April 20, 2001

Mr. Gregory M. Rueger
Senior Vice President, Generation and
 Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Nuclear Power Plant
P. O. Box 3
Avila Beach, CA 94177

SUBJECT: DIABLO CANYON NUCLEAR POWER PLANT (DCPP), UNIT 2 - ISSUANCE OF AMENDMENT RE: REVISION OF TECHNICAL SPECIFICATION (TS) SECTION 3.5.2 – ONE-TIME INCREASE IN CHARGING PUMP COMPLETION TIME DURING CYCLE 10 FROM 72 HOURS TO 7 DAYS (TAC NO. MA9132)

Dear Mr. Rueger:

The Commission has issued the enclosed Amendment No. 146 to Facility Operating License No. DPR-82 for the DCPP Unit 2. The amendment is in response to your application dated June 2, 2000, as supplemented by your letters dated December 15, 2000, and February 14, 2001.

The amendment revises TS Section 3.5.2, "ECCS - Operating," Action A to allow a one-time increase in the allowed outage time for centrifugal charging pump (CCP) 2-1 during Unit 2's Cycle 10 from 72 hours to 7 days. This change will allow for a potential on-line repair or a potential replacement of CCP 2-1. This pump is currently experiencing elevated vibration levels due to a structural resonance in the outboard bearing support structure and has been on an increased testing frequency since May 1996, due to high vibration.

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.

Sincerely,

/RA/

Girija S. Shukla, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-323

Enclosures: 1. Amendment No.   to DPR-82
 2. Safety Evaluation

cc w/encls: See next page

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PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 146
License No. DPR-82

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

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated June 2, 2000, as supplemented by letters dated December 15, 2000, and February 14, 2001, complies 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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 146, 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 within 30 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 20, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 146

FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NO. 50-323

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3.5-3	3.5-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 146 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT 2

DOCKET NO. 50-323

1.0 INTRODUCTION

By application dated June 2, 2000, Pacific Gas and Electric Company, (the licensee) requested changes to the technical specifications (TS) and to Facility Operating License No. DPR-82 for Diablo Canyon Power Plant (DCPP) Unit 2. The proposed change allows a one-time increase in the allowed completion time, or allowed outage time (AOT)[1], for centrifugal charging pump (CCP) 2-1 during Unit 2's Cycle 10, from 72 hours to 7 days. This change will allow for a potential on-line repair or a potential replacement of CCP 2-1. This pump is currently experiencing elevated vibration levels due to a structural resonance in the outboard bearing support structure and has been on an increased testing frequency since May 1996 due to high vibration. The original application was clarified during conference calls between the staff and the licensee on October 30, 2000, and January 10, 2001.

The supplemental letters dated December 15, 2000, and February 14, 2001, 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 published in the *Federal Register* on July 12, 2000 (65 FR 43051).

2.0 BACKGROUND

Since the mid-1980s, the NRC has been reviewing and granting improvements to TS that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements of July 22, 1993, the NRC stated that it...

> ...expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA (probabilistic safety assessment)[2] or risk survey and any available literature on risk insights and PSAs. Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical

---

[1] allowed completion time and allowed outage time are used interchangeably herein.

[2] PSA and PRA are used interchangeably herein.

> Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

The NRC reiterated this point when it issued the revision to 10 CFR Part 50.36, "Technical Specifications," in July 1995. In August 1995, the NRC adopted a final policy statement on the use of probabilistic safety assessment (PRA) methods in nuclear regulatory activities that improve safety decisionmaking and regulatory efficiency. The PRA policy statement included the following points:

1. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

2. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements.

3. PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

In April 1996, during Unit 2's seventh refueling outage (2R7), as part of a program to eliminate the potential of shaft cracking identified at other plants, CCP 2-1 was replaced by the licensee. Routine surveillance testing in May 1996 identified the outboard bearing vibration in the horizontal direction to be greater than the alert level of 0.325 inches/second. This resulted in the pump being placed on alert in accordance with ASME Section XI. The other bearing vibration measurements did not exceed the alert limits. A pump on alert is required to be tested every 42 days. Data collected from the required testing demonstrated that the pump bearing vibration has been between 0.174 inches/second and 0.510 inches/second. The licensee has stated that if the vibration action level of 0.700 inches/second were exceeded, the pump would be declared inoperable. However, the outboard bearing horizontal vibration level has appeared to stabilize at approximately 0.500 inches/second.

Multiple actions have been taken by the licensee to try to reduce the pump vibration. This included altering the structural resonance, increasing the torque on the hold down bolts, replacing the coupling, realigning the pump and gearbox couplings, and reducing the gap between the discharge head and casing. These actions have been unsuccessful in permanently reducing the vibration below the alert level.

Based on discussions with the pump manufacturer, and investigation of similar CCP vibration problems at another plant, the licensee has stated it expects that replacement of the CCP discharge head with an upgraded design will reduce the vibration below the alert level. The upgraded head provides additional support to the outboard bearing. An upgraded head was installed to resolve a similar problem at another plant and proved to be effective in reducing the vibration. If the CCP vibration remains below the action level of 0.700 inches/second, the licensee has indicated it plans to replace the pump discharge head during Unit 2 ‘s tenth refueling outage (2R10), which is expected to begin in May 2001. However, if the CCP vibration cannot be maintained below the action level between now and 2R10, corrective action will need to be taken prior to 2R10.

If repair to the CCP 2-1 pump needs to be performed prior to 2R10, the licensee has indicated that the most desirable option is to replace the pump discharge head. An alternative identified by the licensee to replacing the discharge head is to replace the entire pump. A spare charging pump with new pump elements is available in the DCPP warehouse for pump replacement. The licensee stated that replacement of the discharge head is expected to take up to 5 days, and the replacement of the entire CCP pump is expected to take just under 7 days. These durations assume the replacement activities are worked on a 24-hour schedule until completion.

3.0 EVALUATION

The staff evaluated the licensee’s proposed amendment to the TS using traditional engineering analysis, PRA methods, and a review of operating experience. The staff’s traditional analysis evaluated the capabilities of the plant to mitigate design basis events with one CCP inoperable. The staff then used insights derived from the use of PRA methods to determine the risk significance of the proposed changes. The results of these evaluations were used in combination by the staff to determine the safety impact of extending the completion times for one inoperable CCP.

3.1 Justification for Proposed Change to CCP 2-1 Completion Time from 72 Hours to 7 Days

With one CCP inoperable, the current completion time to restore operability is 72 hours, or be in hot standby within the next 6 hours and in hot shutdown within the following 12 hours. The licensee has stated that increasing the completion time from 72 hours to 7 days would provide a more reasonable completion time for the actions required to perform modifications to the pump or completely replace the pump. Increasing the completion time is consistent with recommendations of NUREG-1024, "Technical Specifications - Enhancing the Safety Impact." NUREG-1024 states:

> Allowable outage times that are too short will subject the plant to unnecessary trips, transients, and fatigue cycling. Outage times that are too short also may result in less thorough repair and post-repair testing before equipment is returned to service."

Maintaining the unit at power during the modification or replacement of CCP 2-1 provides the additional safety benefit of averting transitional risk associated with shutting the unit down.

3.2 Traditional Engineering Evaluation

The function of the emergency core cooling system (ECCS) is to provide core cooling and negative reactivity to ensure that the reactor core is protected after a design-basis accident.

The ECCS consists of three separate subsystems: (1) centrifugal charging (high head), (2) safety injection (SI) (intermediate head), and (3) residual heat removal (RHR) (low head). Each subsystem consists of two 100 percent capacity trains that are interconnected and redundant such that either train is capable of supplying 100 percent of the flow required to mitigate the accident consequences. Each ECCS train consists of a CCP, SI pump, RHR pump, piping, valves, and heat exchangers. The ECCS pumps are normally in a standby mode, although they may sometimes be used during normal operation. For example, the CCPs are used for normal charging. In Modes 1, 2, and 3, two independent (and redundant) ECCS trains are required to protect against a single failure affecting either train.

For high-head safety injection, both CCPs start automatically on an SI signal. Two CCPs, each with 100 percent flow capacity, are available to operate during the injection and recirculation phase following an accident to ensure that the safety injection function is fulfilled even assuming a single active failure. On receipt of an SI signal, CCP suction flow is automatically transferred from the volume control tank (VCT) to the refueling water storage tank (RWST). The normal charging path is automatically isolated on an SI signal and the ECCS injection path valves are automatically opened to provide flow to the reactor coolant system (RCS) cold legs. When the RWST water inventory is depleted to approximately 33 percent, the RHR pumps are automatically shut off, and the ECCS suction is manually transferred to the containment recirculation sump to place the system in the recirculation mode of operation. During the recirculation mode of operation, the RHR pumps provide suction to the CCPs and SI pumps. The recirculation mode of operation consists of a cold leg recirculation phase in which flow is supplied to the RCS cold legs and a hot leg recirculation phase in which flow is supplied to the RCS hot legs.

The ECCS is credited to provide core cooling and negative reactivity after any of the following accidents: loss-of-coolant accident (LOCA), non-isolable coolant leakage greater than the capability of the normal charging system; rod ejection accident; loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and steam generator tube rupture (SGTR).

The TS Limiting Condition for Operation (LCO) 3.5.2 requires two independent (and redundant) ECCS trains to ensure that sufficient ECCS flow is available to meet the design basis analysis assumptions for the above accidents, assuming a single failure affecting either train. TS 3.5.2, Action A.1 states that with one or more trains inoperable and at least 100 percent of the ECCS flow equivalent to a single operable ECCS train available, the inoperable components must be returned to operable status within 72 hours. The 72-hour completion time is based on a generic NRC reliability evaluation that has shown the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours. During the 72-hour completion time, 100 percent of the ECCS flow required to mitigate accidents can be provided absent the occurrence of particular single failures. A single failure is not required to be postulated during the completion time.

A completion time of 72 hours is usually sufficient to perform necessary preventive or corrective maintenance required on the CCPs. However, replacement of the CCP 2-1 discharge head or replacement of CCP 2-1 is expected to require up to 7 days. Since the CCP 2-1 discharge head or pump replacement is expected to exceed one half of the TS completion time, the replacement activities will be planned by the licensee to be worked on a 24-hour schedule until completion per DCPP Administrative Procedure AD7.1D4, "On-line Maintenance Scheduling." During the 7-day period, 100 percent of the ECCS flow required to mitigate accidents can be provided if no additional single failure occurs. With no single failure, there are no situations in which entry into a 7-day completion time, due to an inoperable CCP 2-1, would result in failure to meet an intended safety function. In addition, the licensee has indicated that it will institute compensatory actions during the replacement activities in order to minimize the increase in risk during the 7-day period when CCP 2-1 is inoperable.

3.3 Probabilistic Risk Assessment Evaluation

The staff used a three-tiered approach to evaluate the risk associated with the proposed TS changes. The first tier evaluated the PRA model and the impact of the completion time extension on plant operational risk. The second tier addressed the need to preclude potentially high risk configurations, should additional equipment outages occur during the time when CCP 2-1 is out of service. The third tier evaluated the licensee's configuration risk management program to ensure that the applicable plant configuration will be appropriately assessed from a risk perspective before entering into or during the proposed completion times. Each tier and the associated findings are discussed below.

3.3.1 Tier 1 Evaluation

The licensee used traditional PRA methodology to evaluate the requested AOT extension for CCP 2-1. The Tier 1 NRC staff review of the licensee's PRA involved three aspects: (1) evaluation of the PRA model and application to the proposed AOT extension, (2) evaluation of PRA results and insights stemming from the application, and (3) discussion of the quality of the PRA.

(1) Evaluation of PRA Model and Application to the AOT Extension

The staff's review focused on the capability of the licensee's PRA model to analyze the risk stemming from the proposed AOT changes for CCP 2-1, and did not involve an in-depth review of the licensee's PRA. The NRC previously performed a review of the licensee's individual plant examination (IPE) submittal. The current review was based on the staff's initial screening process where the staff examined the licensee's internal events PRA results and recent operational experience regarding availability and reliability of centrifugal charging pumps. The staff concludes that the licensee's PRA results are reasonable, and the scope and depth of the PRA analysis support such a finding. Discussions with the licensee regarding recent data for CCP reliability and availability did not indicate any adverse trends.

The licensee performed a PRA to determine the effect of extending the CCP 2-1 completion time from 72 hours to 7 days. The DCPP PRA is a full-scope, level 2 PRA that evaluates the frequency of experiencing reactor and plant damage as a result of both internal and external initiating events. While the PRA was performed for DCPP Unit I only, the staff has previously concluded that it is also applicable to DCPP Unit 2 considering the substantial similarities between the two units. The NRC review and acceptance of the original PRA evaluation, DCPRA-1988, is summarized in the safety evaluation report, Supplement No. 34 (NUREG-0675, June 1991). Much of the review of DCPRA-1988 was performed by

Brookhaven National Laboratory (BNL), and the review is documented in NUREG/CR-5726, published August 1994. As part of the licensee's living PRA program, the DCPP PRA was updated in 1990, 1991, 1993, 1995, and 1997, and is continuing to undergo updates to internal and seismic models to assure that the DCPP PRA reflects current plant design and operation.

(2) Evaluation of PRA Results and Insights

The licensee indicated that if it were assumed that a CCP were removed from service for the full 7 day completion time, the additional core damage frequency (CDF) added from seismic and internal event initiators would be 2.3×10^{-7} per year. The licensee notes that Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," identifies an increase in core damage frequency as "very small" if it were 1×10^{-6} per year or less. Assuming this was the largest core damage frequency increase the NRC would allow for the allowed outage time extension of CCP 2-1, the licensee calculated that the existing combination of power seismic and internal PRA model would support up to a 604 hour (25 days) allowed outage time for CCP 2-1. The proposed increase in CCP 2-1 completion time uses less than one third of the guideline's allotted amount. The licensee did not submit an estimate for the incremental conditional core damage probability (ICCDP) as called for in Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications." Because the licensee is asking for a one-time AOT extension, the determination of the ICCDP is directly calculable from the licensee's submittal. Based on the core damage frequency increase identified by the licensee, the staff estimates the ICCDP to be less than 5×10^{-9}, which is much less than the 5.0×10^{-7} guideline in RG 1.177 used to identify a "small" risk impact for a single AOT change.

The licensee calculated the increase in the large early release frequency (LERF) to be 2.3×10^{-9} per year, which would be identified as a very small increase in RG 1.174. The licensee did not submit an estimate for the incremental conditional large early release probability (ICLERP) as called for in RG 1.177. Based on the large early release frequency increase identified by the licensee, the staff estimates the ICLERP to be less than the 5.0×10^{-8} guideline in RG 1.177 used to identify a "small" risk impact for a single AOT change.

The total CDF (Internal + Seismic) is 1.0×10^{-4} per year. The effect on fire and flood CDF due to an extended AOT for the CCP 2-1 was evaluated qualitatively by the licensee. The licensee found that fire events were not a significant contributor to change in CDF for the charging pump, although the change in fire-induced CDF is not directly calculable on a component level. Similar results were found for internal floods. Per NRC guidelines stated in RG 1.174 and RG 1.177, having the CCP out of service for 7 days is a low risk significant configuration.

(3) Quality of the Diablo Canyon PRA

The NRC evaluation of the original Diablo Canyon PRA submittal, performed primarily by BNL and completed by BNL in 1993, found the PRA was beyond the state-of-the-art of the time. The licensee indicated in its December 15, 2000, submittal that a peer review by the Westinghouse Owners Group (WOG) in May 2000 stated that the licensee has maintained a high quality PRA that is appropriate for risk-informed submittals. The licensee addressed recommendations by the WOG peer reviewers for an area (i.e., human reliability analysis) found to be outdated. The DCPRA model uses industry-wide and plant-specific data in quantifying its core damage frequency estimates.

The licensee states it applied design-basis configuration control processes and procedures to PRA calculations. PG&E Nuclear Power Generation Procedure CF3.ID15 is the governing procedure for PRA calculations. Within this procedure are requirements to conduct independent verification of calculations. The application of this process was reviewed by the WOG peer review team in May 2000. The review found the independent verification process met the standards for a risk-informed submittal.

In a letter dated June 27, 1994, the licensee submitted to NRC the individual plant examination of external events (IPEEE) for the DCPP. The IPEEE submittal provides insights from an extensive evaluation of seismic events. The staff reviewed the DCPP IPEEE and found it capable of detecting vulnerabilities to external event severe accidents.

The staff finds that a small incremental increase in core damage frequency estimated for the change in AOT from 3 to 7 days is consistent with the credit taken for the system in the PRA modeling, and that the extensive licensee review and updating of the PRA models provide reasonable assurance that the models appropriately reflect the equipment and procedural characteristics at the plant.

3.3.2 <u>Tier 2 Evaluation</u>

The second tier addressed the need to preclude potentially high risk configurations by identifying the need for any additional constraints or compensatory actions that, if implemented, would avoid or reduce the probability of a risk-significant configuration during the time when one CCP is out of service. The licensee identified and committed to take the following actions once the license amendment is granted for the CCP 2-1 repair:

- Before beginning work on CCP 2-1, the risk will be assessed per plant procedures as required by 10 CFR 50.65(a)(4) of the Maintenance Rule.

- It will be verified that CCP 2-2 and the system alignment is operable and available to provide injection flow to the reactor coolant system in the event of a safety injection signal.

- No elective maintenance or surveillance testing will be performed that disables the ECCS equipment (except CCP 2-1). This will maximize the availability of ECCS flow to provide the safety injection function.

- The emergency diesel generators (EDGs) will be verified to be operable. Additionally, no elective maintenance or testing will be performed on the EDGs, the 230kV or 500kV systems. This will maximize the availability of onsite AC power should offsite power be lost and ensure that power is available to all ECCS equipment.

- The risk of performing elective maintenance or surveillance testing on other risk significant systems, structures, and components will be assessed and managed for the current plant state per plant procedures.

- Very high risk plant evolutions as described in plant risk assessment procedures will be avoided.

- Elective load changes will not be performed.

These compensatory actions are being taken by the licensee to help assure that the CCP 2-2 pump and other ECCS equipment are operable and capable of being powered, and to minimize the possibility that the ECCS equipment will be required. The actions help decrease the possibility that risk significant equipment is lost and minimize the overall plant risk during the CCP 2-1 outage. With these compensatory actions in place, the licensee has stated that 100 percent of the ECCS flow required to mitigate accidents can be provided if no single failure were assumed.

3.3.3 Tier 3 Evaluation

Tier 3 is the development of a proceduralized program to ensure that the risk impact of out-of-service equipment is appropriately evaluated prior to performing a maintenance activity. A viable program would be one that is able to uncover risk-significant plant equipment outage configurations in a timely manner during normal plant operation. The DCPP plant procedures include requirements to perform a risk calculation and safety function degradation evaluation to assess the effect of components that are taken out of service. The licensee indicated this will be done as part of the preparation for the contingent activities. Additional information on identifying contingencies is supplied in the Tier 2 evaluation above. This AOT extension is only a one-time extension during Cycle 10.

Therefore, in conclusion, the staff finds that the AOT for CCP 2-1 may be extended to 7 days on a one-time basis with a negligible effect on risk.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the 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 amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (65 *FR* 43051). Accordingly, the amendment meets 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 amendment.

6.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Glenn Kelly

Date: April 20, 2001