November 21, 2005

Mr. David H. Oatley, Acting Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION TO TECHNICAL SPECIFICATIONS 3.7.17 AND 4.3 FOR A TEMPORARY CASK PIT SPENT FUEL STORAGE RACK FOR CYCLES 14 TO 16 (TAC NOS. MC5143 AND MC5144)

Dear Mr. Oatley:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 183 to Facility Operating License No. DPR-80 and Amendment No. 185 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated November 3, 2004, and its supplements dated February 24, June 23, and September 30, 2005. The supplement letter dated September 30, 2005 revised the original requested implementation from within 90 days to upon installation of the temporary cask pit spent fuel rack.

The amendments allow installation and use of a temporary cask pit spent fuel storage rack for Units 1 and 2. The cask pit rack would allow the storage of an additional 154 spent fuel assemblies for each unit. The total spent fuel pool storage capacity for each unit would be increased from the current 1324 spent fuel assemblies to 1478 assemblies for Cycles 14-16.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice. In addition, comments by Mothers for Peace, dated February 10, 2005, are addressed in the enclosure of this letter.

Sincerely,

/RA/
Girija S. Shukla, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275
and 50-323

Enclosures:
1. Amendment No. 183 to DPR-80
2. Amendment No. 185 to DPR-82
3. Safety Evaluation
4. Staff's response to Mothers for Peace Comments

cc w/encls: See next page

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PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 183
License No. DPR-80

1. The Nuclear Regulatory Commission (Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (licensee) dated November 3, 2004, and its supplements dated February 24, June 23, and September 30, 2005, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-80 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 183, are hereby incorporated in the license. Pacific Gas and Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance, and shall be implemented upon installation of the temporary cask pit spent fuel rack.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 21, 2005

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 185
License No. DPR-82

1. The Nuclear Regulatory Commission (Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (licensee) dated November 3, 2004, and its supplements dated February 24, June 23, and September 30, 2005, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-82 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 185, are hereby incorporated in the license. Pacific Gas and Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance, and shall be implemented upon installation of the temporary cask pit spent fuel rack.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 21, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 183

TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 185 TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NOS. 50-275 AND 50-323

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.7-28	3.7-28
3.7-29	3.7-29
3.7-30	3.7-30
3.7-31	3.7-31
3.7-32	3.7-32
4.0-1	4.0-1
4.0-2	4.0-2
----	4.0-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 183 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 185 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By application dated November 3, 2004, and its supplements dated February 24, June 23, and September 30, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession Numbers ML043130392, ML050630338, ML051800264, and ML052790486, respectively), Pacific Gas and Electric Company (PG&E or licensee) requested changes to the Technical Specifications (TS; Appendix A to Facility Operating License Nos. DPR-80 and DPR-82) for the Diablo Canyon Power Plant, Units 1 and 2 (DCPP). The supplemental letters dated February 24, June 23, and September 30, 2005, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 21, 2004 (69 FR 76481).

The requested changes would revise the DCPP TS to allow installation and use of a temporary cask pit spent fuel storage rack for Units 1 and 2. The cask pit rack would allow the storage of an additional 154 spent fuel assemblies for each unit. The total spent fuel pool (SFP) storage capacity for each unit would be increased from the current 1,324 spent fuel assemblies to 1,478 spent fuel assemblies for Cycles 14-16. At the end of Cycle 16, the licensee will remove the temporary cask pit spent fuel storage rack.

Based on the current inventory of fuel assemblies stored in the SFP and anticipated discharges of spent fuel, Unit 1 will lose full core offload capability (FCOC) in 2007, and Unit 2 will lose FCOC in 2008. On September 26, 2003, PG&E received Amendment No. 162 to Facility Operating License No. DPR-80, and Amendment No. 163 to Facility Operating License No. DPR-82 for DCPP Units 1 and 2. These amendments authorize handling and loading of Holtec International's (Holtec's) multi-purpose canisters and transfer cask in the DCPP Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) facilities. On March 19, 2004, PG&E received a 10 CFR Part 72 license for an Independent Spent Fuel Storage Installation (ISFSI) at Diablo Canyon. PG&E is presently in the process of obtaining a Coastal Development Permit (CDP) from the California Coastal Commission, which is required for construction of the ISFSI. The delay in the issuance of the CDP has resulted in a delay of the final design, construction, and projected operational date of the ISFSI, such that PG&E finds it necessary to provide additional temporary spent fuel storage to ensure that FCOC is retained in accordance with the current operational practice for both Units.

2.0 REGULATORY EVALUATION

2.1 Criticality Evaluation Regulatory Basis

The spent fuel pool (SFP) criticality is governed by 10 CFR 50.68. Additional guidance is provided in Standard Review Plan (SRP), NUREG-800, Revision 2, Section 9.1.2, "Spent Fuel Storage," July 1981, in the L. Kopp letter to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," dated August 19, 1998, and in Regulatory Guide 1.13, Revision 2, "Spent Fuel Storage Facility Design Basis," December 1981. Finally, General Design Criterion (GDC) 62, provides design requirements for prevention of criticality in fuel storage and handling.

Regarding the heat transfer aspects of fuel storage, GDC 61, "Fuel storage and handling and radioactivity control," requires, in part, that reliable and testable means be provided for the removal of the decay heat. In the absence of specific requirements, the thermal hydraulic characteristics of the spent fuel pool should be the same as the existing pool design basis.

2.2 Crane and Heavy Loads Evaluation Regulatory Basis

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides guidelines and recommendations to assure safe handling of heavy loads by prohibiting, to the extent practicable, heavy load travel over stored spent fuel assemblies, fuel in reactor core, safety-related equipment, and equipment needed for decay heat removal.

NUREG-0612 endorses a defense-in-depth approach for handling of heavy loads near spent fuel and safe shutdown systems. General guidelines for overhead handling systems that are used to handle heavy loads in the area of the reactor vessel and SFP are given in Section 5.1.1 of NUREG-0612. They are as follows: (1) definition of safe load paths; (2) development of procedures for load handling operations; (3) training and qualification of crane operators in accordance with Chapter 2-3 of the American National Standards Institute (ANSI) B30.2 - 1976; (4) use of special lifting devices that meet guidelines in ANSI N14.6 - 1978; (5) installation and use of non-custom lifting devices in accordance with ANSI B30.9 - 1971; (6) inspection, testing, and maintenance of cranes in accordance with Chapter 2-2 of ANSI B30.2 - 1976; and (7) design of the crane in accordance with chapter 2-1 of ANSI B30.2- 1976 and CMMA-70.

Section 5.1.2 of NUREG-0612 provides additional guidelines for control of heavy loads in the spent fuel pool area of pressurized-water reactors. Recommended supplemental actions include either using a single-failure proof handling system or evaluating the effects of a drop against the criteria of Section 5.1 of NUREG-0612. Appendix A of NUREG-0612 includes guidelines for evaluating the effects of load drops.

2.3 SPF Cooling Evaluation Regulatory Basis

Appendix A of 10 CFR Part 50, GDC 61, specifies, in part, that fuel storage systems shall be designed with residual heat removal capability having reliability and testability that reflect the importance to safety of decay heat removal, and with the capability to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Attachment 2 to Matrix 5 of Section 2.1 of NRC Review Standard RS-001, Revision 0, provides the Nuclear Regulatory Commission (NRC) staff with review guidance for determining the adequacy of spent fuel pool cooling capability. It supercedes the guidance of paragraphs III.1.d. and III.1.h. of Standard Review Plan (NUREG-0800), Section 9.1.3. Section 3.1 of RS-001 states that "the licensee demonstrates adequate SFP cooling capacity by either performing a bounding evaluation or committing to a method of performing outage-specific evaluations." The analysis conditions to be assumed for bounding and cycle-specific analysis are given in Section 3.1.1 of RS-001. Section 3.2 of RS-001 provides guidance for ensuring adequate makeup supply.

The SFP cooling system is described in Chapter 9 of the DCPP FSAR. Section 9.1.3 provides design-basis information for the SFP, including the SFP temperature limits for both normal (planned) and abnormal (emergency) refueling scenarios. The system description along with the applicable design-basis information included in Chapter 9 provides the criteria needed to evaluate the impact that the increased SFP heat load has on the ability of the SFP system to comply with the plant design basis and GDC 61. In meeting the GDC, the licensee must demonstrate that sufficient spent fuel pool cooling capacity and make-up sources are available during refueling, and time is available prior to pool boiling to supply makeup water following a loss-of-forced cooling.

2.4 Seismic and Structural Integrity Evaluation Regulatory Basis

Appendix A to 10 CFR 50, General Design Criterion 2, "Design bases for protection against natural phenomena," and General Design Criteria 4, "Environmental and dynamic effects design bases," are the fundamental regulatory bases governing the seismic and structural qualification of the DCPP cask pit rack and platform structures. Specifically, the cask pit rack and platform structural evaluation used the DCPP licensing basis load combinations, acceptance criteria, and methodology provided in the updated DCPP FSAR. Four earthquake time histories (i.e., Design Earthquake (DE), Double-Design Earthquake (DDE), Hosgri Earthquake (HE) and Long-Term Seismic Program (LTSP) Earthquake) defined in the DCPP FSAR and related documents are used to seismically qualify the cask pit and platform structure. The structural damping values used in the cask pit rack and platform structure seismic analysis were based on Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," 1973. Additional guidance is provided in Sections 3.7 of the SRP, NUREG-800, June 1987.

The criteria, provided in the NRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, and January 18, 1979, amendment thereto, were used for the seismic qualification of the cask pit rack. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NF and Subsection NB limits were used to demonstrate the cask pit rack's compliance with allowable stress and fatigue analysis requirements, respectively.

The criteria including load combinations, analysis methods, and allowable limits provided in (1) the DCPP FSAR, Sections 3.7 and 3.8, (2) American Concrete Institute Building Code Requirements for Reinforced Concrete (ACI 318-63), (3) Code Requirements for Nuclear Safety-Related Concrete Structures, ACI 349-80, and (4) American Institute of Steel Construction (AISC) Manual of Steel Construction, 1969 Edition were used for assessing the seismic adequacy and structural integrity of the DCPP spent fuel pool reinforced concrete structure and the pool liners.

3.0 TECHNICAL EVALUATION FOR CRITICALITY

3.1 Description of the SFP

There are two SFPs located in the fuel-handling building (i.e., one for each reactor). Each pool is 48x58 feet and 46 feet deep. The pools are founded on a 5-foot reinforced concrete mat and surrounded by a 6-foot reinforced concrete wall. The walls and the floor are lined with a 1/8 inch and 1/4 inch stainless steel liner, respectively. The liner acts only as a water barrier and is not a structural member. Each pool contains 16 freestanding spent fuel rack modules containing 1324 storage locations. The proposed cask pit racks will be configured as 12x13 cells with two cells at the ends eliminated for an additional 154 storage positions for a grand total of 1478 positions. The cask will be installed in a 10x10 square foot part of the pool which is recessed about 4.5 feet below the pool floor. Below the pool floor elevation the cask pit is lined with 1/4-inch thick stainless steel and with a 3/4-inch thick carbon steel backing. In order for the cask pit rack to be in the same elevation with the existing spent fuel racks a platform will be installed in the cask pit. The platform is designed with side shims to ensure a tight fit within the four cask pit walls. Likewise, design features assure that the cask will not move in the horizontal direction on the platform (in case of an earthquake) nor tip away from the walls. This cask pit rack will be removed at the end of cycle 16 because the cask pit will be needed to load fuel onto transfer casks.

3.2 Criticality

The criticality calculations were performed by Holtec, the rack vendor. The regulatory requirements for maintaining subcritical conditions in the SFP are in 10 CFR 50.68, "Criticality accident requirements," for nuclear power plants. For the prevention of criticality in SFPs, the requirements are: (1) if credit is taken for soluble boron, the effective multiplication factor (k_{eff}) shall be less than or equal to 0.95 with the pool fully flooded with borated water and (2) if the pool is flooded with unborated water, k_{eff} must be less than 1.0.

3.2.1 Codes and Methodology

The criticality analysis was performed using the Los Alamos Monte Carlo Code 'MCNP4a.' The code has been used, benchmarked, and verified extensively, for pool criticality calculations. The benchmark parameters include: enrichment, soluble boron concentration, lattice spacing, fuel pellet diameter, and solid boron loading.

Associated fuel depletion calculations were performed using the CASMO-4 Code. CASMO-4 is a two and a half group neutron diffusion code which has been benchmarked and is widely accepted for depletion calculations. The depletion calculations determine the fuel isotopic composition and the associated reactivity effect. In addition, CASMO-4 is used to estimate uncertainties due to fuel and rack fabrication tolerances.

In performing the calculations a number of conservative assumptions are made. For example, reflective boundary conditions are used to simulate an infinite array, during depletion a representative fuel assembly is chosen with a full compliment of poison rods, the water temperature is set at 38 °F to maximize moderation because the moderator temperature coefficient in the pool is negative. To determine the maximum k_{eff}, the fabrication uncertainties,

the computer code bias, and the calculational uncertainty are statistically combined and added to k_{eff}. This is consistent with the requirements of 10 CFR 50.68. The objective of the criticality calculations is to assure that the boron concentration is adequate to retain k_{eff} at or below .95 under normal and accident conditions when flooded with borated water and that k_{eff} is less than 1.0 if flooded with unborated water. The NRC staff finds that the methodologies used for criticality calculation are acceptable because they have been benchmarked and are generally accepted for similar calculations.

3.2.2 Criticality Calculations for Normal Conditions

The licensee stated that the fuel assemblies to be loaded in the proposed rack have decayed for at least 10 years. These are burned assemblies of the standard Westinghouse fuel (LOPAR) and the Westinghouse optimized fuel assembly fuel (VANTAGE 5). The spent fuel rack will be located in the loading cask area away from the existing racks. The minimum distance is more than a foot and with a thermal neutron mean-free path of about an inch (or less in borated water) there is no question of neutronic coupling of the new rack with the existing fuel. Therefore, the existing racks are not considered in the criticality analysis. Criticality control is achieved using Metamic (a trade name) boron absorber panels containing an areal boron-10 density of .0185 g/cm^2 which are held in place and protected against damage with stainless steel sheathing. All fuel assemblies in these racks are surrounded with Metamic absorber sheets. The results of the criticality calculations demonstrate that fuel assemblies with less than 4.1 percent initial enrichment and minimum burnup of 28.53 gigawatt days per metric ton (GWD/MT) maintain k_{eff} less or equal to .95, if flooded with borated water and less than 1.0, if flooded with unborated water.

3.2.3 Criticality Calculations for Abnormal and Accident Conditions

The reactivity effects of abnormal or accident conditions in the pool are examined, in the context of the double contingency principle. The double contingency principle (NRC letter of April 14, 1978) specifies that it shall require at least two unlikely independent and concurrent events to produce a criticality accident. Therefore, it is not necessary to consider more than one accident condition for this analysis. The following accident conditions were considered:

! Temperature and Void Effects

  CASMO-4 analysis demonstrated that the pool has a negative temperature coefficient, therefore, the value was calculated at 38 °F the maximum water density and maximum moderation. This temperature is very conservative and unlikely to be achieved in the pool.

! Dropped Assembly - Horizontal

  In this case an assembly is assumed to have been dropped on the top of the cask pit racks. Accounting for the maximum deformation of the rack the minimum separation from the fuel assemblies is greater than a foot, thus, due to neutronic separation the k_{eff} value will not be affected.

! Dropped Assembly - Vertical

The vertical drop could cause a deformation at the top of the rack, be dropped in an empty position or on the side of the rack. The maximum calculated deformation is relatively small and inconsequential for k_{eff} . However, either case is bounded by the misloaded assembly, therefore, no further analysis has been performed for the vertical drop.

! Misloaded Fresh Fuel Assembly

Misloading a fresh assembly of the maximum possible enrichment of 5.0 percent into the fuel rack will need a minimum of 800 ppm of boron. The DCPP technical specifications (TSs) require a minimum boron dilution of 2000 ppm. This is more than adequate to maintain k_{eff} # .95. The case of concurrent dilution and assembly misloading is not required to be analyzed per the double contingency principle.

! Mislocated Fresh Fuel Assembly

The possibility of a mislocated assembly that is to be placed outside the rack within neutronic coupling distance, does not exist because it is geometrically impossible. Thus, this case is not credible.

The results of criticality analysis for normal and accident conditions as described above demonstrated that: (1) k_{eff} is less than 1.0 under normal pool conditions when the pool is flooded with unborated water and (2) that k_{eff} # .95 under accident conditions when flooded with borated water of at least 800 ppm boron concentration. The analysis demonstrated that the rack is qualified for storage of spent assemblies as indicated in the proposed TS Figure 3.1.7-4. The criticality analysis was performed with benchmarked and generally accepted codes, with conservative assumptions and the results regarding k_{eff} satisfy the requirements of 10 CFR 50.68. The analysis used the double contingency principle which is common practice and is acceptable.

The conclusions reached above were based on generic test results performed by the cask vendor. The results are reported in the Holtec Report HI-2043162.

3.3 Thermal-Hydraulic Considerations

The SFP cooling and cleanup system consists of two independent cooling systems, one for each pool. The piping is class 1 and circulation is powered by two parallel full capacity pumps discharging to a single-shell heat exchanger and powered from a class 1E electrical source. Only one pump is operated at a time, providing single-failure protection. The pumps take suction 4 feet below the normal pool level through a strainer. The connections to the pool are provided with anti-siphon devices to preclude inadvertent draining of the pool. In addition, the piping is arranged so that failure of any pipe will not drain the pool below the level required for shielding.

The current SFP design basis supports partial (76 and 96 assembly) and full (193 assembly) core offload. It is assumed that all offloads begin 100 hours after shutdown and proceed at the rate of 4 per hour. This assumption is conservative in that it maximizes decay heat. Plant administrative controls are in place to limit SFP bulk temperature to 140 °F at all times. For the 76, 96, and 193 assembly offloads, the local maximum water temperature is 188 °F, 194 °F, and

220 °F, respectively. The corresponding cladding temperature maxima are 225 °F, 231 °F, and 254 °F, respectively. At these temperatures localized nucleate boiling will take place but no bulk boiling occurs.

Holtec re-analyzed a partial offload for 96 assemblies, a full core offload with 101 assemblies having a burnup of 52,000 GWD/MT and 92 assemblies with 25,000 GWD/MT, and an emergency full core offload commencing 36 days into the cycle with 113 assemblies with burnup of 40,000 GWD/MT and 113 assemblies with 3,000 GWD/MT. In addition, an estimate is provided for the minimum time to boil and maximum boil-off rate. The above scenarios are conservative with respect to decay heat load. In addition, conservative assumptions are made regarding heat exchanger performance and heat exchanger cooling water temperature.

3.3.1 Methodology

The thermal hydraulic analysis was based on the BULKTEM Code which incorporates the ORIGEN2 Code for fission product decay heat as a function of time. The vendor states that BULKTEM is benchmarked and quality assurance validated. No particular information was provided regarding benchmarking or quality assurance, however, the NRC staff has reviewed and accepted numerous Holtec products. Therefore, a separate review of the Holtec methodology and codes was deemed unnecessary.

3.3.2 Thermal-Hydraulic Results

The calculated maximum bulk temperature for a partial (96 assemblies) offload is 127 °F versus 150 °F for the current design basis. Likewise, the full core offload maximum bulk temperature is 157 °F versus 174 °F for the current value. The emergency full core offload maximum bulk temperature is 162 °F (not part of the current design basis) which is less than the existing analysis for the full core offload of 174 °F.

For the emergency core offload, assuming that forced circulation was lost when the peak bulk temperature is reached, the calculated minimum time to bulk boiling is 3.76 hours and the corresponding boil-off rate is 87 gpm. Both values are bounded by the current full core offload of 2.5 hours and 93.6 gpm.

To calculate the local water temperature Holtec used the commercially available code FLUENT. Holtec stated in the Holtec Report that FLUENT has been benchmarked under their own quality assurance program. To simulate a conservative scenario, the highest bulk temperature, the highest decay loads and the highest flow resistance were assumed. In addition, the flow resistance was increased by assuming an assembly lying horizontally on top of every cell in the rack.

A separate calculation was then performed to determine the fuel clad superheat, which was then added to the local temperature to determine the peak fuel cladding temperature. The calculated peak local water temperature was calculated to be 188 °F, which is below the water saturation temperature of 240 °F at the fuel depth.

3.3.3 Administrative Controls

The DCPP administrative controls limit the SFP bulk water temperature to 140 °F. Procedural controls suspend offload activities at bulk pool temperature of 125 °F so as not to exceed 140 °F. The submittal states that operating experience has shown that offload temperatures typically do not exceed 115 °F.

3.3.4 Revised SFP Thermal Licensing Basis

The licensee requests that the SFP thermal hydraulic calculations become the calculations of record. The staff finds that the proposed calculations were performed using benchmarked codes using conservative assumptions. The calculations include an emergency core offload which is more conservative that the full core offload and is not included in the existing licensing basis. In addition, after the removal of the temporary cask at the end of cycle 16, the pool load will be even more conservative. Therefore, the NRC staff finds that the proposed thermal hydraulic analysis qualifies to replace the SFP thermal hydraulic analysis of record.

3.4 Technical Specification Changes

TS 3.7.17, "Spent Fuel Assembly Storage," is modified to reflect and accommodate the new rack in the SFP. The TSs explicitly state the major assumptions, i.e., enrichment less than or equal to 4.1 percent, limited applicability to Cycles 14-16, minimum decay time of the assemblies to be used in the new rack of 10 years, and the new total number of assemblies to be stored in each pool of 1433. The added new TS Figure 3.7.17-4 also reflects these limitations. TS 4.3, "Fuel Storage," distinguishes permanent and the new temporary storage and lists the limitations applicable to the new storage cask. The NRC staff's review finds that the proposed TS changes correctly and completely represent the proposed modifications and, therefore, are acceptable.

3.5 Summary and Conclusions

The NRC staff reviewed the submitted information regarding spent fuel pool modifications at the DCPP and finds that the proposed addition of a temporary new cask pit spent fuel storage rack for cycles 14 through 16 is acceptable. This finding is based on: (1) the methodologies used for the criticality and thermal hydraulic calculations are based on benchmarked and widely accepted computer codes; (2) the assumptions of the calculations are conservative; and (3) the proposed TS changes correctly represent the proposed technical changes.

4.0 TECHNICAL EVALUATION FOR CRANE AND HEAVY LOADS

4.1 Handling of Heavy Loads

The licensee stated that cask pit rack will enter the Fuel Handling Building (FHB) through the roll-up door into the receiving area of the cask wash down area. The rack module will be removed from the shipping trailer in the horizontal position and then uprighted into a vertical position using lifting devices meeting NUREG-0612 guidelines. The 125-ton-rated FHB crane will be used for lifting the new rack and platform into the respective FHB. The maximum lift weight during installation and removal of the cask pit rack will be 28,300 lbs. The maximum lift weight during the installation of the platform will be 26,625 lbs.

Since the installation and removal of the cask pit rack and installation of its associated platform will involve handling of heavy loads in the vicinity of the SFP, this process will be performed consistent with PG&E's Heavy Loads Program commitments. NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” dated July 1980, provides recommendations and guidelines to assure safe handling of heavy loads in the proximity to or over safe shutdown equipment or irradiated fuel in the spent fuel area. The guidelines are meant to ensure that either (1) the potential for a load drop is extremely small, or (2) the consequences of a postulated accidental load drop do not result in the violation of radiological or criticality limits, or compromise safe shutdown. The NRC staff previously evaluated the DCPP program for control of heavy loads and concluded that it is in compliance with the guidelines of NUREG-0612.

The licensee stated that the cask pit rack and platform will not be suspended over any portions of the SFP containing spent fuel assemblies. The licensee intends to vacate a minimum of one row of cells in the adjacent permanent racks. Vacating one row surrounding the cask area, combined with the separation provided by the cask seismic restraint, will create a horizontal separation distance between fuel assemblies stored in the pool and the projected vertical lift envelope of the cask pit rack or platform. This distance will provide a margin to ensure that a postulated drop will not impact stored fuel.

The licensee also stated that the procedures covering the handling of heavy loads will be revised as necessary and new procedures will be developed, specifically for the cask pit rack, platform, and related heavy load lifts and handling in accordance with PG&E's program requirements. These procedures will be comprehensive with respect to load handling, exclusion areas, equipment required, inspection and acceptance criteria before load movement, and steps/sequence to be followed during load movement, as well as safe load paths and special precautions.

NUREG-0612, Section 5.1.2, recommends that in addition to satisfying the general guidelines of Section 5.1.1, one of four criteria be met. One of the criteria specifies that the overhead crane and associated lifting devises used for handling heavy loads in the spent fuel pool area to satisfy the single-failure-proof guidelines of Section 5.1.6 of NUREG-0612. Another of the criteria specifies that the effects of drops of heavy loads should be analyzed and shown to satisfy the evaluation criteria of Section 5.1 of the same report.

Prior to installation of the new cask pit rack and associated platform, PG&E intends to upgrade the FHB crane in accordance with the implementation guidelines of NUREG-0612, Appendix C,

under the provisions of 10 CFR 50.59. Since the current crane is not fully compliant with NUREG-0612, Appendix C, the licensee has also performed heavy load drop analysis to demonstrate conformance with NUREG-0612, Section 5.1.2.4.

4.1.1 Drop Analysis

The Holtec report addresses a cask pit rack drop and a support platform drop. The drop analyses took into consideration the drop height and orientation. The licensee stated that because of the limited gap between platform and the cask pit, the analysis only considers the vertical orientation of the platform. The NRC staff considers that this assumption is acceptable because the spent fuel pool walls and the cask pit walls restrain the support platform from horizontal movement.

The NRC staff reviewed the licensee's submittal and forwarded two requests for additional information (RAI) to the licensee. The RAI requested the licensee to:

a. provide the basis for this assumption and elaborate as to why is conservative to use a lower elevation on the analysis opposed to the highest elevation at which the components are maneuvered,

b. explain how the results of the analysis are impacted if the components fall from the highest elevation at which they are maneuvered.

In its letter dated June 23, 2005, the licensee stated that the final velocities of the rack and platform, just prior to impact, have been computed for different combinations of drop elevation and SFP water level in Holtec Report HI-2043219 (Analysis of Postulated Mechanical Accidents at the DCPP Spent Fuel Pool Cask Pit), Revision 4. The new computed results show that a rack drop from the highest elevation (i.e., 141 feet) has no impact on the results presented in the Holtec Licensing Report HI-2043162, Revision 1. The rack platform exhibits only a slight increase in impact velocity with increasing drop height in air. The percentage increase between the previous calculated results and the revised analysis is less than 1 percent, which is not enough to develop through-thickness cracks in the concrete slab and/or punch through the liner considering the large margins of safety reported in Holtec Licensing Report HI-2043162, Revision 1, for the lower drop elevation (i.e., 139 feet). Note that the calculated results conservatively assume no velocity loss as the platform transitions from air to water. Finally, the LS-DYNA model used in the Holtec Licensing Report HI-2043162, Revision 1, to predict concrete cracking in the cask pit slab does not take any credit for steel reinforcement.

Based on the review of the licensee's submitted information, the NRC staff finds the Holtec Report HI-2043219 includes the considerations and assumptions stated on NUREG-0612, Appendix A. Therefore, the NRC staff concludes that the licensee's proposed installation of the cask pit rack conforms with the guidelines of NUREG-0612.

4.2 SFP Cooling

Each unit has a completely independent SFP cooling and cleanup system. The design of the cooling loop conforms to Design Class I piping criteria. The SFP pool cooling systems consist of two parallel full-capacity pumps discharging to a single shell and tube heat exchanger. Only

one pump is operated at a time, providing for a single active failure without any reduction on system capacity. The SFP cooling pumps are powered from a Class 1E source.

The SFP cooling system heat exchanger is cooled by the component cooling water system, which in turn is cooled by the auxiliary saltwater system that rejects waste heat to the Pacific Ocean. The SFP water is pumped from the pool through the tube side of the SFP heat exchanger and returned to the pool. The pump suction line is protected by a strainer and is located four feet below the normal SFP water level. The connections to the SFP are provided with anti-siphon devices to preclude possible draining of the pool water. The piping of the SFP cooling system is arranged so that failure of any pipe will not drain the SFP below the level required for acceptable radiation shielding.

4.2.1 Current Spent Fuel Pool Licensing Basis

The spent fuel pool cooling analysis includes evaluation of partial core off-load cases of 76 and 96 fuel assemblies, and also a full core, 193 assembly, off-load case. These cases were analyzed assuming maximum heat loads when the heat exchanger is supplied with component cooling water at the design flow and maximum postulated temperature.

In the 76 assembly partial off-load case, the analysis assumed that the pool contained spent fuel from 15 previous discharges of 76 assemblies each at 18-month intervals. With an assumed 19-hour off-load time, the analysis showed that the cooling system can maintain the spent fuel pool water temperature at or below 140 /F.

The 96 assembly partial off-load case assumed that the pool contained spent fuel from 12 previous discharges of 96 assemblies each at 24-month intervals. With an assumed 24-hour off-load time, the analysis showed that the pool temperature would not exceed 150 /F.

For the full core off-load case of 193 assemblies, the analysis assumed that the pool contained spent fuel from 15 previous discharges of 76 assemblies each at 18-month intervals. With an assumed 48-hour off-load time, the analysis showed that the pool temperature would not exceed 175 /F. The criticality analysis performed in support of the 5 percent enrichment license amendment request credits the presence of soluble boron in the spent fuel pool at temperatures above 150 /F to ensure that k_{eff} remains below 0.95. However, the NRC staff cited a limit of 140 /F for the spent fuel pool temperature in the re-rack safety evaluation report for a "normal" refueling, which is a full core off-load. Administrative controls are in place to ensure that the spent fuel pool temperature remains below the 140 /F licensing limit.

Analyses were also performed using conservative assumptions to determine the maximum local water temperatures and maximum fuel cladding temperatures for the above offload scenarios. For the 76-assembly partial offload case, the analysis showed that the maximum local water temperature was 188 /F and the maximum cladding temperature was 225 /F. For the 96-assembly partial offload case, the analysis showed that the maximum local water temperature was 194 /F and the maximum cladding temperature was 231 /F. For the full core offload case of 193 assemblies, the analysis showed that the maximum local water temperature was 220 /F and the maximum cladding temperature was 254 /F. For the limiting full core offload scenario, the maximum cladding temperature would result in localized nucleate boiling but not bulk pool boiling.

A complete loss of the SFP cooling system for an extended period of time was evaluated. The analyses demonstrated that boiling would not occur for approximately 9 hours for partial core offload scenarios and would commence after 2.5 hours for a full core offload scenario.

4.2.2 Thermal Hydraulic Analyses

The licensee stated that the rack vendor, Holtec, developed thermal-hydraulic analyses to determine the peak SFP bulk temperatures, maximum local water, and fuel assembly cladding temperatures with a new cask pit rack installed on each unit. These reports were generated using more accurate computer models than the previous analyses.

The cask pit rack thermal-hydraulic analyses are based on the evaluation of three offload scenarios that bound the past and future operating practices at DCPP:

1. a partial core offload scenario;
2. a full core offload; and
3. an emergency full core offload 36 days after completion of a refueling outage.

All scenarios are evaluated to occur during or, in the case of the emergency offload scenario, shortly after the 15th refueling outage. The number of irradiated fuel assemblies assumed to be stored in the SFP in each of the evaluated scenarios conservatively bounds the actual number of irradiated fuel assemblies that will be stored in the SFP when the cask pit rack is installed.

All of these scenarios have been evaluated with a base decay heat load contribution from previously discharged fuel assemblies using actual operational data for operating Cycles 1 through 11. The contribution to the base decay heat load from fuel that will be discharged in Cycles 12 through 14 is based on an assumed discharge of 104 assemblies each cycle using bounding assumptions on fuel assembly burnup and operating power.

4.2.3 Maximum SFP Bulk Temperatures

The steady-state decay heat load from previously offloaded fuel was calculated using the LONGOR computer program. Inputs to the program were based on known power histories for fuel discharged in Cycles 1 through 11 and a projected fuel offload schedule for Cycles 12 through 14 that conservatively bounds both fuel assembly burnup and the number of fuel assemblies to be offloaded. The decay heat contribution for fuel discharged in Cycles 1 through 11 was based on Unit 2 data since Unit 2 contained more spent fuel assemblies than Unit 1 following the 11th refueling outage (928 versus 900) and also operated at a higher licensed thermal power than Unit 1 (3411 versus 3338 MWth) during this same time period. Additionally, the assumed reactor thermal power was increased by 5 percent to account for burnup uncertainties. This results in a conservatively high estimate of the base decay heat load, which bounds either Unit.

The transient decay heat contribution for each offload scenario is determined using the Holtec quality assurance (QA) validated computer program BULKTEM, which incorporates the ORIGEN2 code for performing decay heat calculations. Each offload is assumed to start at 100 hours after reactor shutdown with an assumed offload rate of four assemblies per hour. Conservative assumptions were made with respect to operating power and fuel burnup to determine a bounding decay heat load contribution for the offloaded fuel.

The partial core offload analysis resulted in a maximum pool bulk temperature of 127 /F. The full core offload analysis resulted in a maximum pool bulk temperature of 157 /F. Both of the scenarios resulted in maximum pool bulk temperatures less than the analyses supporting the current licensing basis, i.e., 150 /F for partial offloads and 174 /F for a full core discharge. The emergency core offload analysis resulted in a peak bulk temperature of less than 162 /F. Although not part of the current licensing basis, the emergency offload results were less than the previous analysis results for a full core discharge (174 /F).

The analyses show that the peak SFP temperatures could reach 162 /F, but the licensee stated that they will continue to maintain administrative controls in place to ensure that peak SFP temperatures remain below 140 /F during the normal full core offload scenario, consistent with the current licensing basis.

The NRC staff has reviewed the licensee submittal and finds that the heat load reduction obtained by using a more precise code for heat load calculation is acceptable and the resulting peak pool temperature remains bounded by that given in the current licensing basis. The licensee also requests that the NRC staff approve the Holtec thermal-hydraulic analysis as the licensing basis of record for future spent fuel storage requirements and the NRC staff find this analysis acceptable.

4.2.4 Minimum Time to Boil

The licensee has performed analyses to demonstrate minimum time-to-boil and the maximum boil-off rate. The time-to-boil evaluation assumed that forced cooling was lost the moment the peak SFP bulk temperature for each case was reached. The SFP time-to-boil and corresponding maximum boil-off rates were then determined. For the worst-case scenario, the emergency core offload, the calculated time-to-boil was determined to be 3.76 hours after a loss-of-forced cooling at the peak SFP bulk temperature. The corresponding maximum boil-off rate for this condition was approximately 87 gpm. Both of these values are bounded by the current thermal-hydraulic analysis values of 2.5 hours and 44,905 lbm/hr (93.6 gpm).

The licensee states that there is adequate time to align and supply sufficient water, from a variety of sources, to the SFP prior to the time-to-boil. Make up water from the condensate storage tank at a rate of up to 200 gpm can be added directly to the SFP by a Design Class 1 piping. The transfer tank is another Design Class 1 available source of make up water to the spent fuel pool. Based on the above, the NRC staff finds the time-to-boil analysis acceptable.

4.3 Conclusion

The NRC staff concludes, based on the review of the licensees submitted information on the handling of heavy loads associated with this amendment request, that licensee has provided adequate assurance that their planned actions for the handling of heavy loads for the installation of the cask pit storage racks are consistent with the "defense-in-depth" approach to safety described in NUREG-0612. In addition, based on the considerations discussed above, the NRC staff concludes that there is adequate cooling water flow to the SFP heat exchanges to remove the decay heat generated by the spent fuel in the pool during normal and abnormal offload conditions. The NRC staff also finds that the licensee has sufficient time and capability, prior to the onset of boiling, to align make up water to the pool, and provide make up at a rate in

excess of the boil-off rate, thus satisfying GDC 61 with respect to maintaining the fuel covered with water under accident conditions. Therefore, the NRC staff finds the amendment request acceptable in regards to the SFP thermal-hydraulics and the handling of heavy loads.

5.0 TECHNICAL EVALUATION FOR SEISMIC AND STRUCTURAL INTEGRITY

5.1 Summary of The Licensee's Seismic and Structural Evaluation Results

The seismic design adequacy and structural integrity evaluation of the cast pit rack and platform structure provided in the license amendment request (LAR) and Enclosure 5 of the LAR discusses several key considerations including: (1) structural characteristics of the new cask pit rack/platform structure, (2) applicable load combinations and seismic time history analysis, (3) analysis methodology and dynamic modeling of the cask pit rack/platform structure, (4) kinematic and stress acceptance criteria and dimensionless stress factors, (5) dynamic simulation and seismic response of the rack/platform structure, (6) assessment of rack/platform structure design margins, and (7) assessment of the margin against fatigue failure. The results of the dynamic simulations demonstrated that the cask pit rack does not impact the SFP walls or the cask seismic restraint framework. The stress levels predicted for the cask pit rack and platform satisfy the applicable criteria of the ASME B&PV Code, Section III, Subsection NF and the calculated cumulative usage factors associated with the fatigue analysis of the cask pit rack and platform assembly satisfy the applicable criteria of the ASME Code, Section III, Subsection NB.

The SFP floor slab, walls, and liner plate were evaluated by the licensee and found to be adequate to support and transfer the reaction loads from the rack and platform. Stresses in the wall and floor slab were determined to be within normal and shear strain allowables, and stresses in the bedrock were found to be within the allowable bearing pressure.

The auxiliary building, which includes the SFPs, was seismically qualified using the criteria outlined in Chapter 3 of the DCPP FSAR Update. The effect of the change in weight on the seismic models due to the cask pit rack was determined to be insignificant and the licensee asserted that there was no change in the seismic responses and forces reported in the FSAR.

5.2 Technical Evaluation and Staff Disposition of Request for Additional Information

With respect to Section 6.0, "Rack Structural Integrity Considerations" of Enclosure 5 to the LAR, the NRC staff raised a number of technical questions that required further information or clarification. The licensee was requested to address the technical questions via an NRC staff RAI. In its letter to NRC dated June 23, 2005, the licensee responded to the RAI. The following provides a discussion of each item in the RAI, licensee response, and the staff disposition of the issue of interest in the RAI.

5.2.1 Modeling Features for the Rack/Platform Structure

Since the rack/platform structure inside the recessed cask pit of the spent fuel pool is different in its structural configuration from the familiar freely standing single rack structure on a spent fuel pool floor, the NRC staff asked the licensee to discuss any unique modeling features employed for the rack/platform structure seismic analysis that support the validity of the seismic responses reported in Tables 6.8.1 and 6.8.2 of the Holtec report.

In its response, the licensee indicated that while the cask pit rack/platform structure seismic analysis does employ some unique modeling features as compared to that of the typical spent fuel rack seismic analyses, all modeling has been performed within the framework of the existing DYNARACK code. DYNARACK is a general purpose dynamics program, which has been validated by Holtec under their 10 CFR 50, Appendix B, Quality Assurance Program, that allows the user to build models using a combination of lumped mass and spring elements in any configuration. What makes the cask pit rack/platform structure model different from the familiar single rack model is that additional mass and spring elements have been added to incorporate the platform structure. The platform structure is modeled in DYNARACK as 6 degrees-of-freedom (i.e., 3 translation degrees of freedom and 3 rotation degrees of freedom) rigid body with the proper overall dimensions and mass properties. Consequently, the total number of degrees-of-freedom in the cask pit rack/platform structure model is 28 versus only 22 in the typical single rack model. The contact interfaces between the platform and the cask pit rack and the platform and the cask pit walls are modeled using gap elements and friction elements in a manner very similar to that utilized in modeling the spent fuel rack support pedestals and the cask pit rack-to-wall gaps (see Figure 6.5.3 of the Holtec Report HI-2043162 for schematic of interface springs). Since the platform is an open frame structure that is firmly wedged in the cask pit when installed, there are no fluid coupling effects associated with the platform.

The licensee further indicated that the seismic analyses of the cask pit rack/platform structure using DYNARACK is not intended to resolve the stresses in the platform structure. The purpose rather is to: (a) show that the cask pit rack/platform model does not impact the SFP walls at top of rack, (b) compute the stress factors in the cask pit rack cell structure, and (c) determine the maximum interface forces acting on the platform during a seismic event. The maximum interface forces are then used to calculate the stresses and safety factors in the platform using ANSYS and basic strength of materials formulas. Thus, only the results in Holtec Report HI-2043162, Table 6.8.1 are directly obtainable from DYNARACK. The results in Holtec Report HI-2043162, Table 6.8.2 are derived from a three-dimensional, linear elastic finite element model of the platform structure using the maximum interface forces from DYNARACK as input. Therefore, the modeling of the platform in DYNARACK as a 6 degrees-of-freedom rigid body with appropriate interface springs is sufficient for the intended purposes and well within the proven capabilities of the DYNARACK code.

The NRC staff finds the above licensee's response adequate and acceptable for closure because the analysis results are obtained based on the use of properly validated computer codes, i.e., DYNARACK and ANSYS, and the structural modeling and the stress analysis method adopted are consistent with applicable FSAR and SRP provisions.

5.2.2 Variability of Analysis Results

Considering that the cask pit rack/platform model analysis methodology used in Section 6.0 of the Holtec report is based on many engineering assumptions (see Section 6.5.2) in its theoretical development, the NRC staff requested the licensee to provide a summary discussion of the potential variabilities of the specific analysis results reported in Tables 6.8.1 and 6.8.2, and the DCPP's basis for judging that these variabilities are adequately accounted for in the design of the cask pit rack and the platform structures.

The licensee responded that the analysis for the spent fuel rack has been developed and improved upon by Holtec International over a period of many years and is used for determination of global displacements of spent fuel racks and rack-to-rack and rack-to-floor reaction forces. Over the years, the methodology has been continually improved to minimize the effect of engineering variabilities. Section 6.5 of the Holtec Report HI-2043162 provides a brief description of the theoretical concepts and conservative assumptions used in the development of the cask pit rack/platform model. The building block for the simulation is the structural model of a single spent fuel rack. Each rack is modeled as a beam-like structure with 12 degrees-of-freedom. The formulation is set up so that classical beam theory (including shear deformation) static solutions are reproduced when the rack model is subject to end loadings. Therefore, to the extent that the "beam properties" of the spent fuel rack are modeled accurately, the predicted results are consistent with the accuracy of beam theory applied to any structure. Essentially, the spent fuel rack is a rugged, nearly rigid honeycomb structure whose beam properties (area and moments of inertia) can be developed with minimal assumptions. In terms of its global response, it can be characterized as a "nearly rigid" structure. Therefore, the results that are obtained from any simulation will differ from the results obtained by considering the racks as rigid (with known mass and inertia properties) only to the extent of the "improvements" included in the characterization of the beam-like deformations. If the behavior of spent fuel racks in a dry pool were being studied, it would be concluded that the results have minimal variability as they are founded in the well-known precepts of multi-body rigid body dynamics. Any variabilities would, in that case, come only from the numerical solution procedure and the step size of integration. Based on numerous convergence studies that have been performed over the years, the results obtained reflect the reality of the scenarios under study.

The licensee stated that since the real scenarios of interest involve racks under water, it is recognized that the responses obtained are very dependent upon the approach used to simulate the hydrodynamic coupling between rack and SFP walls, and between individual fuel assemblies and rack cell walls. The basis of the simulation of fluid coupling effects is a classical analytical solution of a moving fluid-filled cylinder that contains a smaller moving solid cylinder; coupling between the bodies is provided by the fluid filled annulus (See Reference 6.5.7 of Holtec Report HI-2043162, "The Effects of Liquids on the Dynamic Motions of Immersed Solids"). Holtec has extended this solution to the case of rectangular bodies and then further extended the solution to cover multiple rectangular bodies. Recognizing the extent to which the theory was extended, multiple test cases were solved to ensure that the formulation could be supported by both existing and new experimental results. At the request of the NRC staff, Holtec demonstrated that the methodology gave results in full agreement with the experimental results performed at Carnegie Mellon. This comparison was submitted under the Waterford 3 Docket. Holtec also performed its set of experiments involving multiple rectangular bodies

submerged in a fluid and subject to dynamic motion. This data was also submitted under the Waterford 3 docket (contained in Reference 6.5.6 of Holtec Report HI-2043162, “Fluid Coupling in Fuel Racks: Correlation of Theory and Experiment”). The excellent agreement of Holtec’s fluid coupling methodology with all of the available experimental work provided the necessary confidence to utilize it in all of Holtec’s spent fuel re-rack projects. Holtec expects that its experimental work-based fluid coupling methodology would provide results suitable for making sound and practical engineering assessments of the viability of a proposed re-rack design. Holtec recognizes that no analytical approach can exactly predict what will happen to an assemblage of real structures submerged in a seismically excited SFP. Therefore, to the extent possible, all engineering assumptions are made in a manner that ensures that the results will overpredict displacements, forces, etc., and, therefore, will produce conservative estimates of safety factors. In addition, to minimize the effects of variability, adequate safety factors are maintained. The results for the DCPP cask pit rack and platform structure have at least a 15 percent margin to the design-basis limit.

Based on the above discussed theoretical development, experimental verification, and numerous independent reviews of the Holtec methodology (Franklin Institute Research Laboratories, Brookhaven National Laboratories), the licensee concluded that the results should be within what would be accepted as good engineering accuracy, and any variation would likely to be on the conservative side (i.e., over-predictions). To further ensure that variabilities in the results would not inadvertently lead to an adverse conclusion, the licensee stated that structural margins are maintained that are well below the design limit.

The NRC staff finds the above licensee's response adequate and acceptable for closure because adequate amount of engineering analysis results including experimental data and past NRC-approved licensing precedents were provided by the licensee in support of its use of the DYNARACK code.

5.2.3 Approach for Developing Storage Rack Analysis Stiffness Matrices

Referring to Figure 6.5.3, "Schematic of the Dynamic Model of Cask Pit Rack Platform Used in DYNARACK," the NRC staff requested the licensee to describe the approach used in developing the stiffness matrices for: (a) the platform/cask pit floor gap and friction elements, (b) the platform/cask pit wall gap and vertical friction elements, (b) the rack pedestal to platform gap/impact element, and (d) the tension-only elements representing the connector links.

The licensee responded that the development of the stiffness matrices for the various gap and friction elements mentioned is based on finite element analysis and classical strength of materials and elasticity formulas. More specifically, the stiffness value assigned to the platform/cask pit floor gap element is based on the solution for a semi-infinite solid loaded over a rectangular area found in Timoshenko’s Theory of Elasticity (3rd Edition). The semi-infinite solid has the mechanical properties of concrete, and the load is applied over a 120 square inch area. For the platform/cask pit wall gap element, the local concrete stiffness is combined in series with the local stiffness of the platform as determined using ANSYS. The local concrete stiffness is once again calculated using the same formula from Timoshenko’s text. Due to the design of the platform, the net stiffness value is different at the top and bottom shim locations. The rack pedestal to platform lateral impact stiffness is determined from Roark’s Formulas for Stress & Strain, Table 33, case No. 2c. The stiffness of the tension-only elements representing

the connector links is based on the free length of the shaft considered as a bar (i.e., k = EA/L). Finally, the stiffness value assigned to the friction elements is at least 1,000 times greater than the stiffness of the coincident gap element. The use of a very high stiffness value minimizes the amount of extension/compression in the friction element as the friction force ramps up to its threshold limit, which leads to a more accurate prediction of the overall displacement.

Based on the above licensee response, the NRC staff finds the engineering rationale used in developing the stiffness matrices for the four friction or gap elements identified in the RAI fully responsive to the issue of concern. Therefore, the NRC staff considers the licensee's response to the RAI to be adequate.

5.2.4 Development of Time History Accelerograms from In-Structure Seismic Responses

The NRC staff asked the licensee to explain how the in-structure seismic responses in the vicinity of the spent fuel pool structure (per the DCPP FSAR) for the DE, DDE, HE and LTSP earthquakes were utilized in developing the time-history accelerograms shown in Figures 6.4.1 through 6.4.12.

The licensee indicated that the time-history accelerograms used as input to the seismic analyses of the cask pit rack were developed using synthetic time-history generation software to fit the target floor acceleration response spectra at the location of the spent fuel pools (SFP) in the fuel handling area of the auxiliary building. The licensee stated that the DE, DDE, and HE synthetic time-history accelerograms were developed in 1985 in support of the installation of the high-density spent fuel racks in the DCPP SFPs (reference PG&E Calculation 52.15.58, Revision 0). These time-history accelerograms form part of DCPP's licensing basis and their use is described in DCPP FSAR Update, Section 3.8.8.4, "Design and Analysis of Racks." For the LTSP, three orthogonal spectrum-compatible time-history accelerograms were developed to match the LTSP response spectra in the SFP. The NRC Standard Review Plan spectral matching criteria (Section 3.7.1 of NUREG-0800) were followed. These criteria recommend that 75 frequencies be used for comparison of the response spectrum associated with the time-history accelerogram to the target response spectrum and that the computed spectral acceleration at no more than 5 of the 75 frequency points fall below the target spectrum, and that no point fall below 0.9 times the target spectrum. The time-history accelerograms satisfy these requirements.

The NRC staff finds that the above licensee’s description of how the time-history accelerograms were developed from in-structure seismic responses in the vicinity of the spent fuel pool structure to be complete in scope and consistent with applicable provisions of DCPP FSAR.

5.2.5 Impact of Damage to Cast Pit Rack Based on Drop Analysis

Section 7.2 of the Holtec report discussed the licensee’s analyses performed to evaluate the damage to the new cask pit rack, the pool liner, and the concrete slab in the cask pit area subsequent to the impact of a fuel assembly, a rack or a rack platform under various drop scenarios. Results of the analyses covering shallow drop, deep drop, rack drop, and platform drop events are presented in Section 7.5 of the report. The NRC staff requested the licensee to discuss DCPP's basis for judging that the results obtained from these analyses are directly applicable and reasonable for use in the cask pit rack and platform design.

The licensee responded that all aspects of the drop accident models and evaluations were performed by Holtec using DCPP's specific data based on Holtec's design of the rack and platform and the configuration of DCPP's spent fuel pool liner and concrete. The evaluations of the liner and concrete considered the as-built configuration of these items, including material strengths, weld seam locations, liner thickness, leak detection channel locations, and concrete thickness. Bounding weights and material properties have been used to provide conservative results. The computer code LS-DYNA, which is a commercial code developed by Livermore Software Technology Corporation, has been validated under Holtec's 10 CFR 50, Appendix B, Quality Assurance program through comparisons with documented test cases. This program is ideally suited for the solution of impact problems. The analysis methodology, described in Section 7.4 of Holtec Report HI-2043162, has been applied to drop analyses for numerous spent fuel pool re-racking projects and has been previously accepted by the NRC. The licensee stated that its review of the Holtec load drop analyses included review of the modeling analyses' methodology and assumptions to verify that the analyses were appropriate for use at DCPP and the input assumptions were consistent with plant-specific data. The licensee concluded that the results are applicable and reasonable for use in the cask pit rack and platform design because they are derived from DCPP-specific data using a well-established computer code and methodology.

The NRC staff finds that the above licensee response to the RAI adequate and acceptable because it is based on conservative engineering assumptions, DCPP-specific data, a properly validated computer code and a reasonable drop accident analysis methodology.

5.2.6 Justification of Drop Height

The NRC staff noted that Holtec Licensing Report HI-2043162 Revision 1 states that, "the third and fourth classes of drop events assume that a lifted empty rack and a rack platform fall from the top of the SFP water level and impact the floor of the cask pit, respectively." However, it appears that while maneuvering the empty rack and the rack platform into the cask pit area, both components will be positioned at a higher elevation than the one assumed in the Holtec report. The NRC staff issued an RAI and requested the licensee to: (a) provide the basis for this assumption and elaborate as to why it is conservative to use a lower elevation on the analysis opposed to the highest elevation at which the components are maneuvered and (b) explain how the results of the analysis are impacted if the components fall from the highest elevation at which they are maneuvered.

The licensee responded that the results presented in Holtec Licensing Report HI-2043162, Revision 1, assume that the rack and platform are dropped from the SFP maximum water level (El. 139'-0") and fall a distance of 44 feet 3 inches through water before impacting the top of the cask pit liner (El. 94'-9"). While the empty rack and platform are being manipulated above the SFP water during installation, the maximum lift height of either component will be limited by procedures to the 141'-0" elevation. The licensee stated that the basis for assuming the rack and platform drop from the lower 139'-0" elevation, and not the 141'-0" elevation, is that the additional velocity associated with the 2-foot drop in air will be negated by the energy that is dissipated when the rack/platform impacts the surface of the water. However, in order to demonstrate that a potential drop from the 141'-0" elevation has no significant impact on the results presented in HI-2043162, the final velocities of the rack and platform, just prior to impact, have been computed for different combinations of drop elevation and SFP water level in

HI-2043219 (Analysis of Postulated Mechanical Accidents at the Diablo Canyon Spent Fuel Pool Cask Pit), Rev. 4. The calculation results indicated that a rack drop from the highest elevation (i.e., 141’-0”) has no impact on the results presented in the Holtec Licensing Report and the rack platform exhibits only a slight increase in impact velocity with increasing drop height in air. The licensee determined that the percentage increase between cases considered is less than 1 percent, which is not enough to develop through-thickness cracks in the concrete slab and/or punch through the liner considering the large margins of safety reported in Holtec Licensing Report HI-2043162 for the lower drop elevation (i.e., 139’-0”). Finally, the LS-DYNA model used in the Holtec Licensing Report HI-2043162 to predict concrete cracking in the cask pit slab does not take any credit for steel reinforcement; this conservatism offsets the slight increase in the platform’s final impact velocity. The licensee concluded that the potential drop of either the rack or platform from the highest elevation has no impact on the results presented in Holtec Licensing Report HI-2043162, Revision 1.

The NRC staff reviewed the above licensee’s justification, including the fact that large margins of safety are incorporated in the rack/platform drop analysis and that no credit for steel reinforcement of the cask pit concrete slab was taken in the LS-DYNA analysis. The NRC staff determined that the justification provided is reasonable, consistent with good engineering practices and acceptable for resolving the RAI related to the cask pit rack and platform drop elevation.

5.2.7 Description of Operating and Maintenance Experience

The NRC staff requested the licensee to provide a summary discussion of past DCPP's operating and maintenance experience with respect to its spent fuel racks and pool structural elements including potential pool structural deformation or leakage and damages of racks, fuel assemblies, cask pit floor/walls, and spent fuel pool liner.

The licensee indicated that the structural condition of the SFPs and spent fuel racks are visually monitored by the civil engineering group in accordance with Procedure MA1.NE1, "Maintenance Rule Monitoring Program - Civil Implementation." Degradation of these items has not been identified during these inspections. In addition, no significant misalignment or deformation has been identified by operations during fuel handling activities in the SFP. The DCPP's SFPs are concrete structures with a stainless steel liner. Each SFP includes an integral leakage detection and collection system consisting of a network of monitoring trenches that drain to a collection system of six individual collection pipes, terminating in a valve and quick disconnect at the monitoring location. Each collection pipe is individually sampled at the quick disconnect for fluids on a weekly basis. Once per quarter, samples from each collection pipe are analyzed for iron content. The licensee further indicated that in 1988 SFP liner leaks were identified in Unit 1 and were repaired by welding stainless plates over the leaks. No significant leakage has been observed since this repair. Unit 2 has experienced 200-300 ml leakage per week from one of the sample points. The other sample points have exhibited no appreciable leakage. The leakage detection system on both units appears to be free-flowing with no blockage. There has been no evidence of leakage from the SFP through the concrete structure. No chemical compounds have been detected that would indicate degradation of the SFP concrete or reinforcing steel. Additionally, material degradation of the concrete due to any leakage would be negligible since the chemical reaction between the concrete and effluent (Boric acid) would cause negligible degradation of the concrete as reported in ACI 515.1, Table 2.5.2, A Guide to

the Use of Waterproofing, Damp-proofing, Protective, and Decorative Barrier Systems for Concrete.

Since the structural condition of the SFPs and spent fuel racks are visually monitored by the DCCP's civil engineering group in accordance with its Maintenance Rule Monitoring Program and no degradation, significant misalignment or deformation of these items has been identified during these inspections; SFP liner leaks were identified and promptly repaired; no evidence of leakage from the SFP through the concrete structure was observed; the leakage detection system on both units appears to be properly functioning with no blockage; and no chemical compounds have been detected that would indicate degradation of the SFP concrete or reinforcing steel, the NRC staff finds the above response adequate and acceptable.

5.2.8 Changes to Spent Fuel Pool Administration Controls

The NRC staff asked the licensee to provide a discussion of any needed modification to existing DCPP spent fuel pool operation related administrative controls in order to implement the proposed revision to technical specifications 3.7.17 and 4.3 for cycles 14-16.

The licensee responded that a specific procedure will be developed and approved in accordance with the DCPP's QA Program (FSAR Update, Section 17.5) for controlling the fuel that will be allowed to be placed in the cask pit storage rack. This procedure will ensure that only fuel with an initial enrichment of less than or equal to 4.1 weight-percent U-235, a minimum 10 year decay time, and a discharge burn up in the acceptable region of Technical Specification Figure 3.7.17-4 is stored in this rack. This procedure will fully implement all applicable requirements of Technical Specification 3.7.17 and 4.3 for the cask pit storage rack. The licensee indicated that once the cask pit racks have been installed, the rack on each Unit will be fully loaded with spent fuel assemblies from the existing inventory of fuel in the spent fuel pools prior to the 14th refueling outage on each Unit. It is not the licensee's intent to utilize the cask pit racks in the management of the spent fuel during subsequent refueling outages. Thus, it is intended that the procedure that controls which fuel assemblies may be stored in the cask pit racks will only be implemented once on each Unit, shortly after the installation of the racks. This approach allows fuel handling operations during the 14th and 15th refueling outages on each Unit to be performed using the existing storage racks with the procedures and controls currently in place.

The NRC staff finds the above response complete and acceptable since a specific procedure will be developed and approved in accordance with the DCPP's QA Program (FSAR Update, Section 17.5) for controlling the fuel that will be allowed to be placed in the cask pit storage rack and the licensee's planned use of the cask pit storage rack allows fuel handling operations during the 14th and 15th refueling outages on each Unit to be performed using the existing storage racks with the procedures and controls currently in place.

5.2.9 Quality Assurance and Inspection Programs

The NRC staff requested the licensee to discuss DCPP's plant-specific quality assurance and inspection programs to be implemented in order to preclude installation of an irregular or distorted cask pit rack/platform structure, and how DCPP confirms that the actual installed cask pit rack gap configurations are consistent with those gaps assumed in the cask pit rack/platform dynamic analysis and design.

The licensee responded that the procurement specification for the cask pit rack/platform structure contains quality verification requirements in accordance with the DCPP's QA Program (FSAR Update, Section 17.7, "Control of Purchased Material, Equipment, and Services"). The quality verification requirements include source inspection to assure that the structure is manufactured in accordance with the requirements and tolerances specified in Holtec's design and fabrication drawings. Receipt inspections will be performed in accordance with DCPP's QA Program requirements for quality-related components (FSAR Update Section 17.7 and 17.10, “Inspection”). Specified attributes will be verified for conformance with the design and fabrication drawings to assure that the structure has not been damaged during shipping and handling. All the inspections will be performed in accordance with DCPP’s QA Program as specified by FSAR Update, Section 17.10. The licensee further indicated that the cask pit rack/platform dynamic analysis considers two sets of gap dimensions, used simultaneously as input, in order to establish an acceptable range of gap dimensions while at the same time ensuring conservative results. To be more specific, the cask pit rack/platform dynamic model has the capability to differentiate between the fluid coupling gap and the impact gap. This allows the use of an upper bound gap for fluid coupling effects, thereby minimizing the resistance to rack motion, and a lower bound gap for impact tracking. This analytical approach has been used at DCPP to establish an acceptable range of gap dimensions in each direction.

The NRC staff finds the above response complete and acceptable since the procurement specification for the cask pit rack/platform structure contains quality verification requirements in accordance with the DCPP's QA Program and the quality verification requirements include source inspection to assure that the structure is manufactured in accordance with the requirements and tolerances specified in Holtec's design and fabrication drawings; receipt inspections will be performed in accordance with DCPP's QA Program requirements for quality-related components and specified attributes will be verified for conformance with the design and fabrication drawings to assure that the structure has not been damaged during shipping and handling, and a conservative analytical approach has been used at DCPP to establish an acceptable range of gap dimensions.

5.3 Conclusion

Based on the detailed evaluation discussed above, the NRC staff concludes that with respect to the seismic design adequacy and structural integrity evaluation presented in the LAR and its related documents, the licensee has provided adequate and acceptable engineering analysis results and technical justification in support of its request for a Cask Pit Spent Fuel Storage Rack.

6.0 TECHNICAL CONCLUSIONS

Based on the detailed evaluation discussed above, the NRC staff concludes that with respect to the criticality evaluation, SFP thermal-hydraulics, the handling of heavy loads, the seismic design adequacy and the structural integrity, the licensee’s proposed changes are acceptable.

7.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendments. The State official had no comments.

8.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration (69 FR 76481; published December 21, 2004). Although public comments were received on such finding, the enclosed resolution of the public comments does not change the Commission finding that the amendments involve no significant hazards consideration. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

9.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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THE NUCLEAR REGULATORY COMMISSION (NRC) STAFF'S RESPONSE TO

MOTHERS FOR PEACE COMMENTS

ON PACIFIC GAS & ELECTRIC (PG&E) LICENSE AMENDMENT REQUEST FOR

TEMPORARY CASK PIT SPENT FUEL STORAGE RACKS

By letter dated February 10, 2005 (Agencywide Documents Access Management System Accession Number ML050630380), Mothers For Peace (MFP) submitted comments on PG&E's license amendment request for a temporary cask pit spent fuel pool (SFP) storage rack at the Diablo Canyon Power Plant, Units 1 and 2 (DCPP). The NRC Staff response to MFP comments are provided below.

! MFP Comment: Safety Concerns

Despite denials by both PG&E and the NRC, MFP finds that the proposed amendment WOULD (1) involve a significant increase in the probability or consequences of an accident previously evaluated; and WOULD (2) create the possibility of a new or different kind of accident from any accident previously evaluated; and WOULD (3) involve a significant reduction in a margin of safety.

- NRC Staff Response:

MFP did not provide the basis of this comment. As published in the *Federal Register* on December 21, 2004 (69 FR 76481), the NRC staff proposed to determine that the amendment request involves no significant hazards consideration, and requested public comments on this proposed determination. Under the Commission's regulations in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), PG&E has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed PG&E's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has proposed to determine that the amendment request involves no significant hazards consideration.

! MFP Comment: Pool Fires

The increase of storage capacity by another 308 Spent Fuel Assemblies (SFA's) will inevitably lead to an increase of the risk to the citizens of California's Central Coast. DCPP's Spent Fuel Pools were originally designed, licensed, and built to contain no more than one and one third reactor core, i.e., 207 SFA's for each pool, and for no longer than five years. The re-racking which occurred in the 1980's brought the storage capacity to its present capability. The current storage situation is already stretching safety margins. It has substantially prolonged the stay of

SFA's under more intense radiation, and it has created the additional danger of pool fires, where next to none existed prior to the re-racking. The possibility of pool fires has only been recognized by the NRC since October 2000.

- NRC Staff Response:

The NRC staff has considered pool fires in granting the previous DCPP license amendments to re-rack the pools (License Amendment Nos. 22/21, dated October 20, 1987). The NRC staff has also considered potential DCPP spent fuel storage accidents whose consequences might exceed a fuel handling accident, i.e., "beyond design basis events," (NRC letter dated October 15, 1987, "Supplement to the Safety Evaluation and Environmental Assessment-Diablo Canyon Re-Rack"). Occurrences considered included a criticality accident and a zircalloy-clad fire caused by overheating due to loss of SFP cooling caused by a pool failure. The NRC staff has concluded that compliance with General Design Criteria 61 and 62 and adherence to approved industry codes and standards, as described in the PG&E amendment request, provide assurance that such events are of a very low probability. These conclusions are unaffected by the installation of a temporary cask pit spent fuel storage rack, which would be used only for storage of relatively old spent fuel.

! MFP Comment: Security

Moreover, the existing pools were not built with a 9/11 security situation in mind, and these pools lie outside the containment domes. Yet, they contain far more dispersible long-lived radioactivity than the domes which are structurally much stronger. Common sense indicates that any further increase of the radiation load in the pools will make the ramifications for the surrounding environment and population more severe should an act of malice lead to a substantial release of radiation.

- NRC Staff Response:

The NRC has issued guidance to the licensees on mitigating SFP security concerns. PG&E has committed to implement those measures. The staff is satisfied that implementation of these measures is adequate to protect public health and safety.

! MFP Comment: Seismic

With the recent seismic activity (2003 San Simeon and 2004 Parkfield quakes) in the region, such concerns must be taken into consideration. They are not, however, even discussed by either PG&E in its amendment request or by the NRC in its *Federal Register Notice*. Much has been said regarding the integrity of the pools and the racks in the event of an earthquake, but can the fuel assemblies withstand the stress of an earthquake? What effect have the g-forces during earthquakes on older, possibly embrittled SFA's, boraflex panels, racks and racking alignment, and other aging components of the pools? Can the panels or other aged components be replaced? At what cost? Have studies been done to evaluate these questions?

- NRC Staff Response:

PG&E has a continuing Long-Term Seismic Program (LTSP) which is committed to evaluate new seismic data for applicability to DCPP. PG&E has evaluated the 2003 San Simeon and 2004 Parkfield earthquakes and determined that these earthquakes did not require any modification to the existing seismic criteria that were used for evaluating installation of a temporary cask pit rack. Moreover, the Parkfield and San Simeon ground motions experienced were a small fraction of what the cask pit rack is designed to withstand.

PG&E's supplement letter dated June 23, 2005, responded to an NRC request for additional information regarding spent fuel racks operating experience following an earthquake. PG&E's response indicated that inspections of the plant facilities are performed by their operations, maintenance and engineering staff after an occurrence of an earthquake of meaningful magnitude. The results of post-earthquake inspections to date have not identified any observed structural damage to the SFP or racks. Additionally, the handling of new and spent fuel in the SFP subsequent to an earthquake has not identified any misalignment or deformation of the spent fuel racks that would indicate earthquake induced damage.

The existing boraflex panels are no longer credited for criticality prevention and therefore, would have no adverse affect.

! MFP Comment: Full Core Offload Capability (FCOC)

FCOC is apparently not mandatory but only a "current operational practice" [last sentence 3.1, page 2]. PG&E's argument that it must install the pit racks to avoid the loss of FCOC is not compelling, yet this is its sole given purpose for the amendment request. (See 3.1, page 2, "Purpose for Proposed Amendments".) Contrary to PG&E's claim, the proposed amendment will only delay the loss of FCOC, not eliminate it.

When cask loading begins, the SFAs will have to be removed from the pit to the FCOC area in order to remove the pit rack and begin the cask loading. Each cask will take about three working weeks to load (estimated 113 hours; Table 2.5 FEIR for independent spent fuel storage installation (ISFSI), page 2-30). That assumes that all goes smoothly. That means the total time of loading the casks for clearing the FCOC area in the pools would take a minimum of: 154 SFAs, which is equal to 565 working hours. With 40 working hours this is about 14 weeks. For a minimum of 28 weeks, FCOC will be lost if the temporary racks are installed. Even if PG&E or the NRC contests these numbers, provides double shifts, etc., it cannot be denied that the removal of racks from the cask pits and preparations for cask loading will take many working hours, and there will be many weeks when FCOC will be lost.

- NRC Staff Response:

The PG&E amendment request is to allow use of the temporary cask pit racks to maintain FCOC during Cycles 14-16, which is approximately 3-1/2 to 4 years. As discussed in the amendment request, the temporary cask pit racks must be removed prior to any spent fuel cask loading operations. It is correct that during the time when the temporary cask pit racks are being removed and prior to loading of casks, FCOC would not be available. However, as noted in the DCPP ISFSI FSAR Table 7.4-1, the time period to load a single cask is estimated to be approximately 120 hours. To restore FCOC, PG&E would have to load a maximum of five casks per unit. Using a conservative estimate of one week for removal of the cask pit rack and one week each for loading of five casks, each unit would lose FCOC for approximately 6 weeks.

This is a small fraction of the time during which each unit would lose FCOC as compared to 3-1/2 to 4 years without installation of the cask pit rack. In addition, maintaining FCOC is a prudent practice for operational reasons, but is not required for safety.

! MFP Comment: Movement of [spent fuel assemblies] SFAs

Any reshuffling of SFAs in the pools will add additional risks. The pit rack will not be loaded with only new SFAs from the reactor. As with the existing racks, a proper mixture of older and newer SFAs is required. This means that older SFAs need to be moved from the existing racks to the pit racks. PG&E also may have to re-shuffle SFAs in the existing racks in order to accomplish the correct mixture there. This means that there will be juggling of many SFAs in the pools. The older SFAs have been subjected to intense radiation for at least 10 years longer than was anticipated when the racks were placed into pools in the 1980s. As mentioned above in regards to a seismic event, can these older assemblies sustain the stress of movement? What impact does this prolonged radiation exposure have on the older SFAs? Have studies been done on the effects of prolonged radiation on: SFAs, the racking materials, the boraflex panels, other aging pool components? What about material embrittlement due to much longer than anticipated radiation exposure? Will the cladding break down?

Is it assured that the additional moving of older SFAs does not deteriorate their integrity for acceptance at a final repository? Can assurance be given that the older SFAs will not lead to problems of removing them again from the pit racks? Might the assemblies jam? How can cask loading begin if SFAs cannot be removed from the pit racks? Is there a method or equipment in place to remove a “stuck” SFA from the pit rack? None of these concerns or questions has been addressed by PG&E or the NRC.

- NRC Staff Response:

The addition of the temporary cask pit racks do not affect the length of time an individual spent SFA will be stored in the SFP. The length of time that SFAs will remain in the SFP is dependent upon when spent fuel casks are loaded. PG&E's request proposed temporary additional storage of 154 assemblies for each unit for cycles 14-16 that would only be used to maintain FCOC capability until fuel is offloaded to the dry storage casks.

The May 21, 1986 environmental assessment performed for the spent fuel pool re-racking bounds this concern since it was performed for the long-term storage of 1,324 SFAs beyond Cycles 14-16. In addition, the environmental assessment also referenced the “Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel” (NUREG-1575), which concluded that the environmental impact of long-term storage of SFAs in spent fuel pools is negligible.

The environmental assessment of long-term storage of spent fuel in SFPs was updated in NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants." NUREG-1437, Section 6.4.6.2, referenced the NRC’s Waste Confidence Rule (10 CFR 51.23) and its finding that spent fuel can be safely stored on-site for at least 30 years

beyond the licensed life of any plant. NUREG-1437, Section 6.4.6.2, also concluded that zircalloy-clad fuel assemblies do not appear to degrade as a result of long-term storage. In addition, accidental damage to spent fuel bundles through mishandling or component failures during emplacement or removal from pools has occurred very infrequently.

PG&E routinely moves SFAs in the SFP during refueling outages to facilitate SFP maintenance and spent fuel inspections. As discussed in PG&E's supplemental letter dated June 23, 2005, each cask pit rack will be fully loaded once with SFAs from the existing inventory of fuel in the SFPs and it is not PG&E's intent to utilize the temporary cask pit racks in the management of the spent fuel during subsequent refueling outages. It is expected that after the cask pit rack is loaded with spent fuel, the fuel will remain in place until dry cask loading operations necessitate their removal. This fuel movement does not significantly increase the normal frequency of fuel movement in the SFP, such as refueling outage and non-outage fuel shuffles.

PG&E has adequate procedures for the placement and removal of spent fuel in racks. Following installation of the temporary cask pit racks, testing using a dummy fuel assembly will be performed on each cell of the temporary cask pit racks to ensure that fuel can be successfully inserted and removed from the racks.

! MFP Comment: NRC Policy

MFP finds the NRC policy regarding the issuance of this proposed amendment request offensive. As stated in the *Federal Register*, December 21, 2004, Volume 69, Number 244:

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility.

This policy discourages public involvement, for it clearly states that any outside input - regardless of merit - is completely ineffectual. The NRC has obviously already made its determination.

- NRC Staff Response:

The NRC staff does not make a determination to issue a license amendment without a complete and satisfactory review of the amendment request. The NRC staff also considers public comments related to the amendment requests. Normally, it takes more than 60 days to issue a license amendment, and in most cases public comments are considered before issuing a license amendment. However, pursuant to 10 CFR 50.91, and as stated in the *Federal Register*, the Commission may issue the amendment prior to the expiration of the comment period should circumstances change such that failure to act in a timely way would result, for example, in derating or shutdown of the facility. In such a situation, the Commission will publish a notice of issuance providing for opportunity for a hearing and for public comment after issuance. The Commission expects that the need to take such action will occur very infrequently.

Note: Other MFP comments regarding delays and notifications are related to PG&E, and should be addressed to PG&E.

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