water level, water temperature in a loaded MPC, and exit gas temperature prior to re-flooding. There are two parameters that are in the NRC Std. Program that are not specifically included in the DCPP program. One is verification of ambient and pool water temperature, and the other is spent fuel pool boron concentration. DCPP has not included these because they are adequately controlled by the Part 50 license. DCPP believes that the spent fuel pool is part of the Part 50 facility and its parameters should be maintained by that license. The ISFSI TS

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			should control the activities and parameters in the MPCs and the program provided in the ISFSI TS adequately controls those things.
5.1.3 ISFSI Operations Program		5.1.4 ISFSI Operations Program	This program controls the same elements as the NRC Std. Since DCPP SFSCs are bolted in place on studs provided in the pad, the center to center distance can not vary and needs no control program. As a result, we control the SFSC location on to the studs. In addition, DCPP uses a single failure proof lifting and transporting system. As a result, there is no creditable drop accident and a maximum lift distance is not a controlled parameter.
		5.1.6 Cask Transportation Evaluation Program	DCPP has a site-specific program to control specific transportation parameters and Transport operation issues. As a result of the of the distance between the operating plant and the ISFSI facility, various issues become more critical and need to be monitored and controlled. These include transportation route road surface conditions, on-site hazard locations, security, and transporter controls. In addition, there are Transport Operation parameters, which are controlled by this program. They include CTF equipment operability and SFSC auxiliary cooling capability availability. Control of these parameters ensures that the Transport Operations performed at the CTF are successful.

PROPOSED TECHNICAL SPECIFICATIONS

FOR

DIABLO CANYON

INDEPENDENT SPENT FUEL STORAGE INSTALLATION

1.0 USE AND APPLICATION

1.1 Definitions

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CASK TRANSFER FACILITY (CTF)	The CASK TRANSFER FACILITY includes the following components and equipment: (1) A cask transfer structure (built into the transporter) used to lift and stabilize the TRANSFER CASK and MPC during lifts involving spent fuel outside of structures governed by 10 CFR 50, and (2) an in-ground cask transfer structure with a mechanical lifting device used in concert with the transporter to lift the OVERPACK and MPC.
DAMAGED FUEL ASSEMBLY	DAMAGED FUEL ASSEMBLIES are fuel assemblies with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with solid Zircaloy or stainless steel rods, or those that cannot be handled by normal means. DAMAGED FUEL ASSEMBLIES if stored in an MPC, must be stored in a DAMAGED FUEL CONTAINER.
DAMAGED FUEL CONTAINER (DFC)	DFCs are specially designed enclosures for DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS that permit gaseous and liquid media to escape to the atmosphere in the MPC, while minimizing dispersal of gross particulates within the MPC. A DFC can hold one DAMAGED FUEL ASSEMBLY or an amount of FUEL DEBRIS equivalent to that of an INTACT FUEL ASSEMBLY.
FUEL DEBRIS	FUEL DEBRIS is ruptured fuel rods, severed rods, loose fuel pellets or fuel assemblies with known or suspected defects, which cannot be handled by normal means due to fuel cladding damage.

(continued)

1.1 Definitions (continued)

INTACT FUEL ASSEMBLY	INTACT FUEL ASSEMBLY is a fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. A fuel assembly shall not be classified as INTACT FUEL ASSEMBLY unless solid Zircaloy or stainless steel rods are used to replace missing fuel rods and which displace an amount of water equal to that displaced by the original fuel rod(s).
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on a TRANSFER CASK while its contained MPC is being loaded with its approved contents. LOADING OPERATIONS begin when the first fuel assembly is placed in the MPC and end when the TRANSFER CASK is suspended from or secured on the transporter. LOADING OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.
MULTI-PURPOSE CANISTER (MPC)	MPC is a sealed SPENT NUCLEAR FUEL container that consists of a honeycombed fuel basket contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. The MPC provides the confinement boundary for the contained radioactive materials.
NONFUEL HARDWARE	NONFUEL HARDWARE is defined as burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), rod control cluster assemblies (RCCAs), and wet annular burnable absorbers (WABAs).
OPERABLE/OPERABILITY	A system, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instruments, controls, normal or emergency electrical power, and other auxiliary equipment that are required for the system, component, or device to perform its specific safety function(s) are also capable of performing their related support function(s).
OVERPACK	OVERPACK is a cask that receives and contains a sealed MPC for interim storage in the independent spent fuel storage installation (ISFSI). It provides gamma and neutron shielding, and provides for ventilated air flow to promote heat transfer from the MPC to the environs. The OVERPACK does not include the TRANSFER CASK.

(continued)

1.1 Definitions (continued)

SPENT FUEL STORAGE CASKS (SFSCs)	SFSCs are containers approved for the storage of spent fuel assemblies, FUEL DEBRIS, and associated NONFUEL HARDWARE at the ISFSI. The HI-STORM 100 SFSC System consists of the OVERPACK and its integral MPC.
SPENT NUCLEAR FUEL	SPENT NUCLEAR FUEL means fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least one year's decay since being used as a source of energy in a power reactor and has not been chemically separated into its constituent elements by reprocessing. SPENT NUCLEAR FUEL includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI after a loaded SFSC has been anchored to the storage pad within the ISFSI protected area perimeter. STORAGE OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.
TRANSFER CASK	TRANSFER CASKs are containers designed to contain the MPC during and after loading of its approved contents and to transfer the loaded MPC to or from the OVERPACK.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities performed on an OVERPACK or TRANSFER CASK containing a MPC loaded with any approved contents when it is being moved to and from the ISFSI. TRANSPORT OPERATIONS begin when the OVERPACK or TRANSFER CASK is first suspended from or secured on the transporter and end when the OVERPACK or TRANSFER CASK is at its destination and no longer secured on or suspended from the transporter. TRANSPORT OPERATIONS include transfer of the MPC between the OVERPACK and the TRANSFER CASK.

(continued)

1.1 Definitions (continued)

UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on a TRANSFER CASK while its contained MPC is being unloaded of its approved contents. UNLOADING OPERATIONS begin when the TRANSFER CASK is no longer suspended from or secured on the transporter and end when the last of its approved contents is removed from the MPC. UNLOADING OPERATIONS do not include MPC transfer between the TRANSFER CASK and the OVERPACK.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

<u>EXAMPLE 1.2-1</u>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify...... <u>AND</u> A.2 Restore....	

In this example, the logical connector <u>AND</u> is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Stop..... OR A.2.1 Verify.... AND A.2.2.1 Reduce.... OR A.2.2.2 Perform.... OR A.3 Remove....	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three ACTIONS may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS condition unless otherwise specified, providing the cask system is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the condition no longer exists or the cask system is not within the LCO Applicability. Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will <u>not</u> result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1	12 hours
	AND	
	B.2 Perform Action B.2	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completion action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One system not within limit.	A.1 Restore system to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1. AND B.2 Complete action B.2.	12 hours 36 hours

When a system is determined not to meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

---NOTE---
Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore compliance with LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1 AND B.2 Complete action B.2	6 hours 12 hours

The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

(continued)

1.4 Frequency (continued)

EXAMPLES

The following examples illustrate the various ways that frequencies are specified.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment or variables are outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours prior to starting activity AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one-time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicated that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.

"Thereafter" indicated future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is cancelled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

2.0 APPROVED CONTENTS

2.1 Cask content shall be limited to SPENT NUCLEAR FUEL, FUEL DEBRIS and NONFUEL HARDWARE initially approved by the NRC in Section 10.2 of the Diablo Canyon ISFSI Safety Analysis Report (SAR).

2.2 Proposed alternatives to contents listed in Section 10.2 of the SAR may be authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternative contents should demonstrate that:

1 The proposed alternative contents would provide an acceptable level of safety, and

2 The proposed alternative contents are consistent with the applicable requirements.

Requests for alternatives to contents shall be submitted in accordance with 10 CFR 72.4.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.
LCO 3.0.3	Not applicable.
LCO 3.0.4	When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of an SFSC.
LCO 3.0.5	Not applicable.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per ..." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCOs Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with Actions or that are related to the unloading of an SFSC.

3.1 FUEL INTEGRITY

3.1.1 Spent Fuel Storage Cask (SFSC) Heat Removal System

LCO 3.1.1 The SFSC Heat Removal System shall be operable.

APPLICABILITY: During STORAGE OPERATIONS while MPC is in the storage OVERPACK

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore SFSC Heat Removal System to operable status.	8 hours
	<u>OR</u>	
	A.2.1 Verify adequate heat removal to prevent exceeding short-term fuel temperature limit;	Immediately
	<u>AND</u>	
	A.2.2 Restore SFSC Heat Removal System to operable status.	30 days
B. Required Actions and associated Completion Time not met.	B.1 Transfer the MPC into a TRANSFER CASK.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify all OVERPACK inlet and outlet air duct screens are free of blockage and are intact.	24 hours

3.1 FUEL INTEGRITY

3.1.2 Spent Fuel Storage Cask (SFSC) Time Limitation in Cask Transfer Facility (CTF)

LCO 3.1.2 The SFSC shall not be in the CTF for greater than 22 Hours

APPLICABILITY: During TRANSPORT OPERATIONS while the SFSC is in the CTF and contains a loaded MPC.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Remove SFSC from CTF.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify SFSC containing a loaded MPC in the CTF.	22 hours from the SFSC containing a loaded MPC initially being placed in the CTF or from a loaded MPC initially being lowered into an empty SFSC in the CTF

3.2 Cask Criticality Control Program

3.2.1 Dissolved Boron Concentration

LCO 3.2.1 The dissolved boron concentration in the water of the MPC cavity shall be as follows:

a. For all MPCs with one or more fuel assemblies having initial enrichment of $\leq$ 4.1 wt% ^{235}U: $\geq$ 2000 ppmb.

b. For MPC24/24E/24EF with one or more fuel assemblies having initial enrichment of $>$ 4.1 and $\leq$ 5.0 wt% ^{235}U: $\geq$ 2000 ppmb.

c. For MPC 32 with one or more fuel assemblies having initial enrichment of $>$ 4.1 and $\leq$ 5.0 wt% ^{235}U: $\geq$ 2600 ppmb.

APPLICABILITY: During LOADING OPERATIONS and UNLOADING OPERATIONS with water and at least one fuel assembly in the MPC.

ACTIONS

--NOTE--

Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Dissolved boron concentration not met.	A.1 Suspend LOADING OPERATIONS or UNLOADING OPERATIONS and any other action that increases reactivity.	Immediately
	<u>AND</u>	
	A.2 Restore boron concentration to within limits	Immediately
	<u>AND</u>	
	A.3 Remove all fuel assemblies from MPC	24 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.1.1	Verify the dissolved boron concentration is met using two independent measurements.	Within 8 hours prior to commencing LOADING OPERATIONS <u>AND</u> Every 48 hours thereafter while the MPC is in the spent fuel pool or while water is in the MPC.
SR 3.2.1.2	Verify the dissolved boron concentration is met using two independent measurements.	Within 8 hours prior to commencing UNLOADING OPERATIONS <u>AND</u> Every 48 hours thereafter while the MPC is in the spent fuel pool or while water is in the MPC.

4.0 DESIGN FEATURES

4.1 Design Features Significant to Safety

4.1.1 Criticality Control

a. MULTI-PURPOSE CANISTER (MPC) MPC-24

1. Flux trap size: $\geq$ 1.09 in.

2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.0267 g/cm^2

b MPC-24E and MPC-24EF

1. Flux trap size:

   - Cells 3, 6, 19, and 22: $\geq$ 0.776 in.

   - All Other Cells: $\geq$ 1.076 in.

2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.0372 g/cm^2

c. MPC-32

1. Fuel cell pitch: $\geq$ 9.158 in.

2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.0372 g/cm^2

4.2 Codes and Standards

The following provides information on the governing codes for the confinement boundary (important to Safety) design:

MPC (Shell and Head)	Applicable Codes	Editions/Years
Material Procurement	ASME III, NB-2000	ASME Code, 1995 Edition. 1997 Addenda
Design	ASME III, NB-3200	ASME Code, 1995 Edition. 1997 Addenda
Fabrication	ASME III, NB-4000	ASME Code, 1995 Edition. 1997 Addenda
Examination	ASME III, NB-5000	ASME Code, 1995 Edition. 1997 Addenda

Any specific exceptions to these codes and standards, and the codes and standards for other components followed for the Diablo Canyon ISFSI storage system, are provided in the Diablo Canyon ISFSI Safety Analysis Report (SAR).

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Cask Handling/Cask Transfer Facility

4.3.1 Cask Transporter

A site-specific cask transporter is used to transport the TRANSFER CASK between the power plant and the CASK TRANSFER FACILITY (CTF) and the SPENT FUEL STORAGE CASK (SFSC) between the CTF and ISFSI pad. The requirements for the cask transporter are as follows:

a. TRANSPORT OPERATIONS shall be conducted using the cask transporter.

b. The cask transporter fuel tank shall not contain > 50 gallons of diesel fuel at any time.

c. The cask transporter shall be designed, fabricated, inspected, maintained, operated, and tested in accordance with the applicable guidelines of NUREG-0612.

d. The cask transporter lifting towers shall have redundant drop protection features.

e. Lifting of a SFSC, loaded TRANSFER CASK, or loaded MPC outside of structures governed by 10 CFR 50 shall be performed with lifting devices that are designed, fabricated, inspected, maintained, operated and tested in accordance with the applicable guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

4.3.2 Storage Capacity

The Diablo Canyon ISFSI can accommodate up to 4,400 spent fuel assemblies and other NONFUEL HARDWARE. The ISFSI storage capacity will accommodate up to 140 SFSCs (138 plus 2 spare locations).

4.3.3 OVERPACK Load Handling Equipment

Lifting of a loaded OVERPACK outside of structures governed by 10 CFR 50 shall be performed with load handling equipment that is designed, fabricated, inspected, maintained, operated and tested in accordance with the applicable guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" as clarified by Section 4.5.2 below. Section 4.5.2 below does not apply to the lifting of heavy loads outside the structures governed by the regulations of 10 CFR 50.

4.3.4 CTF Structure Requirements

a. Permanent Load Handling Equipment

1. The weldment structure of the CTF shall be designed to comply with the stress limits of ASME Code, Section III, Subsection NF, Class 3 for linear structures. All compression-loaded members shall satisfy the buckling criteria of ASME Section III, Subsection NF. The applicable loads, load combinations, and associated service condition definitions are provided in Diablo Canyon ISFSI SAR Section 4.4.5.

2. The reinforced concrete structure of the CTF shall be designed in accordance with ACI-349-1997, as clarified in Diablo Canyon ISFSI SAR Section 4.2.1.2.

(continued)

4.0 DESIGN FEATURES (continued)

b. Mobile Load Handling Equipment

Mobile load handling equipment used in lieu of permanent load handling equipment, shall meet the guidelines of NUREG-0612, Section 5.1, with the following clarifications:

1. Mobile lifting devices shall have a minimum safety factor of two over the allowable load table for the lifting device in accordance with the guidance of NUREG-0612, Section 5.1.6(1)(a) and shall be capable of stopping and holding the load during a Design Basis Earthquake (DBE) event.
2. Mobile lifting devices shall conform to the requirements of ASME B30.5, "Mobile and Locomotive Cranes," in lieu of the requirements of ASME B30.2, "Overhead and Gantry Cranes."
3. Mobile cranes are not required to meet the guidance of NUREG-0612, Section 5.1.6(2) for new cranes.
4. Horizontal movements of the TRANSFER CASK and MPC using a mobile crane are prohibited.

5.0 ADMINISTRATIVE CONTROLS

5.1 Administrative Programs

The following programs shall be established, implemented, and maintained:

5.1.1 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these TS.

a. Changes to the TS Bases shall be made under appropriate administrative controls and reviews.

b. Changes to the TS Bases may be made without prior NRC approval in accordance with the criteria in 10 CFR 72.48.

c. The TS Bases Control Program shall contain provisions to ensure that the TS Bases are maintained consistent with the Diablo Canyon ISFSI SAR.

d. Proposed changes that do not meet the criteria of 5.5.1.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the TS Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 72.48 (b) (2).

5.1.2 Radioactive Effluent Control Program

This program is established and maintained to:

a. Implement the requirements of 10 CFR 72.44 (d) or 72.126, as appropriate

b. Provide limits on surface contamination of the TRANSFER CASK and verification of meeting those limits prior to removal of a loaded TRANSFER CASK from the fuel handling building/auxiliary building.

c. Provide MPC leakage rate limits and verification of meeting those limits prior to removal of a loaded TRANSFER CASK from the fuel handling building/auxiliary building.

d. Provide an effluent monitoring program, as appropriate, if the surface contamination limits are greater than the values specified in Regulatory Guide 1.86; or if the leakage rate limits are greater then the values specified as "Leaktight" in ANSI N14.5 – 1997 "Leakage Tests on Packages for Shipment".

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.1.3 MPC and SFSC Loading, Unloading, and Preparation Program

This program shall be established and maintained to implement Diablo Canyon ISFSI SAR Section 10.2 requirements for loading fuel and components into MPCs, unloading fuel and components from MPCs, and preparing the MPCs for storage in the SFSCs. The requirements of the program for loading and preparing the MPC shall be complete prior to removing the MPC from the fuel handling building/auxiliary building. The program provides for evaluation and control of the following requirements during the applicable operation:

a. Verify the maintenance of water in the annular gap between the loaded MPC and TRANSFER CASK during MPC moisture removal operations (loading) or MPC reflooding operations (unloading).

b. The water temperature of a water-filled or partially filled loaded MPC shall be shown by analysis to be less than boiling at all times.

c. Verify that the vacuum drying times and pressures assure that short-term fuel temperature limits are not violated and the MPC is adequately dry.

d. Verify that the inerting backfill pressure and purity assure adequate heat transfer and corrosion control.

e. Verify that leak testing assure adequate MPC integrity and consistency with offsite dose analysis.

f. Verify surface dose rates on the TRANSFER CASK are adequate to assure proper loading and consistency with the offsite dose analysis.

g. Verify surface dose rates on the SFSCs are adequate to assure proper storage and consistency with the offsite dose analysis.

h. During MPC re-flooding, verify the helium exit temperature is such that water quenching or flashing does not occur.

i) Fuel cladding oxide thickness shall be evaluated to determine the average fuel cladding oxide thickness of high burnup (> 45,000 MWD/MTU) SPENT NUCLEAR FUEL assemblies proposed to be stored in the ISFSI facility. Direct physical measurements or an appropriate predictive methodology with due consideration of all significant variables (e.g., in-core flux, cycle length and number, power history, coolant temperature profile, coolant chemistry, and metallurgy of the fuel cladding material) to determine the average oxide thickness on the fuel cladding may be used. If a predictive methodology is used to determine average fuel cladding oxide thickness, a sufficient number of fuel cladding thickness measurements shall be made to adequately benchmark the methodology.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.1.3 (cont'd) In order to classify a high burnup spent fuel assembly as an INTACT FUEL ASSEMBLY, the maximum allowable average fuel cladding oxidation layer thickness (t_{ox}) shall be:

For DCPP LOPAR assemblies without IFBA fuel, t_{ox} = 173.5 micrometers.

For DCPP VANTAGE 5 assemblies without IFBA fuel, t_{ox} = 190.5 micrometers.

For DCPP LOPAR assemblies with IFBA fuel, t_{ox} = 67 micrometers.

For DCPP VANTAGE 5 assemblies without IFBA fuel, t_{ox} = 88 micrometers.

A high burnup spent fuel assembly shall be considered a DAMAGED FUEL ASSEMBLY if the computed or measured average oxidation layer thickness on any fuel rod exceeds the applicable limit above.

This program will control limits, surveillances, compensatory measures and appropriate completion times to assure the integrity of the fuel cladding at all times in preparation of and during LOADING, UNLOADING or TRANSPORT OPERATIONS, as applicable.

5.1.4 ISFSI Operations Program

This program will implement the Diablo Canyon ISFSI SAR requirements for ISFSI operations. It will included criteria to be verified and controlled:

a) SFSC cask storage location.

b) Design features listed in Section 4.0 and design basis ISFSI pad parameters consistent with the Diablo Canyon ISFSI SAR analysis.

5.1.5 Cask Transportation Evaluation Program

This program will evaluate and control the transportation of loaded MPCs between the DCPP fuel handling building/auxiliary building, the CTF and the ISFSI storage pads. Included in this program will be pre-transport evaluation and control during transportation of the following:

- Transportation route road surface conditions.
- Onsite hazards along the transportation route.
- Security
- Transporter control functions and operability
- CTF equipment operability
- SFSC auxiliary cooling capability availability

TECHNICAL SPECIFICATION BASES
FOR
DIABLO CANYON
INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

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B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCO	LCO 3.0.1, 3.0.2, and 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification.)

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS condition is applicable from the point in time that an ACTIONS condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore a system or component or to restore variables to within specified limits. Whether stated as a Required Action or not, correction of the entered condition is an action that may always be considered upon entering ACTIONS. The second type of Required Action specifies the remedial measures that permit continued operation that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.

(continued)

BASES (continued)

LCO 3.0.3	This specification is not applicable to the Diablo Canyon ISFSI because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.
LCO 3.0.4	LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the facility in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist: a. Facility conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in being required to exit the Applicability desired to be entered to comply with the Required Actions. Compliance with Required Actions that permit continued operation of the facility for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the facility. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability. The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS, or that are related to the unloading of an SFSC Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

(continued)

BASES (continued)

LCO 3.0.5	This specification is not applicable to the Diablo Canyon ISFSI because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that systems and components meet the LCO and variables are within specified limits. Failure to complete a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the facility is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service. Upon completion of maintenance, appropriate post-maintenance testing is required. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2.

Post-maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary facility parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a specified condition where other necessary post-maintenance tests can be completed.

(continued)

BASES (continued)

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per....." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ...." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the affected equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

(continued)

BASES (continued)

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by the Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility, which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not complete within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

(continued)

BASES (continued)

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe operation of the facility.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is outside the specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on equipment that has been determined to not meet the LCO. When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements of performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs annotation is found in Diablo Canyon ISFSI Technical Specification Section 1.4, Frequency.

B 3.1 FUEL INTEGRITY

B 3.1.1 Spent Fuel Storage Cask (SFSC) Heat Removal System

BASES

BACKGROUND	The SFSC heat removal system is a passive, air-cooled, convective heat transfer system that ensures heat from the MULTI-PURPOSE CANISTER (MPC) is transferred to the environs by the chimney effect. Relatively cool air is drawn into the annulus between the OVERPACK and the MPC through the four inlet air ducts at the bottom of the OVERPACK. The MPC transfers its heat from the canister surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air is forced back into the environs through the four outlet air ducts at the top of the OVERPACK.
APPLICABLE SAFETY ANALYSIS	The thermal analyses of the SFSC take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the OVERPACK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and other SFSC component temperatures do not exceed applicable limits. Under normal storage conditions, the four inlet and four outlet air ducts are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs. Analyses have been performed for the complete obstruction of two, three, and four inlet air ducts. Blockage of two inlet air ducts reduces airflow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this off-normal condition, no SFSC components exceed the short-term temperature limits. Blockage of three inlet air ducts further reduces airflow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this accident condition, no SFSC components exceed the short-term temperature limits. The complete blockage of all four inlet air ducts stops air-cooling of the MPC. The MPC will continue to radiate heat to the relatively cooler inner shell of the OVERPACK. With the loss of air-cooling, the SFSC component temperatures will increase toward their respective short-term temperature limits. None of the components reach their temperature limits over the 72-hour duration of the analyzed event. Therefore, the limiting component is assumed to be the fuel cladding.

(continued)

BASES (continued)

LCO	The SFSC heat removal system must be verified to be operable to preserve the assumptions of the thermal analyses. The operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environs at a sufficient rate to maintain fuel cladding and other SFSC component temperatures within design limits.
APPLICABILITY	The LCO is applicable during STORAGE OPERATIONS. Once an OVERPACK containing an MPC loaded with approved contents has been placed in storage, the heat removal system must be operable to ensure adequate heat transfer of the decay heat away from the fuel assemblies.
ACTIONS	A note has been added to the ACTIONS, which states that for this LCO, separate condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent condition entry and application of associated Required Actions. <u>A.1</u> If the heat removal system has been determined to be inoperable, it must be restored to operable status within 8 hours. Eight hours is reasonable period of time (typically, one operating shift) to take action to remove the obstructions in the air flow path. <u>A.2.1</u> As an alternative to meeting A.1, adequate heat removal capability must be verified to exist to prevent exceeding the short-term fuel cladding temperature limit. This verification must be performed immediately. Thermal analysis of a fully blocked SFSC shows that without adequate heat removal the fuel cladding short-term temperature limit could be exceeded over time. As a result, requiring immediate verification of adequate heat removal capability will ensure that the SFSC components and the fuel cladding do not exceed their short-term temperature limits. The thermal analysis also shows that only complete blockage of all four vents results in the potential for exceeding short-term fuel cladding or other SFSC component limits. As a result, verifying that there is at least one vent operable or the equivalent cooling of one operable vent will ensure that the short-term limits are not exceeded while the remainder of the inlet vents are returned to operable status under Action A.2.2.

(continued)

BASES

ACTIONS (continued)

A.2.2

In addition to Required Action A.2.1, efforts must continue to restore the heat removal system to operable status.

As long as the adequate heat removal capability that was verified in A.2.1 exists, restoring the SFSC heat removal system to complete operability is not an immediate concern. Therefore, restoring it within 30 days is considered a reasonable period of time.

B.1

If the A.1, A.2.1 and A.2.2 actions cannot be met then the affected MPC must be placed in a safe condition. Transferring the affected MPC from the inoperable SFSC to the TRANSFER CASK will place the MPC in an analyzed condition. The TRANSFER CASK has adequate heat removal capability to ensure that the short-term fuel cladding temperature limit is not exceeded.

Transfer of the MPC into a TRANSFER CASK removes the SFSC from the LCO applicability. STORAGE OPERATIONS does not include time restrictions when the MPC resides in the TRANSFER CASK because of adequate heat transfer in this configuration to maintain peak fuel cladding temperature well below the short term limit.

(continued)

BASES

ACTIONS (continued)	B.1 (continued) If actions A.1, A.2.1 and A.2.2 are not met the Completion Time for this Required Action is immediate. Thermal analysis of a fully blocked SFSC shows that without adequate heat removal the fuel cladding short-term temperature limit could be exceeded over time. The analysis shows that without heat removal for the first 72 hours the SFSC components and the fuel cladding do not exceed their short-term temperature limits. However at that time, the temperatures are continuing to increase. By requiring immediate action, this should allow adequate time to transfer the MPC to the TRANSFER CASK as not to exceed the 72 hours.
SURVEILLANCE REQUIREMENTS	SR 3.1.1 The long-term integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment. Visual observation that all four inlet and outlet air ducts are unobstructed and intact ensures that airflow past the MPC is occurring and heat transfer is taking place. Complete blockage of any one or more inlet or outlet air ducts renders the heat removal system inoperable and this LCO not met. Partial blockage of one or more inlet or outlet air ducts does not constitute inoperability of the heat removal system. However, corrective actions should be taken promptly to remove the obstruction and restore full flow through the affected duct(s). The Frequency of 24 hours is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts.
REFERENCES	1. Diablo Canyon ISFSI SAR Section 3.4, Table 3.4-2 2. Diablo Canyon ISFSI SAR Section 4.4 3. Diablo Canyon ISFSI SAR Sections 7.1, 7.2, and 7.3 4. Diablo Canyon ISFSI SAR Section 8.1 5. Diablo Canyon ISFSI SAR Sections 8.2.11, 8.2.12, and 8.2.15

B 3.1 FUEL INTEGRITY

B 3.1.2 Spent Fuel Storage Case (SFSC) Time Limitation in Cask Transfer Facility (CTF)

BASES

BACKGROUND	The SFSC heat removal system is a passive, air-cooled, convective heat transfer system that ensures heat from the MULTI-PURPOSE CANISTER (MPC) is transferred to the environs by the chimney effect. Relatively cool air is drawn into the annulus between the OVERPACK and the MPC through the four inlet air ducts at the bottom of the OVERPACK. The MPC transfers its heat from the canister surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air is forced back into the environs through the four outlet air ducts at the top of the OVERPACK. However, while the SFSC is in the CTF there is a reduced cooling capability over this normal storage configuration because of ambient air access restrictions. As a result, over time the decay heat produced by the spent fuel may cause exceedance of the short term temperature limit of the fuel cladding or damage the shielding material. To ensure that this does not take place the time that a SFSC, with a loaded MPC, is allowed to be in the CTF shall be limited to 22 hours. If other CTF lifting mechanisms are not operable, the cask transporter is designed and shall be used to remove the loaded SFSC from the CTF without additional assistance using the HI-STORM Lift Links and Lifting Brackets
APPLICABLE SAFETY ANALYSIS	The thermal analyses of the SFSC take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the OVERPACK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and other SFSC component temperatures do not exceed applicable limits. Under normal storage conditions, the four inlet and four outlet air ducts are unobstructed and full airflow (i.e., maximum heat transfer for the given ambient temperature) occurs. However while the SFSC is in the CTF the restricted airflow around the SFSC decreases heat transfer capability. This case has been bounded by an analysis of a loaded TRANSFER CASK being in a loading pit with no external ventilation capability and is provided in the HI-STORM 100 System FSAR, Section 4.5.2. In that analysis, there is assumed to be only 10% of the normal heat transfer capability. Based on this, the temperature inside the MPC is shown not to reach the short-term limit of the fuel cladding within the first 22 hours. This analysis is considered bounding of the SFSC because the thermal inertia of the SFSC is greater then that of the TRANSFER CASK, therefore the heat-up is much slower. As a result, providing a time limitation of 22 hours for the SFSC to be in the CTF is conservative and adequate to ensure that the short-term temperature limits will not be met or exceeded.

(continued)

BASES (continued)

LCO	The SFSC, containing a loaded MPC, must not remain in the CTF for greater than 22 hours. This time limitation ensures that the decay heat generated by the approved content in a loaded MPC does not reach or exceed the approved content or other SFSC component temperature design limits.
APPLICABILITY	The LCO is applicable during TRANSPORT OPERATIONS while a SFSC containing a loaded MPC is in its lowered position in the CTF. If an OVERPACK in the CTF does not contain an MPC, which contains approve contents, then this LCO does not apply.
ACTIONS	A note has been added to the ACTIONS, which states that for this LCO, separate condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent condition entry and application of associated Required Actions. <u>A.1</u> If the LCO cannot be met the loaded SFSC must be removed from the CTF immediately to ensure adequate heat removal capability exist to prevent exceeding the short-term fuel cladding and SFSC component temperature limit. If the normal lifting mechanisms of the CTF are not capable of moving the loaded SFSC out of the CTF, the cask transporter shall be used to remove the loaded SFSC from the CTF. While the SFSC is in the CTF the restricted airflow around the SFSC decreases heat transfer capability. This case has been bounded by an analysis of a loaded TRANSFER CASK being in a loading pit with no external ventilation capability. In that analysis, there is assumed to be only 10% of the normal heat transfer capability. Based on this, the temperature inside the MPC is shown not to reach the short-term temperature limit of the fuel cladding within the first 22 hours. This analysis is considered bounding because the thermal inertia of the SFSC is greater when compared to the TRANSFER CASK, therefore the heat-up is much slower. As a result, providing a time limitation of 22 hours for the SFSC to be in the CTF is conservative and adequate to ensure that the short-term temperature limits will not be met or exceeded.

(continued)

BASES

ACTIONS	A.1 (continued) The Completion Time for this Required Action is immediately. The bounding analysis shows that the temperature inside the MPC does not reach the short-term temperature limit of the fuel cladding within the first 22 hours. As a result, providing a time limitation of 22 hours for the SFSC to be in the CTF is conservative and requiring immediate action to remove the loaded SFSC will ensure that the short-term temperature limits will not be met or exceeded.
SURVEILLANCE REQUIREMENTS	SR 3.1.1 The integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment. Verification that a loaded SFSC does not remain in the CTF for more then 22 hours will ensure that the short-term temperature limits will not be met or exceeded. The Frequency of 22 hours from the initial movement of a loaded SFSC into the CTF or from a loaded MPC being lowered into an empty SFSC in the CTF is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place. This surveillance is only required if the SFSC contains a loaded MPC and is in the CTF.
REFERENCES	1. Diablo Canyon ISFSI SAR Section 3.4, Table 3.4-2 2. Diablo Canyon ISFSI SAR Section 4.4 3. Diablo Canyon ISFSI SAR Sections 7.1, 7.2, and 7.3 4. Diablo Canyon ISFSI SAR Section 8.1 5. Diablo Canyon ISFSI SAR Sections 8.2.11, 8.2.12, 8.2.15, and 8.2.17

B 3.2 SPENT FUEL STORAGE CASK (SFCS) CRITICALITY CONTROL

B 3.2.1 Dissolved Boron Concentration

BASES

BACKGROUND	A TRANSFER CASK with an empty MULTI-PURPOSE CANISTER (MPC) is placed in the spent fuel pool (SFP) and loaded with fuel assemblies and associated NONFUEL HARDWARE meeting the requirements of Section 2.0, Approved Content. After loading the MPC, an MPC lid is placed on the MPC along with a lid retention device attached to the TRANSFER CASK. The TRANSFER CASK with the MPC inside is removed from the SFP to a washdown area. In the washdown area, the MPC lid is welded in place and the MPC is leak tested, drained, dried, and backfilled with helium. The TRANSFER CASK and accessible portions of the contained MPC are also surveyed to ensure that any radioactive contamination is within administrative limits. For those MPCs containing fuel assemblies of relatively high initial enrichment, credit is taken in the criticality analyses for boron in the water within the MPC. To preserve the analysis basis, users must verify that the dissolved boron concentration of the water in the MPC meets specified limits when there is fuel and water in the MPC. This may occur during LOADING OPERATIONS and UNLOADING OPERATIONS.
APPLICABLE SAFETY ANALYSIS	The spent nuclear fuel stored in the SFSC is required to remain subcritical ($k_{eff} \leq 0.95$) under all conditions of storage. The SFSC is analyzed to store a wide variety of spent nuclear fuel assembly types with differing initial enrichments and associated NONFUEL HARDWARE. For all allowed fuel loaded in the MPCs credit was taken in the criticality analyses for neutron poison in the form of soluble boron in the water within the MPC. Compliance with this LCO preserves the assumptions made in the criticality analyses regarding credit for soluble boron.
LCO	Compliance with this LCO ensures that the stored fuel will remain subcritical with a $k_{eff} \leq 0.95$ while water is in the MPC. The LCO provides the minimum concentration of soluble boron required in the MPC water based on type of MPC and the initial enrichment of the fuel. LCO 3.2.1.a provides the minimum concentration of soluble boron required in any of the MPCs if one or more fuel assemblies are loaded with an initial enrichment of $\leq$ 4.1 wt% U-235. LCO 3.2.1.b provides the minimum concentration of soluble boron required in MPC-24/24E/24EF if one or more fuel assemblies are loaded with an initial enrichment of > 4.1 wt% and $\leq$ 5.0 wt% U-235. LCO 3.2.1.c provides the minimum concentration of soluble boron required in MPC-32 if one or more fuel assemblies are loaded with an initial enrichment of > 4.1 wt% and $\leq$ 5.0 wt% U-235.

(continued)

BASES

LCO (continued)	All INTACT FUEL ASSEMBLIES loaded into the MPC-24, MPC-24E, MPC-24EF, and MPC-32 are limited by analysis to maximum enrichments of 5.0 wt% U-235. For all INTACT FUEL ASSEMBLIES loaded into an MPC that contains DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, the maximum initial enrichment of the INTACT FUEL ASSEMBLES is limited to the maximum initial enrichment of the DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS (i.e., 4.0 wt% U-235).
APPLICABILITY	The dissolved boron concentration LCO is applicable whenever an MPC-24, MPC-24E, MPC-24EF, or MPC-32 has at least one fuel assembly in a storage location and water in the MPC. During LOADING OPERATIONS, the LCO is applicable immediately upon the loading of the first fuel assembly in the MPC. It remains applicable until the MPC is drained of water. During UNLOADING OPERATIONS, the LCO is applicable when the MPC is reflooded with water after helium cool-down operations. Note that compliance with SR 3.0.4 ensures that the water to be used to flood the MPC is of the correct dissolved boron concentration to ensure the LCO is met upon entering the Applicability.
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate condition entry is allowed for each MPC. This is acceptable since the Required Actions for each condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent condition entry and application of associated Required Actions. <u>A.1</u> Continuation of LOADING OPERATIONS, UNLOADING OPERATIONS or positive reactivity additions (including ACTIONS to reduce dissolved boron concentration) is contingent upon maintaining the MPC in compliance with the LCO. If the dissolved boron concentration of water in the MPC is less than its limit, LOADING OPERATIONS or UNLOADING OPERATIONS, and any positive reactivity additions must be suspended immediately. Inherent in the required action to stop these activities is the requirement to place any in progress activity, such as the movement of a fuel assemble, in a safe condition.

(continued)

BASES

ACTIONS (continued)

A.2

In addition to immediately suspending LOADING OPERATIONS or UNLOADING OPERATIONS, and any positive reactivity additions, action to restore the concentration to within the limit specified in the LCO must be initiated immediately.

One means of complying with this action is to initiate boration of the affected MPC. In determining the required combination of boration flow rate and concentration, there is no unique design bases event that must be satisfied; only that boration be initiated without delay. In order to raise the boron concentration as quickly as possible, the operator should begin boration with the best source available for existing plant conditions. The methods available for boration should include, but not be limited to, direct boration of the MPC or boration of the SFP if the MPC is located in the pool at the time.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

A.3

In addition to actions A.1 and A.2, all of the approved contents that are loaded in the affected MPC must be removed to ensure that there have been no adverse affects. Once removed and verified as continuing to be acceptable, they may be reloaded in the MPC as long as the boron concentration in the MPC meets the requirements of this LCO.

This action must be completed within 24 hours of the discovered condition. Since this LCO only applies when there is fuel and water in the MPC, the MPC will be in the vicinity of the SFP or actually in the pool. As a result, the 24 hours is considered adequate time to allow for the movement of the affected MPC to a position where it can be unloaded, preparing it to be unloaded, and unloading its contents. Because no further changes in reactivity are allowed by action A.1, and the dissolved boron concentration in the affected MPC has been restored to limits and the contents are assured of remaining subcritical by action A.2, there should be no further potential for criticality or deterioration while the approved contents are in the MPC. As a result, the 24 hours is considered conservative.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

When the MPC is placed in the SFP the dissolved boron concentration in the MPC water must be verified by two independent measurements to be within the applicable limit within 8 hours prior to entering the applicability of the LCO. For LOADING OPERATIONS, this means within 8 hours prior to loading any approved content into the cask.

The use of two independent measurements provides reasonable assurance that the dissolved boron LCO limit is met and maintained. The 8 hours limitation is considered a reasonably short time period which minimizes any potential for changes in the critical dissolved boron concentration prior to loading and still allows flexibility in the operation. Once the dissolved boron concentration has been verified a change in this concentration is not credible unless there is some action specifically taken to modify it. During the period between verification and loading all changes in water volume including additions or subtractions in the SFP or MPC; recirculation of water through the MPC; or the addition or dilution of the dissolved boron concentration in the SFP or MPC to be loaded, will be administratively controlled. If any of these actions or operations takes place during the 8-hour period, the dissolved boron concentration will be re-verified to be within limits prior to loading any authorized contents in the MPC.

In addition, while the MPC is in the SFP or while water is in the MPC the boron concentration will continue to be verified to be within the applicable limits every 48 hours. This reflects the premise that normally there is no real need to re-verify the boron concentration of the water in the MPC after it is removed from the SFP unless water is to be added to, or recirculated through the MPC, because these are the only credible activities that could potentially change the dissolved boron concentration during this time. The 48-hour Completion Time for the re-verification is infrequent enough to prevents the interference of unnecessary sampling activities while lid closure welding and other MPC storage.preparation activities are taking place in an elevated radiation area atop the MPC. However, it is often enough to ensure that any change in the concentration for any reason is detected in a reasonable time to take proper action. Plant procedures shall specifically ensure that any water to be added to, or recirculated through the MPC is at a dissolved boron concentration greater than or equal to the minimum boron concentration specified in the LCO.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2

For UNLOADING OPERATIONS, this means verifying the source of borated water to be used to reflood the MPC within 8 hours prior to commencing reflooding operations. This ensures that when the LCO is applicable (upon introducing water into the MPC), the LCO will be met.

The use of two independent measurements provides reasonable assurance that the dissolved boron LCO limit is met and maintained in the source of water. The 8 hours limitation is considered a reasonably short time period which minimizes any potential for changes in the critical dissolved boron concentration in the source of water prior to introduction into the MPC and still allows flexibility in the operation. Once the dissolved boron concentration has been verified a change in this concentration is not credible unless there is some action specifically taken to modify it. During the period between verification and introducing the water into the MPC all changes in water source or volume including additions or subtractions in the source; or the addition or dilution of the dissolved boron concentration in the source will be administratively controlled. If any of these actions or operations takes place during the 8-hour period, the dissolved boron concentration in the source water will be re-verified prior to introducing any water into the MPC to be unloaded.

In addition, while the MPC to be unloaded is in the SFP or while water is in the MPC to be unloaded the dissolved boron concentration will continue to be verified to be within the applicable limits every 48 hours. This reflects the premise that normally there is no real need to re-verify the dissolved boron concentration of the water in the MPC unless water is to be added to, or recirculated through the MPC, because these are the only credible activities that could potentially change the dissolved boron concentration during this time.

The 48-hour Completion Time for the re-verification is infrequent enough to prevent the interference of unnecessary sampling activities while MPC UNLOADING OPERATIONS are taking place in an elevated radiation area atop the MPC. However, it is often enough to ensure that any change in the concentration for any reason is detected in a reasonable time to take proper action. Plant procedures shall specifically ensure that any water to be added to, or recirculated through the MPC is at a dissolved boron concentration is greater than or equal to the minimum dissolved boron concentration specified in the LCO.

REFERENCES

1. Diablo Canyon ISFSI SAR Section 4.2

ATTACHMENT D

TRAINING PROGRAM

TRAINING PROGRAM

Pursuant to 10 CFR 72.190 and 10 CFR 72.192, all 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 Diablo Canyon Power Plant (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 spent fuel dry cask storage. (Holtec International will participate in the initial development and presentation of the supplemental training material.)

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. The training modules include the following elements:

(1) HI-STORM 100 System design (overview)

(2) ISFSI facility design (overview)

(3) ISFSI systems, structures and components important to safety (overview)

(4) Nuclear engineering principles involved in safe handling and storage of spent fuel

(5) ISFSI licensing basis, Technical Specifications, and Certificate of Compliance conditions

(6) ISFSI security

(7) ISFSI communications

(8) ISFSI operations, emergency, maintenance and administrative procedures, including the following:

a. Procedural overview.
b. Receiving and inspecting HI-STORM 100 System components.
c. Crane and rigging operations, including heavy load requirements.
d. Loading preparations (including inspections and tests).
e. Loading spent fuel into an MPC.
f. Closing an MPC (including welding, leak testing, dewatering, vacuum drying, and helium backfilling).
g. Loading the transfer cask onto the transporter.

TRAINING PROGRAM

h. Operating the transporter.
i. Loading an MPC into the overpack at the cask transfer facility (CTF).
j. Positioning loaded overpack onto the ISFSI pad.
k. Surveillance activities.
l. Decontamination techniques.
m. Off-normal and accident conditions, responses, and corrective actions.

Following completion of the Diablo Canyon ISFSI Training Program, trainees are evaluated via written and practical exams to ensure they understand the important aspects of the information described above. Retention of training records and certificates of proficiency is consistent with that for personnel involved in fuel handling operations.

Training records are maintained in accordance with License Application Attachment E, "Quality Assurance Program." Such records include dates and hours of training and other documentation on training subjects, information on physical requirements, job performance statements, copies of written examinations, information pertaining to walk-through examinations, and retesting.

Personnel involved in spent fuel and cask handling are subject to DCPP health and safety administrative controls to ensure their physical condition and general health meet the requirements of 10 CFR 72.194.

ATTACHMENT E

QUALITY ASSURANCE PROGRAM

DIABLO CANYON ISFSI
LICENSE APPLICATION
SYNOPSIS OF ISFSI-RELATED CHANGES
QUALITY ASSURANCE PROGRAM

Section	Page	Change
General	-	The QA Program applies to ISFSI activities as specified in Section 17.2. Throughout the QA Program, ISFSI was added where "plant" was specified.
17.1	17.1-1	Describes ISFSI activities to which positions identified in Figure 17.1-2 are responsible for.
	17.1-2 17.1-3	Added ISFSI responsibilities for various officers and directors.
	17.1-4	Added ISFSI responsibilities for the Director, Nuclear Quality, Analysis, and Licensing (NQAL).
	17.1-5	Added items that the Director, NQAL is responsible for reviewing.
	17.1-6	Added ISFSI responsibilities for the President's Nuclear Advisory Committee, Nuclear Safety Oversight Committee (NSOC), and the Plant Staff Review Committee (PSRC).
17.2	17.2-1 17.2-2	Added ISFSI activities that the QA Program applies to.
	17.2-2	ISFSI changes that constitute a change to the QA Program will be submitted in a periodic update as required by 10 CFR 72.70.
	17.2-4	Added ISFSI proposed changes that NSOC will review.
	17.2-6	Added ISFSI proposed changes that PSRC will review.
17.3	17.3-2	Address how proposed changes or modifications to the ISFSI will be handled.
17.5	17.5-1	Requires that procedures be established as required by the Technical Specifications (TS) and other ISFSI license requirements.
17.7	17.7-1	Quality verification plans are to consider importance to ISFSI safety.
17.10	17.10-1	Inspection of ISFSI activities are to be in accordance with existing design requirements.
17.16	17.16-1	Significant conditions adverse to quality are to be evaluated for reportability to the NRC in accordance with 10 CFR 72.74, 10 CFR 72.75, and the TS.
17.17	17.17-3 17.17-4	Added ISFSI-related records to be maintained.
17.18	17.18-1	Audit schedule will reflect ISFSI activities. The audit program will include ISFSI SAR commitments.
	17.18-2	Revised items (1), (2), (3), and (11) to include ISFSI activities.
Table 17.1-1	1	Regulatory Guide 1.37 is not applicable to the ISFSI.
	4	PG&E is requesting an exemption from 10 CFR 72.72(d).
	7	BTP PCSB 9.5-1, Appendix A is not applicable to the ISFSI.
	7	Regulatory Guide 1.26 is not applicable to the ISFSI.
	8	Regulatory Guide 1.97 is not applicable to the ISFSI.
Figure 17.1-2		Added the ISFSI Program Manager, who reports to the Vice President, Nuclear Services.

CHAPTER 17

QUALITY ASSURANCE

CONTENTS

CONTENTS (Continued)

Chapter 17

TABLES

Chapter 17

FIGURES

17.1 ORGANIZATION

The Pacific Gas and Electric Company's (PG&E) efforts to assure the quality and safety of its nuclear power plants *and independent spent fuel storage installation (ISFSI)* are organized in a structured manner with clearly defined levels of authority, assignments of responsibility, and lines of communication. Assignment of responsibility for an item or activity includes responsibility for its quality. Figure 17.1-1 depicts the organizational structure of PG&E. The position of the Nuclear Quality, Analysis, and Licensing (NQAL) Department in the utility organization is shown in Figure 17.1-2.

PG&E has assumed full responsibility to its employees, stockholders, the general public, and affected governmental regulatory agencies for the establishment and execution of the Quality Assurance (QA) Program prescribed by Chapter 17 of the FSAR Update, quality-related program directives, and administrative procedures. The work of executing selected portions of the QA Program may be delegated to organizations external to PG&E; however, in all such instances, PG&E retains overall responsibility.

Specific responsibilities pertaining to quality assurance matters are assigned by the QA Program and its implementing procedures and instructions to various individuals throughout PG&E. In each instance, the assignment of a responsibility to an individual includes with it a commensurate delegation of sufficient authority that the person can, in fact, fulfill that responsibility. Unless otherwise specifically prohibited, it is understood that the functions, tasks, and activities necessary to carry out a responsibility may be delegated to and performed by other qualified individuals. All delegations of functions, tasks, activities, and authority shall be documented.

Figure 17.1-2 identifies those individuals and organizational components of PG&E with direct responsibilities related to the quality of the:

- design, maintenance, and operation of PG&E's nuclear power plants, *and*
- *design, fabrication, construction, testing, operation, maintenance, modification, and decommissioning of ISFSI structures, systems, and components that are important to safety.*

The narrative description throughout this section is based primarily on Figure 17.1-2.

THE BOARD OF DIRECTORS OF PG&E CORPORATION are responsible for all facets of PG&E's utility business.

THE CHAIRMAN OF THE BOARD is accountable to the Board of Directors and establishes the corporate policies, goals, and objectives related to all of PG&E's activities and operations.

THE PRESIDENT AND CHIEF EXECUTIVE OFFICER is responsible for and directs the planning, distribution, and development of all the Company's energy resources and nuclear power generation. These functions include such activities as planning and development, engineering, construction, and fossil and nuclear power plant *and ISFSI* operations. Reporting to the President and Chief Executive Officer are the Senior Vice President and Chief of Utility Operations; the Senior Vice President, Generation and Chief Nuclear Officer; and the Senior Vice President, Governmental and Public Relations. In addition, the President's Nuclear Advisory Committee (PNAC) reports to the President and Chief Executive Officer.

THE SENIOR VICE PRESIDENT, GENERATION AND CHIEF NUCLEAR OFFICER, is responsible for the safe and efficient operation of the Company's nuclear power plants. *He is responsible for overall ISFSI safety and for taking measures needed to ensure acceptable performance of the ISFSI 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, is the corporate officer specified by the DCPP Technical Specifications *and Diablo Canyon ISFSI Technical Specifications*, Section 5, who shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety. Reporting directly to him are the Vice President, Diablo Canyon Operations and the Vice President, Nuclear Services. The Senior Vice President, Generation and Chief Nuclear Officer, or his designee, as specified in administrative procedures, approves and signs official company correspondence to the U.S. Nuclear Regulatory Commission (NRC) or its representatives. The Nuclear Safety Oversight Committee (NSOC) and the Diablo Canyon Plant Staff Review Committee (PSRC) report to the Senior Vice President. He approves revisions to the QA Program as described in Chapter 17 of the FSAR Update that constitute a reduction in a commitment made to the NRC. He also approves revisions to program directives.

THE VICE PRESIDENT, DIABLO CANYON OPERATIONS, is responsible for the conduct of all onsite activities related to the safe and efficient maintenance and operation of the plant *as well as activities related to ISFSI operation and decommissioning*. He is responsible to develop, and has been delegated the necessary authority to approve and direct the implementation of, those programs, procedures, and instructions required for the operation of the plant *and ISFSI*, within limits established by the QA Program, Technical Specifications, and administrative guidelines established by the Senior Vice President, Generation and Chief Nuclear Officer. Reporting directly to the Vice President, Diablo Canyon Operations are the Station Director; the Director, Site Services; and the Director, Outage Management.

THE STATION DIRECTOR is responsible for overall safe operation of the plant and has control over those onsite activities necessary for safe operation and maintenance of the plant *and ISFSI*. The Station Director is the plant manager specified in the DCPP Technical Specifications *and Diablo Canyon ISFSI Technical Specifications*, Section 5. Reporting directly to the Station Director are the Director, Maintenance Services and the Manager, NPG Learning Services.

THE VICE PRESIDENT, NUCLEAR SERVICES, is responsible for providing quality assurance oversight, engineering and design services, safety assessments, and licensing services for DCPP, Humboldt Bay Power Plant (HBPP), *and ISFSI preoperational activities*. He is responsible for procurement of material and equipment for the plant *and ISFSI*. Reporting directly to the Vice President, Nuclear Services are the Director, NQAL; the Director, Engineering Services; the Director and Plant Manager, HBPP; the Manager, Procurement Services; the Manager, Nuclear Fuels Purchasing; *and the ISFSI Program Manager*. The Vice President also serves as the Chairman of the NSOC.

THE DIRECTOR, ENGINEERING SERVICES, is responsible for technical aspects of the engineering and design of company nuclear power plant *and ISFSI* systems, structures, and components. He is also responsible for the specification of technical and quality requirements for the purchase of material and equipment. He is also responsible for monitoring system performance and trends, implementation of the maintenance rule, evaluation of industry operating experience, and for reporting trend and performance status information to the Senior Vice President, Generation and Chief Nuclear Officer. In addition, he is specifically charged with development, evaluation, qualification, testing, and improvement of nondestructive examination procedures required by PG&E and for evaluation of these types of procedures that are used at DCPP by other organizations.

THE DIRECTOR, NUCLEAR QUALITY, ANALYSIS, AND LICENSING, is responsible for management of the QA Program and for assuring that the QA Program prescribed by Chapter 17 of the FSAR Update, program directives, and administrative procedures is effectively implemented and complied with by all involved organizations, both internal and external to PG&E. The Chairman of the Board; the President and Chief Executive Officer; the Senior Vice President, Generation and Chief Nuclear Officer; and the Vice President, Nuclear Services, have given him the organizational freedom and delegated the requisite authority to investigate any area or aspect of PG&E's operations as necessary to identify and define problems associated with establishment or execution of the QA Program. They have also delegated to him the authority to initiate, recommend, or provide solutions for such problems to whatever management level is necessary, and to verify that effective corrective action is taken in a timely manner. This delegation includes the authority to assess, audit, and monitor the conduct of quality related activities performed by or for PG&E to assure compliance with the QA Program and other regulatory requirements.

The Director, NQAL, has access to the Senior Vice President, Generation and Chief Nuclear Officer; the Vice President, Nuclear Services; the Vice President, Diablo Canyon Operations; the Station Director; and appropriate directors and managers for any significant quality-related problem or deficiency. He is authorized to prescribe a uniform company-wide method of performing an activity affecting quality by sponsoring or requiring the issuance of procedures when such standardization is considered desirable or essential to the effectiveness of the QA Program. Such uniform methods are contained in program directives and administrative procedures, and compliance with their requirements by all PG&E personnel is mandatory.

The Director, NQAL, will not be responsible for any activities unrelated to responsibilities described in the QA Program that would prevent the required attention to QA matters. Further, the responsibility of the implementation of the QA Program will take precedence over the other non-QA duties.

The Director, NQAL, shall meet the following qualification requirements: management experience through assignments to responsible positions; knowledge of QA regulations, policies, practices, and standards; and experience working in QA or related activity in reactor design, construction, or operation or in a similar highly technological industry. At the time of initial core loading or assignment to the active position, the Director, NQAL, shall have six years experience in implementing quality assurance, preferably at an operating nuclear plant, or nuclear power plant experience in the overall implementation of the QA Program. A minimum of one year of this six-year experience requirement shall be related technical or academic training. A maximum of four years of this six-year experience requirement may be fulfilled by related technical or academic training.

The Director, NQAL, is responsible to regularly assess and report on the status, adequacy, and effectiveness of PG&E's QA Program to the Senior Vice President, Generation and Chief Nuclear Officer, NSOC, and other affected PG&E Management. He is responsible to identify, prepare, and submit for approval such changes to Chapter 17 of the FSAR Update as are necessary to maintain the Program up to date and in conformance with current regulatory requirements and PG&E commitments to the NRC. He is responsible for the review of all regulatory submittals as they pertain to the QA Program, and his concurrence is required prior to submittal. He is responsible for assuring that the QA Program is effectively implemented at the plant *and ISFSI* site. He assures timely and effective corrective actions through regular assessments, trend and status reports, and root cause analysis assistance. Reporting to the Director are the Manager, NQS Operations, Corrective Action and Plant Support; the Manager, NQS Engineering, Procurement, and Maintenance; the Manager, Regulatory Services; the Supervisor, System Transient Analysis; the Supervisor, Probabilistic Risk Assessment; and the Supervising Engineer, Nuclear Safety Employee Concerns Program.

The Manager, NQS Engineering, Procurement, and Maintenance, and Manager, NQS Operations, Corrective Action, and Plant Support, report to the Director, NQAL and are responsible for providing recommendations on solutions to quality problems and performing monitoring, assessments, and audits for the areas of licensing, probabilistic risk assessment (PRA), and transient analysis.

For onsite independent review issues involving the licensing, system transient analysis, and PRA areas, the Manager, NQS Engineering, Procurement, and Maintenance, and Manager, NQS Operations, Corrective Action and Plant Support, have the authority to directly report to and communicate with the Vice President, Nuclear Services.

In the event of a conflict between any NQS manager and any other non-QA activity reporting to the Director, NQAL, the Director, NQAL, will delegate his authority to resolve the conflict to the appropriate NQS manager. The Manager, NQS Engineering, Procurement, and

Maintenance, and the Manager, NQS Operations, Corrective Action and Plant Support, have the authority to report directly to the Vice President, Nuclear Services.

The Director, NQAL, has the authority and responsibility to stop work should there be a serious breach of any part of the QA Program, or of technical or regulatory requirements wherein public health or safety could be involved. If stopping work would involve changing a nuclear generating unit's power level or separating such a unit from the PG&E system, the concurrence of the Senior Vice President, Generation and Chief Nuclear Officer, is required.

The Director, NQAL, is responsible for review of:

(1) plant operating characteristics, plant operations, modifications, maintenance, and surveillance; and

(2) ISFSI design, fabrication, construction, testing, operation, modification, decommissioning, and related activities

to verify independently that these activities are performed correctly and that human errors are reduced as much as practicable. The NQAL organization reviews NRC correspondence, industry advisories, licensee event reports, and other sources of plant *and ISFSI* design and operating experience information that may indicate areas for improving plant *and ISFSI* safety. From these reviews, NQAL makes detailed recommendations for improving plant *and ISFSI* safety.

The Director, NQAL, is responsible for coordinating with the NRC for all NPG matters relating to obtaining, maintaining, amending, revising, and otherwise change in the nuclear plant licenses. He is also responsible for probabilistic risk assessments, transient analyses, and for providing support for the independent review groups and agencies such as NSOC, PNAC, and the Diablo Canyon Independent Safety Committee (DCISC).

THE DIRECTOR, TECHNICAL AND ECOLOGICAL SERVICES, is responsible to the Vice President, General Services, for providing technical investigations, tests, analyses, examinations, and calibration services in support of Diablo Canyon and Humboldt Bay Power Plants. He also provides environmental, radiological, and health physics investigations, analyses, monitoring, and mitigation services. In addition, he is specifically charged with development, evaluation, qualification, testing, and improvement of welding, brazing, and heat-treating procedures required by the company and evaluation support of these procedures.

Upon request, he is further responsible for providing supplier source verification (e.g., quality source surveillance and inspection) services in support of nuclear procurement. While performing the source verification activity, supplier quality control (SQC) personnel functionally report to NQS Engineering, Procurement, and Maintenance within NQAL. SQC source verification personnel qualifications and procedures are approved by NQS.

The following committees function at the managerial level within PG&E to provide review and audit of nuclear power plant *and ISFSI* design, maintenance, testing, and operation activities. The reporting relationships of the committees are identified in the organization chart, Figure 17.1-2.

THE PRESIDENT'S NUCLEAR ADVISORY COMMITTEE regularly assesses and reports to the President and Chief Executive Officer on the status and adequacy of the QA Program. The Committee is responsible to advise the President and Chief Executive Officer on the results of reviews of activities associated with the design, *construction, testing*, maintenance, operation, licensing, and quality assurance of PG&E's nuclear power plants *and ISFSI*. The Committee examines the activities and reports of independent review groups and agencies such as the NSOC, DCISC, and the NRC. The Committee makes recommendations to the President on those items requiring his attention. PNAC is responsible for the periodic independent audit of the QA Program and the operations and activities of NQS to assess their effectiveness and compliance with requirements. The Committee periodically reviews the Corporate Emergency Response Plan for adequacy and investigates or reviews other areas having nuclear safety significance, as directed by the President.

The membership of PNAC is controlled by the Committee Charter, which is approved by the President and Chief Executive Officer.

THE NUCLEAR SAFETY OVERSIGHT COMMITTEE reports to the Senior Vice President, Generation and Chief Nuclear Officer. The Committee is responsible for providing an independent review and audit of activities occurring during the operational phase of PG&E's nuclear power facilities. The Committee has the authority to have reviews and audits performed in such areas as *ISFSI construction*, nuclear power plant and *ISFSI operations*, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, nondestructive testing, mechanical and electrical engineering, administrative controls, security, and QA practices to independently verify that the performance of activities in these areas is satisfactory. NSOC functions, responsibilities, and meeting requirements are described in Section 17.2.

THE PLANT STAFF REVIEW COMMITTEE reports to the Senior Vice President, Generation and Chief Nuclear Officer, and is responsible to advise the Station Director on matters related to nuclear safety. The Committee is responsible for providing timely and continuing monitoring of plant operating *and ISFSI* activities to assist the Station Director in keeping aware of general plant *and ISFSI* conditions and to verify that day-to-day plant operating *and ISFSI* activities are conducted safely and in accordance with applicable administrative controls. The Committee performs periodic reviews of plant *and ISFSI* operating activities to evaluate plant *and ISFSI* operations and to plan future activities. In addition, the PSRC performs special reviews, investigations or analyses, and screens subjects of special concern as requested by NSOC. PSRC functions, responsibilities, and meeting requirements are described in Section 17.2.

Administrative procedures or charters for the above committees provide detailed responsibilities and functions for each committee, as well as their membership, authority, and reporting requirements.

Verification of conformance to established requirements (except designs) is accomplished by individuals or groups within NQS who do not have direct responsibility for performing the work being verified or by individuals or groups trained and qualified in QA concepts and practices and independent of the organization responsible for performing the task. The persons and organizations performing QA and quality control functions have direct access to management levels that assure the ability to: (a) identify quality problems; (b) initiate, recommend, or provide solutions through designated channels; and (c) verify implementation of solutions. They are sufficiently free from direct pressures for cost and schedule and have the responsibility to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming material. (The organizational positions with stop work authority are identified in the implementing procedures.) NQS reviews and documents concurrence with all procedures and instructions that define methods for implementing the QA Program.

Each organization that supports DCPP documents and maintains current a written description of its internal organization. This documentation describes the business unit or department's structure, levels of authority, lines of communication, and assignments of responsibility. Such documentation takes the form of organization charts supported by written job descriptions or other narrative material in sufficient detail that the duties and authority of each individual whose work affects quality is clear. Interfaces between organizations are described in administrative procedures or other documents controlled in accordance with the appropriate requirements of FSAR Update, Section 17.6.

The individuals assigned to the positions having a particular responsibility in program directives and administrative procedures (as described above) are the only individuals who are authorized to perform these activities. However, circumstances may arise where it is considered either necessary or desirable to have such activities, or some portion of them, actually performed by someone else. In such cases, the assigning organization retains responsibility and shall verify that the procedures and instructions to be followed in performing the work are adequate for controlling the work and meet applicable requirements. In such circumstances, the detailed procedures and instructions to be followed in performing the work are reviewed and approved by the person assigned responsibility for the work prior to the commencement of work. The purpose of such review and approval is to verify that such procedures and instructions reflect an acceptable method of performing the work and are in compliance with the requirements of the QA Program. All instances in which authority is to be delegated or support services are to be provided are documented.

Suppliers to PG&E are required to conform to the PG&E QA Program or to their own program approved by PG&E. Supplier QA Programs are required to comply with the applicable portions of both 10 CFR 50, Appendix B, and the applicable regulatory documents and industry standards identified in Table 17.1-1. The quality program is defined in the contract or similar procurement document. Suppliers to PG&E are required to document their internal organizational arrangements to the extent necessary for PG&E to assure the supplier is capable of effectively managing, directing, and executing the requirements of the procurement documents. The authority and responsibility of persons and organizations who perform activities that might affect the quality of the procured items or services shall be clearly established. The Suppliers' organizational structure, levels of authority, and functional assignments of responsibility shall be such that:

(1) The QA function of formally verifying conformance to the technical and quality requirements of the procurement documents is accomplished by qualified personnel who are independent of those who performed or directly supervised the work.

(2) Personnel who perform QA functions have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; to verify implementation of those solutions; and to control further processing of the items or services until proper dispositioning has occurred.

17.2 QUALITY ASSURANCE PROGRAM

17.2.1 Program Applicability

The quality of the:

- safety-related aspects of the design, construction, and operation of PG&E nuclear power plants, and

- *important-to-safety aspects related to the design, fabrication, construction, testing, operation, maintenance, modification, and decommissioning of ISFSI structures, systems, and components*

shall be assured through the QA Program prescribed by this chapter, quality-related program directives, and administrative procedures. The QA Program requirements, as a minimum, apply to those structures, systems, and components classified as Design Class I in Section 3.2 of the FSAR Update. *The major Diablo Canyon ISFSI structures, systems, and components are classified, in ISFSI SAR Section 4.5, as important to safety in order to standardize the QA control applied to activities involving spent fuel storage systems.*

The QA Program also applies to the following:

(1) The design, construction, and operation of structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. The structures, systems, and components that serve these functions are classified as Design Class I. In addition, certain QA Program requirements apply to the nonsafety-related programs listed in (1) through (10) below to provide additional assurance that these objectives are satisfied.

(2) The design, construction, and operation of those portions of structures, systems, or components whose function is not required as above but whose failure could reduce the functioning of the above plant features to an unacceptable level or could incapacitate control room occupants. Certain of these structures, systems, and components are conservatively designated as Design Class I. Other nonsafety-related structures, systems, and components with seismic qualification requirements are subject to the seismic configuration control program listed below. Seismically Induced System Interaction Program requirements are governed by quality-related procedures.

(3) Activities affecting the above plant features.

(4) Managerial and administrative controls to ensure safe operation of the ISFSI, both prior to issuance of a license and throughout the life of the licensed activity.

(5) *Activities that provide confidence that an ISFSI SSC will perform satisfactorily in service, including activities that determine that physical characteristics and quality of materials or components adhere to predetermined requirements.*

In addition, the QA Program includes requirements that apply to nonsafety-related programs for:

(1) Fire Protection

(2) Emergency Plan

(3) Security

(4) Radiation Protection

(5) Environmental Monitoring

(6) Radioactive Waste Management

(7) Fitness for Duty

(8) Regulatory Guide 1.97, Category 2 and 3 Instrumentation

(9) Seismic Configuration Control

(10) Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) Equipment

17.2.2 Program Control

The status and adequacy of this QA Program shall be regularly monitored, and it shall be revised as necessary to improve its effectiveness or to reflect changing conditions.

The Director, NQAL, is responsible for the preparation, issue, interpretation, and control of this chapter, and for concurring with changes to quality-related program directives and administrative procedures that propose a change to the QA Program as it is described in a commitment to a regulatory agency. The Director, NQAL, is responsible to assure the requirements set forth in this chapter, quality-related program directives, and administrative procedures are in compliance with current regulatory requirements and PG&E commitments to the NRC as shown in Table 17.1-1. Proposed changes to program directives are also approved by the Senior Vice President, Generation and Chief Nuclear Officer.

The QA Program documents, including any changes, supplements, or appendices, are issued and maintained as controlled documents. Changes to the QA Program as described in this chapter that do not reduce commitments shall be included in the periodic updates required by

10 CFR 50.71 *and 10 CFR 72.70*. Proposed changes to this chapter that reduce commitments are reviewed and concurred with in writing by the Director, NQAL, are reviewed by PNAC, and are approved by the Senior Vice President, Generation and Chief Nuclear Officer, or his designee, prior to being submitted to and approved by the NRC in accordance with 10 CFR 50.54 prior to issue for use.

Implementation of the QA Program is accomplished through separately issued procedures, instructions, and drawings. Each vice president and director is responsible for the establishment and implementation of detailed procedures and instructions prescribing the activities for which he is responsible. Such documents are derived from the requirements and reflect the responsibilities specified in the QA Program. Activities affecting quality are accomplished in accordance with these instructions, procedures, and drawings. All personnel are instructed that compliance with those requirements, and the requirements of the QA Program, is mandatory.

Questions or disputes involving interpretations of QA Program requirements, or of the commitments and requirements upon which it is based, are referred to the Director, NQAL, for resolution. Questions or disputes involving the responsibilities defined in this chapter and program directives are referred to the Senior Vice President, Generation and Chief Nuclear Officer. Questions or disputes involving other quality matters are resolved by referring the matter in a timely manner to successively higher levels of management until, if necessary, the matter reaches that level which has direct authority over all contesting parties.

Personnel who perform functions addressed by the QA Program are responsible for the quality of their work. They are indoctrinated, trained, and appropriately qualified to assure that they have achieved and maintained suitable proficiency to perform those functions. Qualifications of such personnel are in accordance with applicable codes, standards, and regulatory requirements.

The Director, NQAL, or his designated representative, regularly reports to the Senior Vice President, Generation and Chief Nuclear Officer, responsible company management, and the NSOC on the effectiveness of the QA Program as it relates to *plant and ISFSI* design, maintenance, and operation of DCPP. Such reports are based on the results of audits, inspections, tests, and other observations of activities as prescribed by the QA Program.

The PNAC regularly assesses and reports to the President on the overall status and adequacy of PG&E's QA Program for nuclear power plants *and ISFSI*. Such assessments shall include overview of the NSOC and effectiveness of PG&E's QA Program.

Annually, the Director, NQAL, shall report to the PNAC on the effectiveness of the QA Program and NQAL activities and operations. The assessment shall include an evaluation of NPG's compliance with current regulatory requirements and commitments to the NRC.

17.2.3 Nuclear Safety Oversight Committee

The QA Program also includes an independent review and audit function, directed by the NSOC. The NSOC shall function to provide independent review and audit of designated activities in the areas of:

(1) Nuclear power plant *and ISFSI* operations

(2) Nuclear engineering

(3) Chemistry and radiochemistry

(4) Metallurgy

(5) Instrument and control

(6) Radiological safety

(7) Mechanical and electrical engineering

(8) Quality assurance practices

NSOC shall report to and advise the Senior Vice President, Generation and Chief Nuclear Officer on those areas of responsibility specified in the Review and Audits sections below.

Composition - NSOC shall be composed of a chairman and a minimum of four members. The NSOC Chairman and members shall be appointed in writing by the Senior Vice President, Generation and Chief Nuclear Officer. The NSOC Chairman shall have a minimum of six years of professional level managerial experience in the power field, and NSOC members shall have a minimum of five years of professional level experience in the field of their specialty. The NSOC Chairman and all members shall have qualifications that meet or exceed the requirements and recommendations of Section 4.7 of ANSI/ANS 3.1-1978.

Consultants - Consultants shall be used as determined by the NSOC Chairman to provide expert advice to NSOC.

Meeting Frequency - NSOC shall meet at least once per 6 months.

Quorum - A quorum of NSOC is necessary for the performance of the NSOC functions of this FSAR Update section and shall be a majority (one-half or more) of the members, but no less than four. No more than a minority of the quorum shall have line responsibility for operation of the plant.

Review - NSOC shall review:

(1) The evaluations for: (a) changes to procedures, equipment or systems, and (b) tests or experiments completed under the provision of 10 CFR 50.59 *or 10 CFR 72.48*, to verify that such actions did not require prior NRC approval

(2) Proposed changes to procedures, equipment, or systems that require prior NRC approval as defined in 10 CFR 50.59 *or 10 CFR 72.48*

(3) Proposed tests or experiments that require prior NRC approval as defined in 10 CFR 50.59 *or 10 CFR 72.48*

(4) Proposed changes to Diablo Canyon Power Plant's Technical Specifications or Operating License

(5) Proposed changes to the ISFSI Technical Specifications or license

(6) Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance

(7) Significant operating abnormalities or deviations from normal and expected performance of plant *and ISFSI* equipment that affect nuclear safety

(8) All reportable events

(9) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components that could affect nuclear safety

(10) All recognized indications of an unanticipated deficiency in some aspect of ISFSI design or operation of important-to-safety structures, systems, or components that could affect nuclear safety.

(11) Reports and meeting minutes of the PSRC.

Audits - Audits of plant activities shall be performed under the cognizance of NSOC. See FSAR Update Section 17.18 for minimum audit frequency details.

Records - Records of NSOC activities shall be prepared, approved, and distributed as indicated below:

(1) Minutes of each NSOC meeting shall be prepared, approved, and forwarded to the Senior Vice President, Generation and Chief Nuclear Officer, within 14 working days following each meeting.

(2) Reports of reviews encompassed by the Review section above shall be prepared, approved, and forwarded to the Senior Vice President, Generation and Chief Nuclear Officer, within 14 working days following completion of the review.

(3) Audit reports encompassed by the Audit section, above, shall be forwarded to the Senior Vice President, Generation and Chief Nuclear Officer, and to the management positions responsible for the areas audited within 30 days after completion of the audit.

17.2.4 Plant Staff Review Committee

A PSRC has been established at the plant. The committee satisfies applicable requirements of ANSI N18.7, 1976, and its activities are controlled as described below:

PSRC Function - The PSRC shall function to advise the Station Director on all matters related to nuclear safety.

Composition - The PSRC shall be chaired by the Station Director and shall be composed of a minimum of 8 senior management individuals whose responsibilities include the functional areas of operations, maintenance, radiation protection, support services, technical services, and quality control. All members shall be appointed in writing by the PSRC Chairman. The qualifications of each PSRC member shall meet or exceed the requirements and recommendations of Section 4.7 of ANSI/ANS 3.1-1978.

Alternates - The Chairman may designate in writing other regular members who may serve as the Acting Chairman of PSRC meetings. All alternate members shall be appointed in writing by the PSRC Chairman. Alternates may be designated for specific PSRC members and shall have expertise in the same general area as the regular PSRC member they represent. No more than two alternates shall participate as voting members in PSRC activities at any one time.

Meeting Frequency - The PSRC shall meet at least once per calendar month and as convened by the PSRC Chairman or his designated alternate.

Quorum - The minimum quorum of the PSRC necessary for performance of the PSRC responsibility and authority provisions of this FSAR Update section shall be a majority (more than one-half) of the members of the PSRC. For purposes of the quorum, this majority shall include the Chairman or his designated alternate and no more than two alternate members.

The PSRC shall be responsible for:

(1) Reviewing the documents listed below to verify that proposed actions do not require prior NRC approval or require a change to any Technical Specifications and recommending approval or disapproval in writing to the appropriate approval authority

(a) Evaluations of proposed procedures and procedure changes completed under the provisions of 10 CFR 50.59 *or 10 CFR 72.48*

(b) Evaluations of proposed tests or experiments completed under the provisions of 10 CFR 50.59 *or 10 CFR 72.48*

(c) Evaluations of proposed changes or modifications to plant structures, systems, or equipment completed under the provisions of 10 CFR 50.59 *or 10 CFR 72.48*

(d) Evaluations of proposed changes to the following plans and programs completed under the provisions of 10 CFR 50.59 *or 10 CFR 72.48*

1. Security Plan
2. Emergency Plan
3. Process Control Program
4. Offsite Dose Calculation Procedure
5. Environmental Radiological Monitoring Program
6. Fire Protection Program

(2) Reviewing all proposed changes to DCPP's Technical Specifications *and ISFSI Technical Specifications* and advising the Station Director on their acceptability

(3) Investigating all violations of any Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Senior Vice President, Generation and Chief Nuclear Officer, and to the Chairman of the NSOC; the assessment shall include an assessment of the safety significance of each violation

(4) Reviewing all reportable events in order to advise the Station Director on the acceptability of proposed corrective actions, and forwarding of reports covering evaluation and recommendations to prevent recurrence to NSOC and the Senior Vice President, Generation and Chief Nuclear Officer

(5) Reviewing significant plant *and ISFSI* operating experience or events that may indicate the existence of a nuclear safety hazard, and advising the Station Director on an appropriate course of action

(6) Reviewing the Security Plan and implementing procedures and submitting results and recommended changes to the Chairman of NSOC and the Station Director

(7) Reviewing the Emergency Plan and implementing procedures and submitting results and recommended changes to the Chairman of NSOC and the Station Director

(8) Reviewing any accidental, unplanned, or uncontrolled radioactive release including the preparation and forwarding of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence to the Senior Vice President, Generation and Chief Nuclear Officer, and to the Chairman of NSOC

(9) Recommending in writing to the appropriate approval authority, approval or disapproval of the items considered under paragraphs (1) and (2), above

(10) Rendering determinations in writing with regard to whether each item considered under paragraphs (1) through (4), above, require prior NRC approval

(11) Providing written notification within 24 hours to the Senior Vice President, Generation and Chief Nuclear Officer, of disagreement between the PSRC and the Station Director; however, the Station Director shall have responsibility for resolution of such disagreements

(12) Reviewing, prior to approval, new procedures used to handle heavy loads in exclusion areas and changes directly related to methods and routes used to handle heavy loads in exclusion areas.

Records - The PSRC shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of this FSAR Update section. Copies shall be provided to the Senior Vice President, Generation and Chief Nuclear Officer, and to NSOC.

17.3 DESIGN CONTROL

Design activities shall be performed in an orderly, planned, and controlled manner directed to achieving the plant *and ISFSI* design that best serves the needs of PG&E and its customers without posing an undue risk to the health and safety of the public.

Design activities shall be controlled to assure that design, technical, and quality requirements are correctly translated into design documents and that changes to design and design documents are properly controlled. Design control procedures shall address responsibilities for all phases of design including:

(1) Responsibilities

(2) Interface control

(3) Design input

(4) Design performance

(5) Design verification

(6) Design change

Systematic methods shall be established and documented for communicating needed design information across the external and internal design interfaces, including changes to the design information, as work progresses. The interfaces between the Engineering Services Department and other organizations, either internal or external to PG&E, performing work affecting quality of design shall be identified and documented. This identification shall include those organizations providing criteria, designs, specifications, technical direction, and technical information and shall be in sufficient detail to cover each structure, system, or component and the corresponding design activity.

Provisions for design input shall define the technical objectives for structures, systems, and components being designed or analyzed. For the structure, system, or component being designed, or for the design services being provided (for example, design verification), design input requirements shall be determined, documented, reviewed, approved, and controlled.

Required design analyses (such as physics, stress, thermal, hydraulic, and accident analysis; material compatibility; accessibility for inservice inspection, maintenance, and repair; and ALARA considerations) shall be performed in a planned, controlled, and correct manner. PG&E procedures shall identify the review and approval responsibilities for design analyses.

The preparation and control of design documents (such as specifications, drawings, reports, and installation procedures) shall be performed in a manner to assure design inputs are

correctly translated into design documents (for example, a documented check to verify the dimensional accuracy and completeness of design drawings and specifications).
PG&E shall provide for reviewing, confirming, or substantiating the design to assure that the design meets the specified design inputs. Design verification shall be performed by competent individuals or groups other than those who performed the original design, but who may be from the same department. Individuals performing the verification shall not:

(1) Have immediate supervisory responsibility for the individual performing the design. In exceptional circumstances, the designer's immediate supervisor can perform the verification provided:

   (a) The supervisor is the only technically qualified individual.

   (b) The need is individually documented and approved in advance by the supervisor's management.

   (c) NQS quality assurance audits cover frequency and effectiveness of use of supervisors as design verifiers to guard against abuse.

(2) Have specified a singular design approach.

(3) Have ruled out certain design considerations.

(4) Have established the design inputs for the particular design aspect being verified.

The results of the design verification efforts shall be documented with the identification of the verifier clearly provided. Design verification methods may include, but not be limited to, the following: design reviews, use of alternate calculations, and qualification testing. The design verification shall be identified and documented. The design verification shall be completed prior to relying upon the component system or structure to perform its function. Procedures shall assure that verified computer codes are certified for use and that their applicability is specified.

Proposed changes or modifications to the *ISFSI or* plant systems or equipment that affect nuclear safety shall be designed by a qualified individual or organization, and reviewed by a qualified individual/group other than the individual/group who prepared the change or modification, but who may be from the same organization. These reviews shall include a determination as to whether additional cross-discipline reviews are necessary. If deemed necessary, they shall be performed by review personnel of the appropriate discipline(s). These reviews shall also determine whether an evaluation per 10 CFR 50.59 *or 10 CFR 72.48* is necessary. If necessary, one shall be prepared and presented to the PSRC for review prior to approval.

Each change or modification shall be approved by the Station Director or his designee, as specified in administrative procedures, prior to implementation.

Procedures for implementing design changes, including field changes, shall assure that the impact of the change is carefully considered, required actions documented, and information concerning the change transmitted to all affected persons and organizations. These changes shall be subjected to design control measures commensurate with those applied to the original design. Design changes shall be reviewed and approved by the same organization or group that was responsible for the original design.

Document control measures shall be established for design documents that reflect the commitments of the FSAR Update *and ISFSI SAR*. These design documents shall include, but are not limited to, specifications, calculations, computer programs, system descriptions, the FSAR Update *and ISFSI SAR* when used as a design document, and drawings including flow diagrams, piping and instrument diagrams, control logic diagrams, electrical single line diagrams, structural drawings for major facilities, site arrangements, and equipment locations.

Nonconforming activities such as procedure violations, deviations, or errors and deficiencies in approved design documents, including design methods (such as computer codes), shall be controlled as described in Sections 17.15 and 17.16.

17.4 PROCUREMENT DOCUMENT CONTROL

The procurement documents shall include those requirements necessary to assure that the items and services to be provided will be of the desired quality.

The procurement documents shall also include provisions for the following, as appropriate:

(1) *Basic Technical Requirements* - These include drawings, specifications, codes, and industrial standards with applicable revision data; test and inspection requirements; and special instructions and requirements, such as for designing, fabricating, cleaning, erecting, packaging, handling, shipping, and, if applicable, extended storage in the field.

(2) *Quality Assurance Requirements* - These include the requirements for the supplier to have an acceptable QA Program; provisions for access to the supplier's facilities and records for source inspection and audit when the need for such inspection and audit has been determined; and provisions for extending applicable QA Program and other requirements of procurement documents to subcontractors and suppliers, including PG&E's access to facilities and records.

(3) *Documentation Requirements* - These shall include records to be prepared, maintained, submitted or made available for review and instructions on record retention and disposition.

The procedures that implement procurement document control shall describe the organizational responsibilities for procurement planning; preparation, review, approval and control of procurement documents; supplier selection; bid evaluations; and review and evaluation of supplier QA Programs prior to initiation of activities affected by the program.

Procedures shall be established to review the adequacy of technical and quality assurance requirements stated in procurement documents; determine that requirements are correctly stated, inspectable, and controllable; assure adequate acceptance and rejection criteria; and provide for the preparation, review, and approval of procurement documents in accordance with QA Program requirements. The review and documented concurrence of the adequacy of quality assurance requirements stated in procurement documents shall be performed by independent personnel trained and qualified in applicable QA practices and concepts.

Changes to procurement documents shall be subject to the same control as the original document.

17.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Activities affecting quality shall be prescribed by and accomplished in accordance with documented procedures, instructions, and drawings.

The Vice President in charge of each PG&E organizational unit that performs activities affecting quality is responsible for the establishment and implementation of instructions, procedures, or drawings prescribing such activities. Standard guidelines for the format, content, and review and approval processes shall be established and set forth in a procedure or instruction issued by that organizational unit.

The method of performing activities affecting quality shall be prescribed in documented instructions, procedures, or drawings of a type appropriate to the circumstances. This may include shop drawings, process specifications, job descriptions, planning sheets, travelers, QA manuals, checklists, or any other written or pictorial form provided that the activity is described in sufficient detail such that competent personnel could be expected to satisfactorily perform the work functions without direct supervision.

Within the constraints, limitations, or other conditions as may be imposed by the specific plant Technical Specifications and other license requirements or commitments, procedures prescribing a preplanned method of conducting the following aspects of plant operations shall be established in accordance with the applicable regulations, codes, standards, and specifications: preoperational tests, systems operations, general plant activities, startup, shutdown, power operations and load changing, process monitoring, fuel handling, maintenance, modifications, radiation control, calibrations and tests, chemical-radiochemical control, abnormal or alarm conditions, emergency plan, tests and inspections, emergencies, and significant events.

Within the constraints, limitations, or other conditions as may be imposed by the ISFSI Technical Specifications and other ISFSI license requirements or commitments, procedures prescribing a preplanned method of conducting the activities and programs specified shall be established in accordance with the applicable regulations, codes, standards, and specifications.

In addition to the above, plant *and ISFSI* procedures and programs shall be established and controlled as described below.

(1) Written procedures shall be established, implemented, and maintained covering the activities referenced in *the ISFSI Technical Specifications and* Specification 5.4.1 of the Diablo Canyon Power Plant's Technical Specifications.

(2) Each procedure of paragraph (1) above, and changes thereto, and all proposed tests or experiments that affect nuclear safety shall be reviewed and approved prior to implementation in accordance with the review and approval requirements below. Each

procedure of paragraph (1) above, as modified by Table 17.1-1, shall also be reviewed periodically as set forth in administrative procedures.

These procedure review and approval requirements apply when approving plant *and ISFSI* programs and procedures, or changes to those programs and procedures. They also apply when approving or changing corporate procedures and procedures used by support organizations if they could have an immediate effect on plant *and ISFSI* operations or the operational status of plant safety-related structures, systems, or components *and ISFSI structures, systems, or components; and ISFSI structures, systems, or components which are important-to-safety*. They do not apply to editorial or typographical changes.

(3) Each procedure or program required by paragraph (1) above, and other procedures, tests, and experiments that affect nuclear safety or the treatment of radwaste, and changes thereto, shall be prepared by a qualified individual/group. Each procedure, program, test, or experiment, and changes thereto, shall be reviewed by an individual/group other than the individual/group who prepared the proposed document or change, but who may be from the same organization as the individual/group who prepared it, and shall be approved, prior to implementation, by the Station Director or his designee, as identified in administrative procedures.

(4) A responsible organization shall be assigned for each program or procedure required by paragraph (1) above. The responsible organization shall assign reviews of proposed procedures, programs, and changes to qualified personnel of the appropriate discipline(s).

(5) Individuals responsible for the above reviews shall be knowledgeable in the document's subject area, shall meet or exceed the qualification requirements of Section 4.7.2 of ANSI/ANS 3.1-1978, and shall be designated as qualified reviewers by the Station Director or his designee.

(6) The reviews specified in paragraph (3) above shall include a determination as to whether additional cross-discipline reviews are necessary. If deemed necessary, they shall be performed by review personnel of the appropriate discipline(s).

(7) The reviews specified in paragraph (3) above shall also determine whether an evaluation per 10 CFR 50.59 *or 10 CFR 72.48* is necessary. If necessary, one shall be prepared and presented to the PSRC for review prior to approval.

(8) Temporary changes to procedures of paragraph (1) above may be made provided:

(a) The intent of the original procedure is not altered

(b) The change is approved by at least two management staff members who meet applicable qualification requirements of ANSI/ANS 3.1, 1978, and are knowledgeable in the subject area of the procedure. For changes to procedures listed below, at least one approver shall hold a Senior Reactor Operators license.

1. All Operations Section procedures

2. Surveillance Test Procedures

3. Emergency Plan Implementing Procedures

4. Any other procedure if the proposed change affects equipment or system operating status

If the approving Senior Reactor Operator is not the Shift Foreman of the affected unit, that individual shall determine whether the Shift Foreman should be notified of the change immediately, and shall notify him/her if appropriate.

(c) The change is documented, reviewed as described above, and approved by the appropriate approval authority within 14 days of implementation.

17.6 DOCUMENT CONTROL

Documents and changes to documents that prescribe or verify activities affecting quality shall be controlled in a manner that precludes the use of inappropriate or outdated documents. As a minimum, controlled documents include: design documents, including documents related to computer codes; procurement documents; instructions and procedures for such activities as fabrication, construction, modification, installation, test, operation, maintenance, and inspection; as-built documents; quality assurance and quality control manuals and quality-affecting procedures; FSAR Update; *ISFSI SAR*; and nonconformance reports.

The organization responsible for establishing instructions, procedures, drawings, or other documents prescribing activities affecting quality is also responsible to develop and implement systematic methods for the control of such documents in accordance with the requirements herein. In those instances where such documents directly involve organizational interfaces, that organization with ultimate responsibility for the issuance of the documents is responsible for establishing the methods for their control.

Procedures and instructions shall assure that documents, including changes, are prepared; reviewed by a qualified individual other than the person who generated the document; approved for release by authorized personnel; distributed to the location where the activity is performed prior to commencing work; and used in performing the activity. Procedures and instructions shall require the development of as-built drawings and the removal or appropriate identification of obsolete or superseded documents.

Procedures and instructions that define methods for implementing the QA Program requirements shall be reviewed and concurred with by NQS for compliance and alignment with the Program. Revisions to these documents shall also be reviewed and concurred with by NQS if they propose a change to the QA Program as it is described in a commitment to a regulatory agency.

The controls shall identify those responsible for preparing, reviewing, approving, and issuing documents to be used. They shall also define the coordination and control of interfacing documents and shall require the establishment of current and updated distribution lists.

A document control system shall be established to identify the current revision of instructions, procedures, specifications, drawings, and procurement documents. Master lists, when utilized as an element of the document control system, shall be updated and distributed to predetermined responsible personnel.

17.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

Supplier activities in providing purchased material, equipment, and services shall be monitored as planned and necessary to assure such items and services meet procurement document requirements.

Procedures shall describe each organization's responsibilities for the control of purchased material, equipment, and services, including the interfaces between all affected organizations.

All materials, equipment, and services shall meet the specified technical and quality requirements. Verification that a supplier can meet the specified technical and quality requirements shall be by one or a combination of the following:

(1) Evaluation of the supplier's history

(2) Evaluation of current supplier quality records

(3) Evaluation of the supplier's facilities, personnel, and implementation of a QA Program

Such evaluations shall be documented. Suppliers whose QA Programs have been found by NQS to satisfy specified quality requirements shall be listed on the PG&E Qualified Suppliers List, which is controlled by NQS.

A quality verification plan shall be established and documented that applies to each procurement and identifies the manner by which PG&E intends (with appropriate NQS organization involvement) to assure the quality of the material, equipment, or service as defined in the procurement documents and to accept those items or services from the supplier.

The quality verification plan shall identify inspection, audit, and/or surveillance activities to be performed including the characteristics or processes to be witnessed, inspected, or verified; the method of surveillance; and the extent of documentation required. The timing and sequence of the activities shall be planned to identify any system or product deficiencies before subsequent activities may preclude their disclosure.

The plan shall also be based on consideration of:

(1) Importance to plant *and ISFSI* safety

(2) Complexity of inspectable characteristics

(3) Uniqueness of the item or service

Supplier performance and compliance with procurement documents may be monitored by either source verification, receiving inspection, or a combination of the two.

Source verification activities may consist of inspections, audits, surveillance, or a combination thereof and are conducted at the supplier's facility. When source verification activities are specified in the quality verification plan, the timing and sequence of these activities are to be delineated.

Receiving inspection activities, as required by the quality verification plan, shall be coordinated with source verification activities performed prior to shipments. If sampling is performed, it shall be in accordance with procedures and/or recognized standards. Receipt inspection shall include a review, which verifies that supplier quality, records required by procurement documents are acceptable and that items are properly identified and traceable to appropriate documentation.

Records of quality verification activities shall be traceable to the materials, equipment, or services to which they apply. Documentation of acceptance in accordance with the procurement quality verification plan shall be available at the site prior to installation or acceptance for use. Documentary evidence that procurement document requirements have been met shall clearly reflect each requirement. Supplier's Certificates of Conformance are periodically evaluated by audits and independent inspections or tests to assure they are valid and the results documented.

When spare or replacement parts are procured, supplier selection and quality verification activities shall be planned and implemented to verify compliance with requirements meeting or exceeding those of the original.

17.8 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

Materials, parts, and components shall be identified and controlled in a manner to preclude the use of incorrect or defective items.

All materials, parts, and components, including partially fabricated subassemblies, batches, lots, and consumables, shall be identified in a manner that each can be related to its applicable drawing, specification, or other technical documentation at any stage from initial receipt through fabrication, installation, repair, or modification. Controls and implementing procedures shall ensure that only correct and accepted items are used during all stages and describe the responsibilities of the involved organizations.

Physical identification of items shall be used whenever possible and practical. Controls may, however, be through physical separation, procedure, or other appropriate means. Identification may be either on the item or on records traceable to the item.

Identification marking, where employed, shall be clear, unambiguous, and indelible and its application shall not impair the function of the identified item or any other item. When an item is subdivided, the identifying marking shall be transferred to each resulting part. Markings shall not be rendered illegible by treatment, process, assembly, installation, or coating unless other means of identification and determining acceptability are provided.

Verification activities, such as inspection, shall be performed to ensure that the provisions of this policy and related implementing procedures are followed for items prior to release for fabrication, assembly, shipping, installation, and use.

When required by code, standard, or specification, traceability of materials, parts, or components to specific inspection or test records shall be provided for and verified.

17.9 SPECIAL PROCESSES

Special processes shall be controlled and performed by qualified personnel using qualified procedures or instructions in accordance with applicable codes, standards, specifications, criteria, or other special requirements.

A special process is an activity in which the quality of the result is highly dependent upon either process variables or the skill and performance of the person doing the work, and the specified quality is difficult to verify by inspection and test after the process is completed.

Special processes include, but are not limited to:

(1) Welding

(2) Heat treating

(3) Nondestructive examination

(4) Chemical cleaning

(5) Others as specified in design and procurement documents (examples are certain protective coating applications and concrete batch plant operations, which are controlled by specifications on a case-by-case basis)

The implementing instructions shall contain the criteria for assuring proper process control and shall be qualified and controlled to assure compliance with applicable codes, standards, QA procedures, and design specifications. Substantiating records of qualifications and controls shall be maintained.

17.10 INSPECTION

A comprehensive program of inspection of items and activities affecting quality shall be conducted to verify conformance with established requirements. Procedures shall describe the organizational responsibilities necessary to carry out the inspection program.

The objective of the inspection program shall be to verify the quality of the items and activities and conformance to the applicable documented instructions, procedures, and drawings for accomplishing activities affecting quality. The inspection program, including information relative to individual inspections to be performed, shall be developed based on a review of the design drawings, specifications, and other controlled documents which prescribe items and activities affecting quality. Inspections shall be performed utilizing appropriate inspection procedures and instructions together with the necessary drawings, specifications, and other controlled documents. The inspections shall be documented and evaluated.

Inspection procedures, instructions, or checklists shall provide for the following: identification of characteristics and activities to be inspected; a description of the method of inspection; identification of the individuals or groups responsible for performing the inspection operation; acceptance and rejection criteria; identification of required procedures, drawings, and specifications and revisions; recording the name of the inspector or data recorder and the results of the inspection operation; and specifying necessary measuring and test equipment including accuracy requirements. The inspection program shall include, but not be limited to, those inspections required by applicable codes, standards, specifications, and plant *and ISFSI* Technical Specifications. The inspection program shall also require the following during the operational phase of a plant:

(1) Inspection of modifications, repairs, and replacements to be in accordance with the original design requirements or appropriately approved equivalents

(2) Verification of the cleanness of those portions of *plant* safety-related systems that have been subject to potential contamination during maintenance and modification activities through an inspection performed immediately prior to closure of the portion of the system

During the ISFSI operational phase, the inspection program shall require inspection of modifications, repairs, and replacements to be in accordance with existing design requirements.

The inspection program shall require inspection and/or test of items for each work operation where such is necessary to assure quality. If inspection of processed items is impossible or disadvantageous, indirect control by monitoring of process shall be required. Both inspection and process monitoring shall be required when control is inadequate without both. Both inspection and process control shall be performed when required by applicable code, standard, or specification.

Mandatory inspection hold points shall be identified in the inspection program. When mandatory inspection hold points are required, the specific hold points shall be indicated in the drawings, procedures, or instructions that prescribe the work activity. Work shall not proceed beyond such hold points without the documented consent of NQS.

When the inspection program permits or requires a sample of a large group of items that are amenable to statistical analysis, the sampling procedures to be used shall be based on recognized standard practices.

Inspections to verify the quality of work shall be performed by qualified individuals other than those who performed or directly supervised the activity being inspected. During the inspection, such persons shall not report directly to the immediate supervisors who are responsible for the work being inspected.

Personnel performing inspections shall be qualified in accordance with applicable regulations, codes, standards, and specifications.

Inspection records shall contain the following where applicable: a description of the type of observation, the date and results of the inspection, information related to conditions adverse to quality, inspector or data recorder identification, evidence as to the acceptability of the results, and action taken to resolve any discrepancies noted.

17.11 TEST CONTROL

A program of testing shall be conducted as necessary to demonstrate that structures, systems, and components will perform satisfactorily in service. This program shall ensure that the necessary testing is identified and performed at the appropriate time in accordance with written test procedures that incorporate or reference the requirements and acceptance limits contained in the applicable design documents.

The program shall cover all required tests, including tests prior to installation, preoperational tests, and operational tests.

The procedures that implement testing shall provide for meeting appropriate prerequisites for the test (for example, environmental conditions, specification of instrumentation, and completeness of tested item), sufficient instruction for the performance of the test, specification of any witness or hold points, acceptance and rejection criteria and limits, and the documentation of the test. The procedures shall provide for evaluation and documentation of the test results and data and their acceptability as determined by a qualified person or group.

Test records shall contain the following where applicable: a description of the type of observation, the date and results of the test, information related to conditions adverse to quality, inspector or data recorder identification, evidence as to the acceptability of the results, and action taken to resolve any discrepancies noted.

17.12 CONTROL OF MEASURING AND TEST EQUIPMENT

Organizational responsibilities shall be delineated for establishing, implementing, and assuring the effectiveness of the calibration program for measuring and test equipment (M&TE). This program shall include the generation, review, and documented concurrence of calibration procedures; the calibration of measuring and test equipment; and the maintenance and use of calibration standards.

M&TE, including reference standards, used to determine the acceptability of items or activities shall be strictly maintained within prescribed accuracy limits.

M&TE, including reference standards, shall be of suitable range, type, and accuracy to verify conformance with requirements.

Procedures for control of M&TE shall provide for the identification (labeling, codes, or alternate documented control system), recall, and calibration (including documented precalibration checks) of the M&TE. The calibration procedures shall delineate any necessary environmental controls, limits, or compensations in excess of those which may be inherent to the general program.

The calibrations shall utilize documented valid relationships to nationally recognized standards or accepted values of natural physical constants. Where national standards do not exist, the basis for the calibration shall be documented. Calibration of M&TE shall be against standards that have an accuracy of at least four times the required accuracy of the equipment being calibrated or, when this is not practical, have an accuracy that assures the equipment being calibrated will be within required tolerance and that the basis of acceptance is documented and authorized by responsible management of the PG&E organization performing that activity.

Calibrating standards have greater accuracy than standards being calibrated. Calibrating standards with the same accuracy may be used if it can be shown to be adequate for the requirements and the basis of acceptance is documented and authorized by responsible management.

The calibration intervals, whether calendar- or usage-based, shall be predetermined and documented. Indication of expiration, if feasible, will be displayed on or with the M&TE. Significant environmental or usage restrictions will be indicated on or with the equipment or be factored into the documented system used to control the issuance of the M&TE. Special calibration shall be required whenever the accuracy of the equipment is suspect.

Records shall be maintained to show that established schedules and procedures for the calibration of the M&TE have been followed. M&TE shall be identified and traceable to the calibration test data. Records of the usage of the M&TE shall be maintained to facilitate corrective action in the event of the discovery of a deficiency concerning the calibration or use of M&TE, so that measures may be taken and documented to determine the validity of previous inspections performed and of the acceptability of items inspected or tested since the previous calibration of the deficient M&TE.

17.13 HANDLING, STORAGE, AND SHIPPING

Material and equipment shall be handled, stored, and shipped in accordance with design and procurement requirements in a manner that will prevent damage, deterioration, or loss.

Special coverings, equipment, and protective environments shall be specified and provided where necessary for the protection of particular items from damage or deterioration. When such special protective features are required, their existence shall be verified and monitored as necessary to assure they continue to serve their intended function.

Special handling tools and equipment shall be provided where necessary to ensure items can be handled safely and without damage. Special handling tools and equipment shall be controlled and maintained in a manner such that they will be ready and fit to serve their intended function when needed. Such control shall include periodic inspection and testing to verify that special handling tools and equipment have been properly maintained.

Special attention shall be given to marking and labeling items during packaging, shipment, and storage. Such additional marking or labeling shall be provided as is necessary to ensure that items can be properly maintained and preserved. This shall include indication of the presence of special environments or the need for special control. Provisions shall be described for the storage of chemicals, reagents (including control of shelf life), lubricants, and other consumable materials.

Special handling, preservation, storage, cleaning, packaging, and shipping requirements are established and accomplished by suitably trained individuals in accordance with predetermined work and inspection instructions.

17.14 INSPECTION, TEST, AND OPERATING STATUS

The inspection, test, and/or operating status of material, equipment, and operating systems shall be readily apparent and verifiable.

The procedures used to indicate status shall provide means for assuring that required inspections and tests are performed in the prescribed sequence; acceptability is indicated; and nonconforming items are clearly identified throughout fabrication, installation, test, maintenance, repairs, and modification to prevent inadvertent use or operation. Items accepted and released are identified to indicate their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work. Deviations from the prescribed sequence shall be subject to the same level of control as the generation of the original sequence to prevent the bypassing or omission of a required test or inspection.

Identification of status may be by such means as, but not limited to, tags, stamps, markings, labels, or travelers. In some instances, records traceable to the item may be used. The procedures implementing control of inspection, test, and operating status shall clearly delineate authority for the application, change, or removal of a status identifier.

17.15 CONTROL OF NONCONFORMING CONDITIONS

Items and activities that do not conform to requirements shall be controlled in a manner that will prevent their inadvertent use or installation. Technical decisions as to the disposition of each nonconforming condition shall be made by personnel with assigned authority in the relevant disciplines. The control, review, and disposition of nonconforming conditions shall be accomplished and documented in accordance with approved written procedures and instructions.

Nonconforming conditions shall be documented and affected organizations notified of such conditions. Further processing of the nonconforming conditions and other items affected by them shall be controlled in a manner to prevent their inadvertent use or installation pending a decision on their disposition.

The responsibility and authority for the disposition of nonconforming conditions shall be established and set forth in the applicable procedures and instructions for their control. The rework or repair of nonconforming items and the disposition of operational nonconforming conditions shall be accomplished in accordance with written procedures and instructions. Dispositions involving design changes shall be approved by the organization with the authority for design.

The acceptability of rework or repair of materials, parts, components, systems, or structures shall be verified by reinspecting and retesting the item as originally inspected and tested, or by a method that is at least equal to the original inspection or testing method. Reworked and repaired items shall be reinspected in accordance with applicable procedures and instructions. The acceptability of nonconforming items that have been dispositioned "repair" or "accept-as-is" shall be documented. Such documentation shall include a description of the change, waiver, or deviation that has been accepted in order to record the change and, if applicable, denote the as-built condition.

When all actions associated with the disposition have been completed, personnel who have no direct responsibility for the disposition shall review and verify satisfactory completion.

In cases where required documentary evidence that items have passed required inspections and tests is not available, the associated materials or equipment shall be considered nonconforming. Until suitable documentary evidence is available to show that the material or equipment is in conformance, affected systems shall be considered to be inoperable and reliance shall not be placed on such systems to fulfill their intended safety functions.

Nonconforming conditions that require reporting to the NRC shall be reviewed by the NSOC. Such review shall include the results of any investigations made and the recommendations resulting from such investigations to preclude or reduce the probability of recurrence of the event or circumstance.

17.16 CORRECTIVE ACTION

Each individual condition adverse to quality shall be identified, controlled, and evaluated, and a disposition shall be determined for the remedial action and corrective action as soon as practicable. These activities shall be performed consistent with Section 17.15, Control of Nonconforming Conditions.

Systematic review and evaluation of all conditions adverse to quality shall be conducted and documented. Conditions adverse to quality shall include, but not be limited to: engineering, design, and drafting errors; equipment failures and malfunctions; abnormal occurrences; deficiencies; deviations; and defective material, equipment, and services.

The review and evaluation shall include identification of quality trends, repetitive occurrences, and significant conditions adverse to quality. The quality trends and other significant review findings shall be analyzed and appropriate corrective action determined. Findings and actual or recommended corrective action shall be reported to management by the responsible organization for review and assessment.

Significant conditions adverse to quality shall be investigated to the extent necessary to assess the root causes and to determine the corrective action required to prevent recurrence of the same or similar conditions. The corrective action required for significant conditions adverse to quality shall be accomplished in a timely manner. Follow-up reviews shall be conducted by NQS to verify that the corrective action was properly implemented, performed in a timely manner, and that it was effective in correcting the identified condition.

Significant conditions adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to management. Significant conditions adverse to quality that are related to plant *or ISFSI* operations or maintenance shall be reported to the NSOC.

Significant conditions adverse to quality shall be evaluated for reportability to the NRC in accordance with 10 CFR 21, 10 CFR 50.72, 10 CFR 50.73, 10 CFR 50.9, *10 CFR 72.74, 10 CFR 72.75, the plant and ISFSI Technical Specifications*, and other applicable regulations and shall be reported as required.

17.17 QUALITY ASSURANCE RECORDS

Sufficient records shall be maintained to furnish evidence of both the quality of items and activities affecting quality and to meet applicable code, standard, and regulatory requirements. The records include all documents referred to or described in other sections of this chapter or required by implementing procedures such as operating logs, maintenance and modification procedures, related inspection results, reportable occurrences, *and other records required by the Technical Specifications and the Code of Federal Regulations*. In addition to the records of the results of reviews, *designs, fabrication, installation*, inspections, calibrations, tests, *maintenance*, surveillances, audits, personnel qualification, special process qualification, and material analyses for PG&E quality-related activities *and ISFSI structures, systems, and components that are important to safety*; those of vendors, suppliers, subcontractors, and contractors shall also be maintained.

A management control system for the collection, storage, and maintenance of completed quality assurance records shall be maintained. This records management program shall be designed and implemented to assure that the quality assurance records are complete, readily retrievable when needed, and protected from damage or destruction during storage by fire, flooding, theft, environmental conditions, or other causes.

Quality assurance records stored electronically will follow the guidance for electronic records management given in the Nuclear Information and Records Management Association (NIRMA) technical guidelines, TG 11-1998, "Authentication of Records;" TG 15-1998, "Management of Electronic Records;" TG 16-1998, "Software Configuration Management and Quality Assurance;" and TG 21-1998, "Electronic Records Protection and Restoration." Quality assurance records will only be stored on electronic media (that is, optical disk, magnetic tape, network array, etc.) meeting the requirements of the NIRMA guidelines. Information Systems will determine the appropriate electronic media. Regardless of the electronic media selected, the process must be capable of producing legible, accurate, and complete records during the required retention period.

Backup copies of in-process electronic media records will be maintained in multiple, physically-independent electronic locations. Backup copies of quality assurance records in electronic media will be maintained in multiple, physically-independent electronic locations until such time as images of these records are created, copied, and verified on two copies of an appropriate electronic storage media. The two copies will then be stored in separate physical locations. File legibility verification will be completed on all quality assurance records stored on electronic media by either visually verifying the file legibility or by electronically verifying exact binary file transfer.

Periodic media inspections to monitor image degradation will be conducted in accordance with the NIRMA guidelines or media manufacturers recommendations. These periodic inspections shall be documented.

Quality assurance records stored on electronic media will be refreshed or copied on to new media and subsequently verified if the projected lifetime of that media does not exceed the retention period of the records stored on that media. These requirements meet the intent of Generic Letter 88-18, "Plant Record Storage on Optical Disk," dated October 20, 1998.

Detailed records for items or activities shall be specified by instructions, procedures, drawings, or specification or other documents that prescribe the item or activity and shall be generated by the organization responsible for the item or activity including PG&E and non-PG&E organizations. Each department generating quality assurance records is responsible for transmitting those records to the records processing organization for archival purposes.

All records shall be assigned a retention period in conformance with Title 10, Code of Federal Regulations, other applicable codes, standards, and specifications.

17.17.1 DCPP Lifetime Records

The following records will be retained for the duration of the unit Operating License:

(1) Records and drawing changes reflecting unit design modifications made to systems and equipment described in the FSAR Update

(2) Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories

(3) Records of radiation exposure for all individuals entering radiation control areas

(4) Records of gaseous and liquid radioactive material released to the environs

(5) Records of transient or operational cycles for those unit components identified in FSAR Update, Table 5.2-4

(6) Records of reactor tests and experiments

(7) Records of training and qualification for current members of the unit staff

(8) Records of in-service inspection performed pursuant to the Technical Specifications

(9) Records of quality assurance activities required by the FSAR Update, Chapter 17

(10) Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59

(11) Records of PSRC and NSOC meetings

(12) Records of analyses required by the Radiological Environmental Monitoring Program (Reg. Guide 4.15)

(13) Records of service lives of all hydraulic and mechanical snubbers required by the FSAR Update including the date at which the service life commences and associated installation and maintenance records

(14) Records of secondary water sampling and water quality

(15) Records of reviews performed for changes made to the Offsite Dose Calculation Manual; and

(16) Records of reviews performed for changes made to the Process Control Program.

17.17.2 DCPP Nonpermanent Records

The following records will be retained for at least five years:

(1) Records and logs of unit operation covering time interval at each power level

(2) Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety

(3) All reportable events

(4) Records of surveillance activities, inspections, and calibrations required by the Technical Specifications

(5) Records of changes made to procedures required by Technical Specification 5.4.1

(6) Records of radioactive shipments

(7) Records of sealed source and fission detector leak tests and results; and

(8) Records of annual physical inventory of all sealed source material of record.

17.17.3 Diablo Canyon ISFSI Records

The following Records will be maintained as required for the Diablo Canyon ISFSI:

(1) Radiation protection program and survey records (10 CFR 20.2101 to 20.2110)

(2) *Records associated with reporting defects and noncompliance (10 CFR 21.51)*

(3) *Records important to decommissioning (10 CFR 72.30(d))*

(4) *Records of changes to the physical security plan made without prior NRC approval (10 CFR 72.44(e) and 72.186 (b))*

(5) *Records of changes, tests and experiments, and of changes to procedures described in the SAR (10 CFR 72.48(b)(1))*

(6) *Records showing receipt, inventory, location, disposal, acquisition, and transfer of spent fuel (10 CFR 72.72(a) and 10 CFR 72.72(d))*

(7) *A copy of the current inventory of spent fuel in storage at the Diablo Canyon ISFSI (10 CFR 72.72(b))*

(8) *A copy of the current material control and accounting procedures (10 CFR 72.72(c))*

(9) *Other records required by license conditions or by NRC rules, regulations or orders (10 CFR 72.80)*

(10) *Records of the occurrence and severity of important natural phenomena that affect the Diablo Canyon ISFSI design (10 CFR 72.92(b))*

(11) *Quality assurance records (including records pertaining to the design, fabrication, erection, testing, maintenance, and use of structures, systems, and components important to safety; and results of reviews, inspections, tests, audits, monitoring of work performance, and material analyses) (10 CFR 72.174)*

(12) *A copy of the current physical security plan, plus any superseded portions of the plan (10 CFR 72.180)*

(13) *A copy of the current safeguards contingency plan procedures, plus any superseded portions of the procedures (10 CFR 72.184)*

(14) *Operating records, including maintenance, alterations or additions made*

(15) *Records of off-normal occurrences and events*

(16) *Environmental survey records*

(17) *Records of employee qualifications and certifications*

(18) *Record copies of:*

- *SAR and updates*
- *Reports of accidental criticality or loss of special nuclear material*
- *Material status reports*
- *Nuclear material transfer reports*
- *Reports of pre-operational test acceptance criteria and results*
- *Procedures*
- *Environmental Report*
- *Emergency Plan*

(19) *Construction Records; and*

(20) *Records of events associated with radioactive releases.*

Facilities for the temporary or permanent storage of completed quality assurance records shall be established in predetermined locations as necessary to meet the requirements of codes, standards, and regulatory agencies. Such facilities shall be constructed and maintained so as to protect the contents from possible damage or destruction.

17.18 AUDITS

The adequacy and effectiveness of the QA Program shall be continually monitored through a comprehensive system of internal and supplier audits. The audit system implemented by NQAL includes all aspects of the QA Program. The audit system shall:

(1) Verify, through examination and evaluation of objective evidence, that this QA Program has been implemented as required

(2) Identify any deficiencies or nonconformances in this QA Program

(3) Verify the correction of any identified deficiencies or nonconformances

(4) Assess the adequacy and effectiveness of this QA Program

A comprehensive plan for the audit system shall be established and documented. This plan shall identify the scope of individual audits that are to be performed, the aspects of this QA Program covered by each audit, and the schedule for performing audits. The audit system plan shall be reviewed at least semiannually, and revised as necessary, to assure that coverage and schedule reflect current activities and that audits of plant operational phase activities *and ISFSI activities* are being accomplished in accordance with applicable requirements. Other associated activities included as part of the audit program are: indoctrination and training programs; the qualification and verification of implementation of QA programs of contractors and suppliers; interface control among the applicant and the principal contractors; audits by contractors and suppliers; corrective action, calibration, and nonconformance control systems; FSAR Update *and ISFSI SAR* commitments; and activities associated with computer codes.

Auditors shall be independent of direct responsibility for the performance of the activities that they audit, have experience or training commensurate with the scope and complexity of their audit responsibility, and be qualified in accordance with applicable standards.

Auditing shall be initiated as early in the life of an activity as is practicable and consistent with the schedule for accomplishing the activity. In any case, auditing shall be initiated early enough to assure that this QA Program is effectively implemented throughout each activity. Individual audits shall be regularly scheduled on the basis of the status and importance of the activities that they address. These audits may be supplemented by additional audits at the discretion of the Director, NQAL.

For audits, other than those who's scheduled frequency is mandated by regulation (such as the Emergency Preparedness Program, Safeguards Contingency Plans, or the Security Program), a grace period of up to 90-days may be utilized when the urgency of other priorities makes meeting the specified schedule dates impractical. For audit activities deferred by using a grace period, the next scheduled due date shall be based on the original schedule date but may not exceed the original due date plus 90 days.

Audit reports shall be prepared, signed by the Audit Team Leader, and issued to responsible management of both the audited and auditing organizations.

Audits are performed either biennially or annually. The audits are regularly scheduled on a formal audit schedule prepared by NQS. The audit schedule is reviewed regularly by the Director, NQAL, and NSOC, and the schedule is revised as necessary to assure adequate coverage as commensurate with activities and past performance. Audits are performed by trained personnel not having direct responsibilities in the area being audited and in accordance with approved audit plans. Additional audits may be performed as requested by NSOC; the Senior Vice President, Generation and Chief Nuclear Officer; the Vice President, Nuclear Services; the Director, NQAL; or NQS managers for audit in their area of responsibility.

The following areas shall be audited at least once per 24 months:

(1) The conformance of plant *and ISFSI* operation to provisions contained within the applicable Technical Specifications and applicable licenses.

(2) The performance, training, and qualifications of the entire plant *and ISFSI* staff

(3) The results of actions taken to correct deficiencies occurring in plant *and ISFSI* equipment, structures, systems, or method of operation that affect nuclear safety.

(4) The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50.

(5) The Radiological Environmental Monitoring Program, implementing procedures, and program results.

(6) The Offsite Dose Calculation Procedure and its implementing procedures.

(7) The Process Control Program and implementing procedures for processing and packaging radioactive wastes.

(8) The Nonradiological Environmental Monitoring Program.

(9) A representative sample of routine plant *and ISFSI* procedures that are used more frequently than every two years. This audit is to ensure the acceptability of the procedures and to verify that the procedures review and revision program is being implemented effectively.

(10) The performance of activities required to be audited by ANS-3.2/ANSI N18.7-1976, Section 4.5.

(11) Review of design documents and process to ensure compliance with the FSAR Update, Section 17.3 and *ISFSI SAR, Section 3.0* (for example, use of supervisors as

design verifiers). In addition, NQS shall sample and review specifications and design drawings to assure that the documents are prepared, reviewed, and approved in accordance with PG&E procedures and that the documents contain the necessary QA requirements, acceptance requirements, and quality documentation requirements.

(12) NQS shall audit the departments that quality personnel and procedures to assure that the process qualification activity, records, and personnel meet the applicable requirements. NQS shall also audit the organizations implementing special processes to provide assurance that the processes are carried out in accordance with approved procedures by qualified personnel using qualified equipment and that the required records are properly maintained.

The following activities shall be audited at least once per 12 months unless specified otherwise. However, if the audit frequencies required by the governing regulations are changed, audit frequencies shall at least meet the revised minimum requirements.

(1) The performance of activities required by the Quality Assurance Program for the Radioactive Effluent Controls Program.

(2) The Security Program in accordance with 10 CFR 73.55(g)(4) and 10 CFR 73.56(g).

(3) The Access Authorization Program in accordance with 10 CFR 73.56(g)(1) - at least once per 24 months. If a contractor's or vendor's Access Authorization Program is accepted, that contractor's or vendor's Access Authorization Program shall be audited in accordance with 10 CFR 73.56(g)(2) - at least once every 12 months.

(4) The Emergency Response Program in accordance with 10 CFR 50.54(t).

(5) The Fitness for Duty Program in accordance with 10 CFR 26.80.

(6) The Radiation Protection Program in accordance with 10 CFR 20

(7) Any other area of operation considered appropriate by NSOC; the Senior Vice President, Generation and Chief Nuclear Officer; or the Director, NQAL, at a frequency determined by NSOC or the requesting individual.

(8) The Fire Protection and Loss Prevention Program is audited in accordance with the annual, biennial, and triennial audit requirements of NRC Generic Letter 82-21.

Management of the audited organization shall review the audit report and respond to any quality problem reports, investigate any significant findings to identify their cause and determine the extent of corrective action required, including action to prevent recurrence. They shall schedule such corrective action and also take appropriate action to assure it is accomplished as scheduled. They shall respond to NQS regarding each significant finding stating the root cause, immediate action taken, and the corrective action taken or planned to prevent recurrence.

NQS shall review the written responses to all audit findings, evaluate the adequacy of each response, assure that corrective action to prevent recurrence is identified and taken for each significant finding, and confirm that corrective action is accomplished as scheduled.

Audit records shall be generated and retained by the NQAL Department for all audits.

TABLE 17.1-1

CURRENT REGULATORY REQUIREMENTS AND PG&E COMMITMENTS
PERTAINING TO THE QUALITY ASSURANCE PROGRAM

The Quality Assurance Program described in Chapter 17 of the FSAR Update, program directives, and administrative procedures complies with the requirements set forth in the Code of Federal Regulations. In addition, it complies with the regulatory documents and industry standards listed below. Changes to this list are not made without the review and concurrence of the Director, Nuclear Quality, Analysis, and Licensing

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions
(S.G.) 28	6/72	ANSI N45.2	1971	Quality Assurance Program Requirements for Nuclear Power Plants	
1.37	3/73	ANSI N45.2.1	1973	Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	*Not applicable to the independent spent fuel storage installation.*
1.38	5/77	ANSI N45.2.2	1972	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	Warehouse personnel will normally visually scrutinize incoming shipments for damage of the types listed in Section 5.2.1. This activity is not necessarily performed prior to unloading. Separate documentation of the shipping damage is not necessary. Release of the transport agent after unloading and the signing for receipt of the shipment provides adequate documentation of completion of the shipping damage inspection. Any damage noted will be documented and dispositioned. Persons performing this visual scrutiny are not considered to be performing an inspection function as defined under Regulatory Guide 1.74; therefore they do not require certification as an inspector under Regulatory Guide 1.58.

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.39	9/77	ANSI N45.2.3	1973	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	Housekeeping zones established at the power plants and ISFSIs differ from those described in the standard; however, PG&E is in compliance with the intent of the standard.
1.30	8/72	ANSI N45.2.4	1972	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	The evaluation of (data sheets) acceptability is indicated on the results and data sheets by the approval signature (paragraph 2.4). No visual examination for contact corrosion is made on breaker and starter contacts unless there is evidence of water damage or condensation. Contact resistance tests are made on breakers rated at 4 kV and above. No contact resistance test is made on lower voltage breakers or starters (paragraph 3[4]). No system test incorporates a noise measurement. If the system under test meets the test criteria, then noise is not a problem (paragraph 6.2.2).
1.94	4/76	ANSI N45.2.5	1974	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	Except PG&E will not require manufacturer's certification for material suitability as inferred in ANSI N45.2.5, Sections 3.1 and 3.2 when PG&E procures: (a) material from a supplier that has a QA program that meets the relevant requirements of 10CFR50, Appendix B and the supplier is included ASME Section III (NCA-3800/NCA-4000) or on the PG&E Qualified Suppler List; or (b) material as a "Commercial-Grade" item and dedicates it in accordance with PG&E's Commercial-Grade Dedication Program.

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.29	9/78			Seismic Design Classification	
1.58	9/80	ANSI N45.2.6	1978	Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel	ANSI N45. 2. 6 applies to individuals conducting independent QC inspections, examinations, and tests. ANSI/ANS 3.1-1978 applies to personnel conducting inspections and tests of items or activities for which they are responsible (e.g., plant and ISFSI surveillance tests, maintenance tests, etc.). Except that Inspector/Examiner reevaluation due dates may extend a maximum of 90 days. The next reevaluation due date shall be based on the original scheduled due date but shall not exceed the original due date plus 90 days. NDE personnel shall be qualified and certified in accordance with SNT-TC-1A 1984 Edition. "Should" shall be interpreted as "shall" to comply with the intent of ASME Section XI. NDE personnel who perform examinations of the containment structure per the requirements of Section XI, Subsection IWE and IWL, visual examinations and ultrasonic thickness measurement only, shall be qualified and certified to ANSI/ASNT CP-189-1991

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions
					ISI ultrasonic examiners shall meet the additional requirements of ASME Section XI, Appendix VIII, 1995 Edition with 1996 Addenda. The required implementation dates for the various supplements are as specified in 10 CFR Part 50, RIN 3150-AE26.
1.116	5/77	ANSI N45.2.8	1975	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	
1.88	10/76	ANSI N45.2.9	1974	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records	Except PG&E will comply with the 2-hour rating of Section 5.6 of ANSI N45.2.9 issued July 15, 1979. Except PG&E will also meet the intent of the guidelines for the storage of QA records in electronic media, as endorsed by Generic Letter 88-18, "Plant Record Storage on Optical Disks," issued October 20, 1988, and Regulatory Issues Summary 2000-18, "Guidance on Managing Quality Assurance Records in Electronic Media," issued October 23, 2000. *Note: PG&E will maintain records of spent fuel and high-level radioactive waste in storage in accordance with ANSI N45.2.9-1974 rather than 10 CFR 72.72(d). Refer to ISFSI SAR Section 9.4.2.*
1.74	2/74	ANSI N45.2.10	1973	Quality Assurance Terms and Definitions	

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.64	6/76	ANSI N45.2.11	1974	Quality Assurance Requirements for the Design of Nuclear Power Plants	Except PG&E will allow the designer's immediate supervisor to perform design verification in exceptional circumstances and with the controls as described in NUREG-0800, Revision 2, July 1981.
1.144	1/79	ANSI N45.2.12	1977	Auditing of Quality Assurance Programs for Nuclear Power Plants	Except the schedule date for triennial vendor audits, and annual supplier evaluations may be extended a maximum of 90 days. The next schedule due date shall be based on the original scheduled due date but shall not exceed the original due date plus 90 days. Except that the corrective action program stipulated in the NPG QA Program may be used instead of the requirements of Section 4.5.1 as long as the appropriate time limits are applied to significant conditions adverse to quality. Also, no additional documentation is necessary if needed corrective actions are taken and verified prior to audit report issuance.
1.123	7/77	ANSI N45.2.13	1976	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	In addition to ANSI N45.2.13, Section 10.3.3, PG&E will accept items and services which are complex or involve special processes, environmental qualification, or critical characteristics which are difficult to verify upon receipt by suppliers' Certificate of Conformance if and only if the supplier has been evaluated and qualified utilizing Performance Based Supplier Audit techniques.

TABLE 17.1-1

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.146	8/80	ANSI N45.2.23	1978	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants	Except that auditor recertification due dates may be extended a maximum of 90 days. The next recertification due date shall be based on the original scheduled due date but shall not exceed the original due date plus 90 days.
1.33	2/78	ANSI N18.7	1976	Quality Assurance Program Requirements (Operation)	Except that PG&E will not perform biennial review of plant procedures. DCPP has programmatic controls procedures in place that replaces the biennial review process except under the conditions described in note below. These controls are described in PG&E letter to NRC No. DCL-92-204. See note at end of table. Except that item 5, administrative control for providing suggested procedure enhancements feedback to sponsors, has been changed to allow the use of other means, along with the AR, as long as those means are both reliable and secure to the extent appropriate to provide feedback. Except for temporary changes to procedures, PG&E will require a review by an individual who holds a Senior Reactor Operators license only if the procedure is one of the types listed in Section 17.5 (8) of this FSAR Update. Furthermore, this individual need not be the supervisor in charge of the shift. Except that audit frequencies specified in Regulatory Guide 1.33, Revision 2, need not be met. Audits shall be performed at the frequencies specified in Section 17.18 of this FSAR Update.

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions
					Except that a grace period of up to 90 days will be allowed for audit scheduling, except where the schedule is mandated by regulation. The next schedule due date shall be based on the original scheduled date but shall not exceed the original due date plus 90-days.
1.8	2/79	ANSI/ANS 3.1	1978	Personnel Selection and Training	Except that the one year of qualifying nuclear power plant experience in the overall implementation of the Quality Assurance program can be obtained outside the Quality Assurance organizations. Except certain personnel are trained and qualified to the Institute of Nuclear Power Operations (INPO) criteria as described in FSAR Update Chapter 13. Except that a retraining and replacement training program for the plant staff meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR 55. This exception is based on the NRC letter to PG&E, dated July 19, 1989, issuing License Amendments No. 43 and 42. Except that the Radiation Protection Manager's qualifications shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 2, April 1987 for the Radiation Protection Manager.

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions
					Except that personnel performing the Independent Technical Review function shall have at least 3 years of related experience and a Bachelor's Degree in Engineering or a related field; or shall have at least 8 years of related experience. This exception is based on NRC letter to PG&E dated March 31, 1994, issuing License Amendments No. 91/90. Except that the Operations Manager shall meet the requirements of the Technical Specifications.
4.15	2/79			Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment	Record retention requirements are stated in Section 17.17.
BTP PCSB 9.5-1 Appendix A	5/76			Guidelines for Fire Protection for Nuclear Power Plants	The fire protection program for DCPP satisfies the requirements of GDC 3 (1967) by complying with the guideline of Appendix A to NRC Branch Technical Position (BTP) (APCSB) 9.5-1, and with the provisions of 10 CFR 50.48 and Appendix R, Section III.G, J, L, and O, as stipulated by Operating License Condition 2.C(5) and 2.C(4) for Units 1 and 2, respectively. Approved deviations from Appendix A to BTP (APCSB) 9.5-1, and Appendix R sections are identified in Supplement Numbers 8, 9, 13, 23, 27, and 31 to the Safety Evaluation Report.

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions
					Due to the absence of combustible materials within the ISFSI, other than the fuel in the onsite transporter, and based upon an analysis of a transporter fuel tank fire, it is concluded that a fire protection program is not required for the ISFSI. Thus, this BTP is not applicable.
1.26	2/76			Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants	Design and construction of Diablo Canyon Power Plant started in 1965 and most of the work cannot comply with the specific requirements of Regulatory Guide 1.26, February 1976. The intent of the Reg. Guide has been followed as shown by comparing the Reg. Guide with Table 3.2-2 in the FSAR and the Q-List (Reference 8 of Section 3.2). *This Regulatory Guide does not apply to the ISFSI.*
		NCIG-01	2	Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants	
		NCIG-02	2	Sampling Plan for Visual Re-inspection of Welds	
		NCIG-03	1	Training Manual for Inspection of Structural Weld at Nuclear Power Plants Using the Acceptance Criteria of NCIG-01	

TABLE 17.1-1

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.97	05/83	ANSI/ANS 4.5	1980	Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant And Environs Conditions During And Following An Accident	*This Regulatory Guide does not apply to the ISFSI.*

Note for Regulatory Guide 1.33:

The following controls shall be conformed with as requested by the NRC in response to PG&E Letter DCL-92-204. These controls were requested by the NRC in their letter of March 2, 1993.

1. All procedures potentially affected by a modification shall be identified and reviewed by those departments or sections that are potentially affected by the modifications. Necessary procedure changes or revisions shall be made prior to the use of the procedure in the new configuration.

   Note: Procedures which are not required to be revised to place the modified SSC in service, but may not be used on the modified configuration until revised, shall be placed on administrative hold to prevent their use per clarifying answer to the PSRC on 10/20/00 (PSRC meeting 00-064).

2. The problem resolution program shall include consideration of procedure changes and revisions that may be required as corrective actions to prevent recurrence.

3. Revision to the Technical Specifications and FSAR Update shall be evaluated for impact on procedures to identify needed procedure revisions.

4. If a procedure cannot be performed as written, or if it induces additional risk to equipment or personnel, the user shall stop work, place the system/component in a safe condition, and contact supervision. Procedure changes are then to be made on either a temporary or permanent basis to allow continued use of the procedure.

5. The administrative controls program shall provide for user feedback to sponsors of procedures. Documentation of suggested enhancements are provided by action request in PIMS or by the use of other means, as long as those means are both reliable and secure to the extent appropriate to provide that feedback.

6. The DCPP industry operating experience assessment program requires the review of industry operating experience data for applicability to DCPP and for determination of action required. This review shall include an evaluation of applicable procedures and the initiation of any required procedure changes.

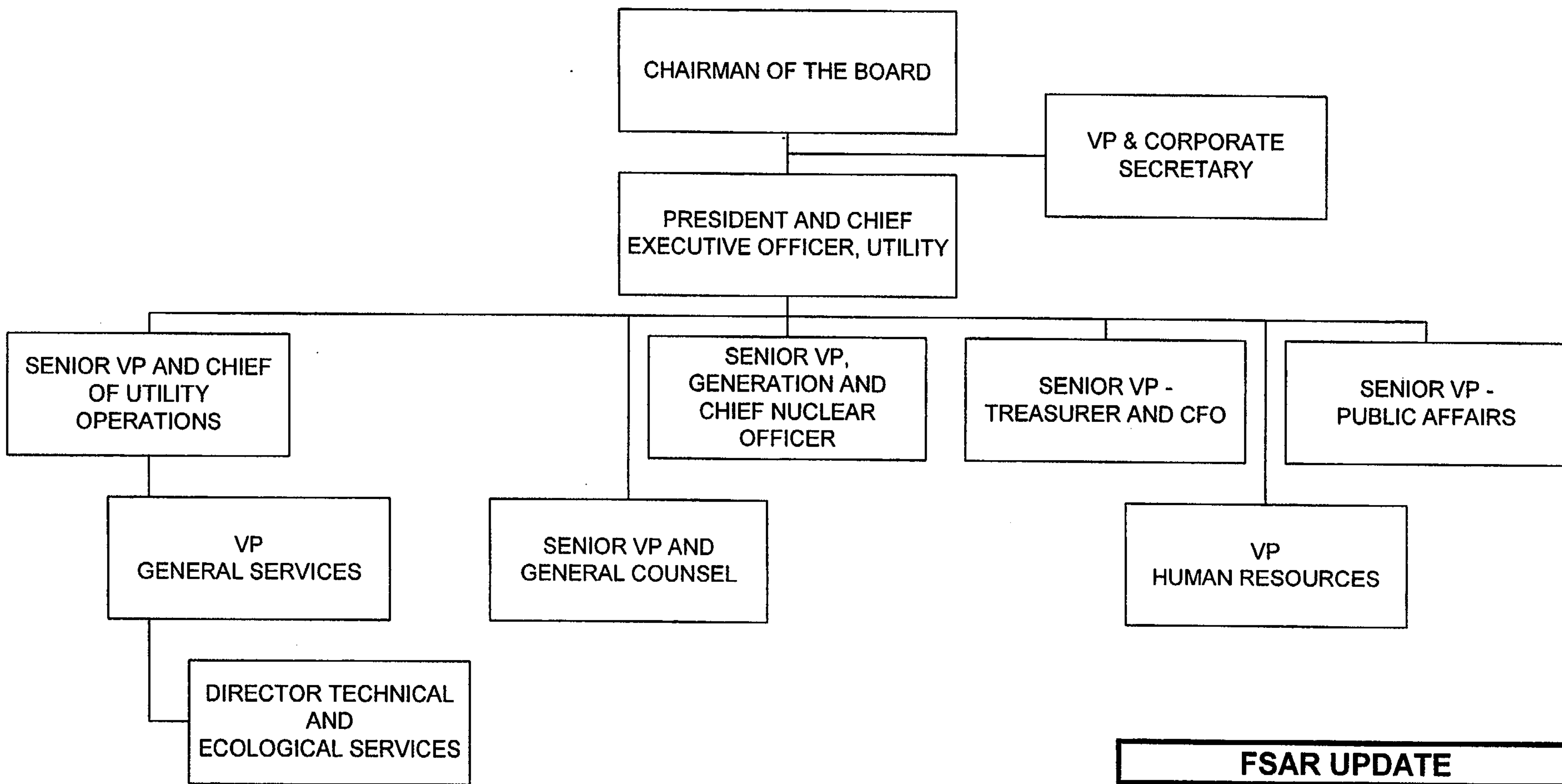
7. Commitments or corrective actions identified as a result of regulatory issues shall address any required procedure changes or revisions. This includes responses to violations, other issues raised in inspection reports, GLs, and other regulatory concerns.

TABLE 17.1-1

Reg. Guide	Date	Standard No.	Rev.	Title/Subject	Exceptions

These controls are in addition to those specified above per NRC reply to DCL-92-204.

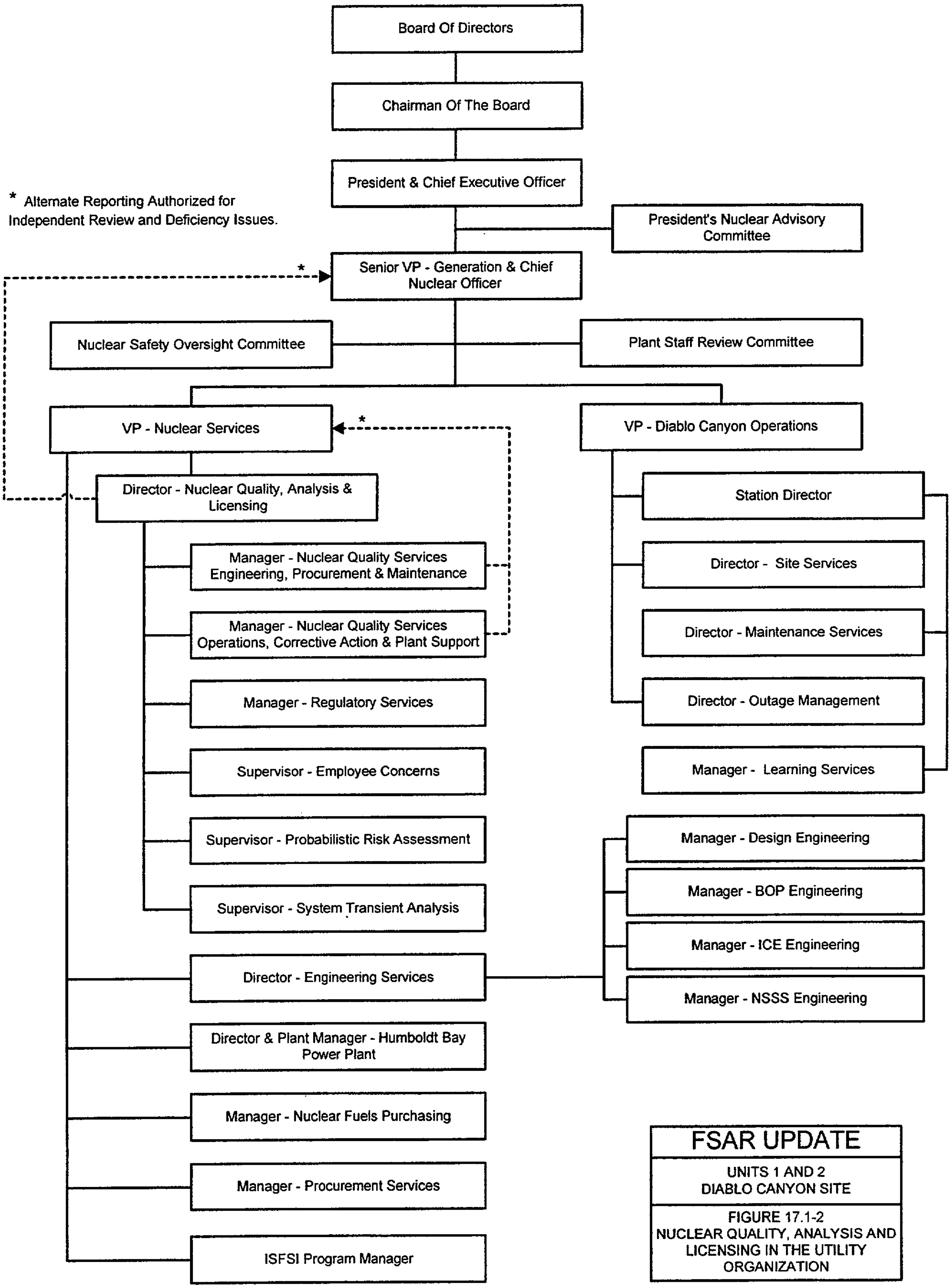
1. All applicable plant procedures shall be reviewed following an unusual incident, such as an accident, unexpected transient, significant operator error, or equipment malfunction, as specified by Section 5.2 of ANSI N18.7/ANS 3.2.
2. Non-routine procedures (e.g. emergency operating procedures, procedures which implement the emergency plan, and other procedures whose usage may be dictated by an event) shall be reviewed at least every two years and revised as appropriate.
3. Routine plant procedures that have not been used for two years shall be reviewed before use to determine if changes are necessary or desirable.



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 17.1-1
PACIFIC GAS AND ELECTRIC COMPANY UTILITY ORGANIZATION



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 17.1-2
NUCLEAR QUALITY, ANALYSIS AND LICENSING IN THE UTILITY ORGANIZATION

ATTACHMENT F

PRELIMINARY DECOMMISSIONING PLAN

DIABLO CANYON ISFSI
PRELIMINARY DECOMMISSIONING PLAN

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CHAPTER 1

INTRODUCTION

Prior to the end of the Diablo Canyon ISFSI life, the multi- purpose canisters (MPCs) containing spent fuel elements will be transferred from storage overpacks into transportation casks and transported offsite. Since the MPCs are designed to meet DOE 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 Diablo Canyon ISFSI will be decommissioned by the timely identification and removal of any residual radioactive materials above the applicable NRC limits for unrestricted use; releasing the site for unrestricted use in accordance with Regulatory Guide 1.86 (Reference 1); and terminating the NRC license.

This Preliminary Decommissioning Plan has been prepared to comply with the requirements of 10 CFR 72.30 and describes the conceptual program for decontamination and decommissioning of the Diablo Canyon ISFSI, including the proposed practices and procedures for decontamination of the site and facilities, the disposal of radioactive materials, and the cost estimate associated with decommissioning. The specific methods and details of Diablo Canyon ISFSI decommissioning will be included in a formal decommissioning plan, that will be submitted for NRC review and approval prior to the commencement of decommissioning activities. Additional information regarding design features that facilitate decommissioning is provided in the Holtec references cited in Section 1.5 of the Diablo Canyon ISFSI Safety Analysis Report (SAR).

CHAPTER 2

DECOMMISSIONING OBJECTIVE, ACTIVITIES, AND TASKS

2.1 DECOMMISSIONING OBJECTIVE

The objective of decommissioning activities for the Diablo Canyon ISFSI is to remove all radioactive materials having activities above the applicable NRC release limits (currently 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination") in order that the site may be released for unrestricted use, and the NRC license terminated.

2.2 DECOMMISSIONING ACTIVITIES

Detailed information on proposed practices and procedures for decommissioning activities will be provided in a final decommissioning plan. The extent of any required decontamination efforts cannot be quantified at this time, especially in light of the Diablo Canyon Power Plant's (DCPP) "start clean/stay clean" philosophy and the efforts that will be taken throughout the life of the facility to minimize the potential for any contamination. Actual decontamination efforts and sequences of work will depend on facility operating history and whether any contamination actually exists. The descriptions presented here provide a conceptual plan for detailed engineering and planning that will occur at the end of facility operations.

The loading of spent fuel into the MPCs occurs in DCPP fuel handling building/auxiliary building, as described in Sections 4.4 and 5.1 of the Diablo Canyon ISFSI SAR. As part of each loading operation, the components that have been in contact with contaminated spent fuel pool water (i.e., the transfer cask and the top of the MPC lid) are checked for surface contamination, and are decontaminated as necessary before being transported to the cask transfer facility (CTF). Because of this requirement, it is anticipated that at the time of decommissioning, the transfer cask can be decontaminated to free release levels; if this is not the case, then it will be disposed of at an appropriate facility.

It is not anticipated that either the storage overpacks or the storage pads will have residual radioactive contamination once the MPCs are removed because: (a) the MPCs are sealed by welding that precludes leakage, (b) measures are applied when fuel is loaded into the MPCs to prevent contamination of their outer surfaces, and (c) neutron flux levels generated by the spent fuel are sufficiently low that activation of storage cask and pad materials will be insignificant, with radiation levels that support either unrestricted release of materials or release as low specific activity (LSA) material. It is anticipated that HI-STORM 100SA overpacks, which meet applicable free release criteria, may be reused at other nuclear facilities following their use at the Diablo Canyon ISFSI. The LSA material will be suitable for burial in a near-surface disposal site.

Because of the administrative controls used to check for and remove (if possible) any contamination from the HI-TRAC transfer cask prior to its leaving the fuel handling building, it is anticipated that the CTF, transporter, and fences will not be contaminated at the time of decommissioning. Therefore, they will require no decontamination or special handling and will be left in place or removed as determined by PG&E. If this is not the case, they will be decontaminated to free release levels or disposed of at an appropriate facility.

PG&E intends to submit a final decommissioning plan to the NRC at least one year prior to the final removal of MPCs from the site, and in no case later than one year prior to the expiration of the NRC operating license. The final decommissioning plan will address decontamination of the site, removal of radioactive materials, and termination of the facility-operating license, and will include a description of how the Diablo Canyon ISFSI will continue to protect the public health and the environment during decommissioning. The final decommissioning plan will be developed in accordance with the applicable NRC regulations in effect at the time of preparation of the plan. Decommissioning activities will be planned using as low as is reasonably achievable (ALARA) goals and criteria for protection of personnel from exposure to radiation and radioactive material. The final decommissioning plan will include such information as follows:

- A description of the current conditions of the ISFSI site sufficient to evaluate the acceptability of the plan.
- The choice of the alternative for decommissioning with a description of the activities involved.
- A description of controls and limits on procedures and equipment to protect occupational and public health and safety.
- A description of the planned final radiation survey.
- An updated detailed cost estimate for the chosen alternative for decommissioning; a comparison of that estimate with present funds set aside for decommissioning; and the plan for assuring the availability of adequate funds for completion of decommissioning, including means for adjusting cost estimates and associated funding levels over any storage or surveillance period.
- A description of technical specifications and quality assurance provisions in place during decommissioning.

2.3 DECOMMISSIONING TASKS

Prior to the commencement of Diablo Canyon ISFSI decommissioning activities, the MPCs will be shipped offsite in licensed transportation casks. The empty overpacks will then be surveyed to determined activation and contamination levels.

Overpacks with activation and contamination levels below the applicable NRC limits for unrestricted release will be disposed of as noncontrolled material. Overpacks with contamination or activation levels above the applicable NRC limits for unrestricted release will be dismantled, with the activated or contaminated portions segregated and disposed of as low-level waste. The dismantled portions or components of overpacks that are below the applicable NRC limits for unrestricted release will be disposed of as noncontrolled material. Storage cask decontamination and decommissioning may be performed at any time following the removal of the MPC. This will allow overpack decommissioning efforts to be essentially complete by the end of MPC shipping operations. Likewise, the transfer cask will be similarly surveyed for contamination, decontaminated, or dismantled and disposed of as low-level waste after its use is no longer required.

Characterization surveys will be performed to verify that the storage pads and storage site area are free of contamination (i.e., with radiation and radioactivity levels below the applicable NRC limits for unrestricted release). In the event that the characterization surveys identify contamination levels above the applicable NRC limits for unrestricted release, the structures or components will be decontaminated using conventional decontamination techniques that minimize the volume and processing of the resulting radwaste. All low-level radioactive waste generated during decontamination efforts, and portions of any structures or components that remain contaminated, will be shipped offsite for disposal at an appropriate licensed facility.

After all the MPCs have been shipped from the Diablo Canyon ISFSI, and the overpacks and transfer cask decommissioned, a detailed radiological characterization survey will be performed of the CTF, with particular attention focused on any areas of known or historic contamination. CTF equipment or structures that may have contamination levels above applicable NRC limits for unrestricted release will be decontaminated to the extent practical using conventional methods. All radioactive material above the applicable NRC limits for unrestricted release will be removed from the site and disposed of as low-level waste.

A final radiation survey will be conducted to ensure that the ISFSI site is suitable for release in accordance with the 10 CFR 20, Subpart E criteria for decommissioning.

2.4 DECOMMISSIONING ORGANIZATION

The decommissioning organization and staff requirements will be defined in the final decommissioning plan. Trained and qualified personnel will be used to perform the technical, field, and administrative tasks required during decommissioning. To the extent practicable, the decommissioning organization will include staff from the PG&E DCPP organization to capitalize on their knowledge and familiarity with the facility. Contractors may be used to provide specialized services, or to supplement the facility staff when warranted.

CHAPTER 3

DECOMMISSIONING RECORDS

The following records will be maintained until the Diablo Canyon ISFSI is released for unrestricted use, in accordance with 10 CFR 72.30(d), and will be used to plan the actual decommissioning efforts:

- Records of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site. These records will include any known information on identification of nuclides, quantities, forms, and concentrations.

- As-built drawings and modifications of structure and equipment in restricted areas.

- A document, which is updated a minimum of every 2 years, containing: (a) a list of all areas designated at any time as restricted areas as defined in 10 CFR 20.1003; and (b) a list of all areas outside of restricted areas involved in a spread of contamination as required by 10 CFR 72.30(d)(1).

- Records of decommissioning cost estimates and the funding method used.

These records will be stored at DCPP as part of the records management program, which is discussed in the ISFSI License Application, Attachment E, "Quality Assurance Program."

CHAPTER 4

DECOMMISSIONING COST ESTIMATE

Decommissioning the Diablo Canyon ISFSI will be a multiphase effort, with radioactive contamination removed upon discovery, as possible, during the operational phase. The amount of decontamination required and the extent of decommissioning efforts will be based on the usage and the history of the facility. 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.

Nonetheless, a cost estimate for decommissioning has been performed that assumes certain areas and components will require decontamination. This cost estimate is part of the total estimate performed by TLG Services, Inc. for DCPP Units 1 and 2. This detailed estimate is contained in the PG&E March 2001 Decommissioning Funding Report to the NRC (Reference 2), as required by 10 CFR 50.75(f)(1). As shown therein, it is estimated that decommissioning the Diablo Canyon ISFSI will cost about $12.5 million when escalated to 2001 dollars - for the DECON alternative. The major cost contributors are cost of labor, radioactive waste disposal, and radiological surveys. The costs are based on several key assumptions, including regulatory requirements, estimating methodology, contingency requirements (a composite average of 26 percent was assumed), low-level radioactive waste disposal availability, high-level radioactive waste disposal options, and site restoration requirements. This ISFSI decommissioning cost estimate of $12.5 million only covers the costs for decontamination and disposal of low-level waste; it does not cover the costs for demolition and disposal of noncontaminated material, which are estimated at $6.5 million in 2001 dollars.

In developing this estimate, TLG Services had to make some assumptions regarding the spent fuel storage system and the size of the ISFSI due to PG&E having not yet selected the storage system vendor. TLG Services assumed "NUHOMS" storage casks would be used. The TLG Services' cost estimate will be updated to reflect the Holtec International HI-STORM 100 Storage system. This update will be contained in the applicable biennial PG&E Decommissioning Funding Report to the NRC; thereafter, this Preliminary Decommissioning Plan will be updated accordingly.

CHAPTER 5

DECOMMISSIONING FUNDING PLAN

PG&E has established an external sinking trust fund account for decommissioning DCPP Units 1 and 2. This account contains monies for decommissioning the Diablo Canyon ISFSI. This financial assurance mechanism is prepared in conformance with the guidance of NRC Regulatory Guide 3.66 (Reference 3) and complies with the requirements of 10 CFR 72.3(c).

The status of this account is provided in the PG&E March 2001 Decommissioning Funding Report to the NRC, as required by 10 CFR 50.75(f)(1). As shown therein, and based upon current guidelines and assumptions, PG&E is confident that this trust fund account will contain sufficient funds to accommodate the decommissioning of the Diablo Canyon ISFSI.

CHAPTER 6

DECOMMISSIONING FACILITATION

The sources of contamination are the spent fuel itself and the spent fuel pool water. In conformance with 10 CFR 72.130, the spread of contamination from these sources can be controlled via various ISFSI design features and health physics measures as described herein.

The design features of the HI-STORM 100 System, plus a "start clean/stay clean" philosophy, will facilitate decommissioning the Diablo Canyon ISFSI. Radioactive materials associated with spent fuel assemblies are contained within MPCs, which have been welded before leaving the DCPP fuel handling building/auxiliary building. The MPC conforms to requirements of Section III of the ASME code and provides assurance that radioactive material will not be released from the MPC over the life of the Diablo Canyon ISFSI. Health physics measures to ensure MPC external surfaces are maintained in a clean condition are implemented during the MPC loading operations. These measures minimize contaminated fuel pool water from contacting the external surfaces of the MPC. Following fuel loading operations, a swipe survey is performed on the MPC lid and on the transfer cask. Using administrative controls, transport of the transfer cask and MPC to the CTF and storage pads is not permitted if removable contamination levels exceed defined limits. Therefore, it is expected that the transfer cask and MPCs will have minimal, if any, contamination on external surfaces. Since the MPCs are sealed to preclude release of radioactive material from inside the MPCs, minimizing contamination on the external surfaces of the MPCs transported to the ISFSI storage pads minimizes the quantity of radioactive waste and contaminated equipment.

The HI-STORM 100 System overpacks that house the MPCs are clean and have no radioactive contamination when they are fabricated. The overpacks are not used inside DCPP. Under normal conditions of MPC transfer and storage operations, the potential does not exist for contaminating the overpacks. However, the interior design of the overpacks facilitates decontamination, if necessary. The cavities of the overpacks are mostly lined with steel and coated – including the cylindrical walls, pedestal that supports the MPC, and the lid – making them relatively easy to decontaminate.

Radiation protection technicians monitor the MPC transfer operations, and perform swipe surveys of the transfer cask, MPC lid, transporter, CTF, and overpack during and following each MPC transfer operation. If the transfer cask has contamination levels on its outer surfaces above those established by administrative controls to minimize the spread of contamination, it will be decontaminated prior to movement to the CTF. These measures help to minimize the spread of any contamination to the CTF and from the CTF to the storage pads.

As shown in Section 2.4 of the HI-STORM 100 System Final Safety Analysis Report (Reference 4), the overpack materials will be only slightly activated as a result of their long-term exposure to the relatively small neutron flux emanating from the spent fuel. This will allow the slightly activated overpack materials to qualify as Class A waste in stable form in accordance with 10 CFR 61.55 (Reference 5). As such, the material would be suitable for burial in a near-surface disposal site as low specific activity (LSA) material. The results for the overpacks can be conservatively applied to the ISFSI storage pads because the overpacks shield most of the neutron flux from the spent fuel. Hence, any tasks necessary to decommission overpacks and the storage pads are expected to involve only surface decontamination, as necessary, and not removal of activation products at depths below the surface.

The design of the transfer cask also facilitates its decontamination. It has layers of gamma (lead) and neutron shield materials sandwiched between steel. The inner and outer liners both consist of coated carbon steel, which is relatively easy to decontaminate.

In order to facilitate decommissioning of the CTF, nonthreaded surfaces, where practical, are covered with a nonporous coating. This provision helps to ensure that decontamination can be performed by wiping down surfaces or stripping the coating, without the need to use more aggressive methods (e.g., abrasive blasting, scabbling) that require removal of surface concrete.

Radioactive waste generated during decontamination operations will be packaged and temporarily staged for disposal in the low level waste holding area of the CTF. It is anticipated that this low-level waste holding area will be decommissioned last, following decommissioning of the storage casks, pads, and the remainder of the CTF.

Minimal nonradioactive hazardous materials may be used or stored at the Diablo Canyon ISFSI and any that are needed to support the ISFSI operations will be identified and controlled in accordance with procedures. Strict measures will be applied to prevent any hazardous materials from contacting radioactive contamination, so that mixed hazardous and radioactive waste will not be generated at the Diablo Canyon ISFSI.

CHAPTER 7

REFERENCES

1. Regulatory Guide 1.86, Terminating of Operating License for Nuclear Reactors, USNRC, June 1974.

2. PG&E's Decommissioning Funding Reports for Diablo Canyon Power Plant Units 1 and 2 and Humboldt Bay Power Plant, Letters DCL-01-026 and HBL-01-005 to the NRC, March 30, 2001.

3. Regulatory Guide 3.66, Standard Format and Content of Financial Assurance Mechanisms Required for Decommissioning Under 10 CFR Parts 30, 40, 72 and 72, USNRC, June 1990.

4. Final Safety Analysis Report for the HI-STORM 100 System, Holtec International Report No. HI-2002444, Revision 0, July 2000.

5. 10 CFR 61.55, Waste Classification.