December 21, 2005

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

SUBJECT: DIABLO CANYON POWER PLANT UNIT NO. 1 AND UNIT NO. 2 - CORRECTION TO SAFETY EVALUATION ISSUED IN AMENDMENT NOS. 183 AND 185 (TAC NOS. MC5143 AND MC5144)

Dear Mr. Oatley:

In its letter dated November 21, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML052970272), the Nuclear Regulatory Commission (NRC) issued the 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 (DCPP), respectively. In those amendments, changes to the Technical Specifications (TSs) were issued in response to your application dated November 3, 2004, as supplemented by letters dated February 24, June 23, and September 30, 2005 (ADAMS Accession Nos. ML043130392, ML050630338, ML051800264, and ML052790486, respectively).

Pacific Gas and Electric Company has informed the NRC staff of several typographical errors in the safety evaluation. Based on our review of your comments, we agree that the comments are typographical and, therefore, we have revised pages 4, 6, 7, 8, and 9 of the safety evaluation.

Sincerely,

/RA/

Alan Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275
and 50-323

Enclosure: Revised Safety Evaluation Pages 4, 6, 7, 8, and 9

cc w/encl: See next page

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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 rack 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 rack 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 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 rack 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 EF at all times. For the 76, 96, and 193 assembly offloads, the local maximum water temperature is 188 EF, 194 EF, and 220 EF, respectively. The corresponding cladding temperature maxima are 225 EF, 231 EF, and 254 EF, 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 MWD/MT and 92 assemblies with 25,000 MWD/MT, and an emergency full core offload commencing 36 days into the cycle with 113 assemblies with burnup of 40,000 MWD/MT and 80 assemblies with 3,000 MWD/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 EF versus 150 EF for the current design basis. Likewise, the full core offload maximum bulk temperature is 157 EF versus 174 EF for the current value. The emergency full core offload maximum bulk temperature is 162 EF (not part of the current design basis) which is less than the existing analysis for the full core offload of 174 EF.

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 EF, which is below the water saturation temperature of 240 EF at the fuel depth.

3.3.3 Administrative Controls

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

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 rack 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 25,625 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,

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