June 26, 2007

Mr. John S. Keenan
Senior Vice President and Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 770000
San Francisco, CA 94177-0001

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TECHNICAL SPECIFICATION 5.5.16, “CONTAINMENT LEAKAGE RATE TESTING PROGRAM,” FOR CONSISTENCY WITH 10 CFR 50.55a(g)(40) (TAC NOS. MD3977 AND MD3978)

Dear Mr. Keenan:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 197 to Facility Operating License No. DPR-80 and Amendment No. 198 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 December 29, 2006.

The amendments revise TS 5.5.16, “Containment Leakage Rate Testing Program” to comply with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC.

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/

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

Enclosures:
1. Amendment No. 197 to DPR-80
2. Amendment No. 198 to DPR-82
3. Safety Evaluation

cc w/encls: See next page

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ADAMS Accession Nos.: PKG: ML071370728, Amdt/License ML071370731, TS Pages ML071370733

OFFICE	DORL/LPL4/PM	DORL/LPL4/LA	DIRS/ITSB/BC	DSS/SVCB/BC	OGC	DORL/LPL4/BC
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DATE	6/13/07	6/12/07	4/25/07	5/30/07	6/22/07	6/25/07

OFFICIAL RECORD COPY

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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. 197
License No. DPR-80

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

   A. The application for amendment by Pacific Gas and Electric Company (the licensee), dated December 29, 2006, 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 and paragraph 2.C.(2) of Facility Operating License No. DPR-80 as indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License and Technical Specifications

Date of Issuance: June 26, 2007

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 198
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 December 29, 2006, 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 and paragraph 2.C.(2) of Facility Operating License No. DPR-82 as indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License and Technical Specifications

Date of Issuance: June 26, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 197

TO FACILITY OPERATING LICENSE NO. DPR-80 AND

AMENDMENT NO. 198 TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NOS. 50-275 AND 50-323

Replace the following pages of the Facility Operating License Nos. DPR-80 and DPR-82, and 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.

Operating License DPR-80

REMOVE	INSERT
-3-	-3-

Operating License DPR-82

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
5.0-24	5.0-24
5.0-24a	5.0-24a

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 197, are hereby incorporated in the license. Pacific Gas & 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) Initial Test Program

The Pacific Gas and Electric Company shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Pacific Gas and Electric Company's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of PG&E's Final Safety Analysis Report as amended as being essential;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incor-porated below:

(1) Maximum Power Level

The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.

(2) Technical Specifications (SSER 32, Section 8)* and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 198, are hereby incorporated in the license. Pacific Gas & 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) Initial Test Program (SSER 31, Section 4.4.1)

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

*The parenthetical notation following the title of many license conditions denotes the Section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 197 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 198 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 December 29, 2006 (Agencywide Documents Access and Management System Accession No. ML070160261), Pacific Gas and Electric Company (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License Nos. DPR-80 and DPR-82) for the Diablo Canyon Power Plant, Units 1 and 2 (DCPP).

The proposed amendments would revise Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program," to comply with the requirements of paragraph 50.55a(g)(4) of Title 10 of the *Code of Federal Regulations* (10 CFR) for components classified as Code Class CC consistent with TS Task Force (TSTF) Traveler number TSTF-343, "Containment Structural Integrity." This regulation requires licensees to update their containment inservice inspection requirements in accordance with subsections IWE and IWL of Section XI, Division I, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR 50.55a (b)(2)(ix).

2.0 REGULATORY EVALUATION

The regulatory requirements and the guidance upon which the staff based its review of the requested change are based on 10 CFR Part 50:

(1) Appendix A, General Design Criterion (GDC) 16 as it relates to the containment and associated systems establishing a leak-tight barrier against the uncontrolled release of radioactivity to the environment and assuring that the containment design conditions important to safety are not exceeded for as long as the postulated accident requires,

(2) Appendix J, Option B as it relates to the general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity,

(3) Section 50.55a as it relates to the inservice inspection (ISI) requirements of ASME Code, Class CC components.

When implementing Option B of Appendix J to 10 CFR Part 50, "Performance-Based Leakage-Test Requirements," TS 5.5.16 states that the licensee's program shall be in accordance with the guidelines in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." Item 3 of RG 1.163, Section C, discusses visual examinations of accessible interior and exterior surfaces of the containment system. Specifically, item 3 states, "examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years. . ."

In addition, Section 50.55a of 10 CFR Part 50 requires licensees to perform their containment ISI requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Code. Paragraph 50.55a(g)(4) of 10 CFR requires licensees to update their containment ISI requirements in accordance with subsections IWE and IWL of Section XI, Division I, of the ASME Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(vii) and 10 CFR 50.55a(b)(2)(ix).

The proposed change to the TSs revises the containment leakage rate testing program for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as ASME Code CC. Specifically, the licensee proposes to revise TS Section 5.5.16, "Containment Leakage Rate Testing Program," to allow the performance of visual examinations of the containment pursuant to ASME Code, Section XI, Subsections IWE and IWL, in lieu of the visual examinations performed pursuant to RG 1.163.

3.0 TECHNICAL EVALUATION

DCPP, Units 1 and 2, are Westinghouse-type pressurized-water reactors (PWRs) having four primary loops connected in-parallel to the reactor vessel. According to the Updated Final Safety Analysis Report, the containment structures for Units 1 and 2 are essentially identical, as mirror images. The containment system consists primarily of a steel-lined, reinforced concrete building of cylindrical shape with a dome roof that completely encloses the reactor and the reactor coolant system. It is designed to prevent significant release of radioactive materials to the environment that could result from accidents inside the containment. The containment floor is a concrete pad. The inside of the dome, cylinder, and the base pad is lined with welded steel plate, which forms a leak-tight membrane. The containment vessel is penetrated by access openings, process piping, and electrical penetrations. GDC 16 requires that the containment establish a leak-tight barrier against the uncontrolled release of radioactivity to the environment. The leak-tightness of the penetrations is verified through 10 CFR Part 50, Appendix J, Type B and Type C local leak rate tests, and the overall leak tightness of the containment vessel is verified through a 10 CFR Part 50, Appendix J, Type A integrated leak rate test.

3.1 TS 5.5.16

The licensee has proposed to change TS 5.5.16.a by replacing the last sentence of the paragraph with the following exceptions, which modify compliance with RG 1.163:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

3. The ten-year interval between performance of the integrated leakage rate (Type A) test, beginning May 4, 1994, for Unit 1 and April 30, 1993, for Unit 2, has been extended to 15 years.

The licensee has stated that the changes to TS 5.5.16.a are consistent with the changes proposed in TSTF-343, Revision 1. As proposed, the amendment would not change the current requirements that the leakage rate testing program shall meet the requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. However, the current requirement that the program shall be in accordance with RG 1.163 shall be modified by adding two exceptions to the RG.

In addition, as a result of the change, TS Sections 5.5.16.d, e, and f have been moved to page 5.0-24a. The U.S. Nuclear Regulatory Commission (NRC) staff has determined that this change is editorial in nature and acceptable.

3.2 EVALUATION

TSTF-343 proposed to revise the Pre-Stressed Containment Surveillance Program and Containment Leakage Rate Program in the Standard Technical Specifications (STS) to reflect changes made to 10 CFR 50.55a in 1996. The final rule became effective on September 9, 1996, and required licensees to implement Subsections IWE and IWL of Section XI, Division I, of the ASME Code. Consistent with the 1996 changes to 10 CFR 50.55a, TSTF-343 proposed that the STS be modified to require concrete surfaces of the containment to be visually examined in accordance with ASME Code, Section XI, Subsection IWL, and the liner plate inside containment to be visually examined in accordance with Subsection IWE. In addition, under the changes proposed in TSTF-343 the licensee will continue to meet the other of guidelines in RG 1.163 and Nuclear Energy Institute (NEI) 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J.” The NRC staff concluded that the revision proposed in TSTF-343 to adopt the changes in 10 CFR 50.55a was acceptable, since the requirements of 10 CFR 50.55a adequately provided for the testing of containment leakage and containment tendons.

The proposed change would require the containment visual examination be performed pursuant to ASME Code, Section XI, Subsections IWL and IWE rather than the visual inspection guidelines in RG 1.163. The containment leak rate testing Subsection IWE requires the licensee to perform the general visual examinations of the containment liner three times in a 10-year interval consistent with the current TSs. Subsection IWL requires the licensee to perform general visual examinations of the accessible concrete surfaces once every five years alternating between units. Therefore, the rule change for a dual unit site, resulted in the licensee performing two less visual inspections per unit of the containment concrete surfaces during the 10-year interval. However, the requirements for inspection in Subsections IWE and IWL of the ASME Code, Section XI, are more rigorous than those currently performed. For the inspection of Class MC and metallic liners of Class CC components, the ASME Code requires that the examiner be knowledgeable in the requirements for design, inservice inspection, and testing of the components and that examinations be performed by an examiner with visual acuity sufficient to detect evidence of degradation.

For the inspection of structural concrete the ASME Code requires that the inspections be performed by a Responsible Engineer. The Responsible Engineer ensures that a comprehensive visual examination of the concrete is performed in accordance with ASME Code requirements. This would include the development of plans and procedures for the examination of the concrete surfaces with instruction and training for the examiner. The Responsible Engineer is also responsible for the evaluation of the results and any repair/replacement activities as result of the examination. In addition, both Subsections IWL and IWE require that the visual examinations be reviewed by an inspector employed by a State or municipality of the U.S. or an inspector regularly employed by an insurance company authorized to write boiler and pressure vessel insurance.

Based on the above, the NRC staff concluded that the licensee is adopting a more rigorous containment visual inspection with a required third-party review that compensates for fewer inspections and therefore, the proposed TS change is acceptable. TS 5.5.16.a will continue to require the containment leakage rate testing program to be in accordance with RG 1.163 except for the visual inspections, which will now be performed in accordance with the ASME Code.

The licensee proposes to revise TS Section 5.5.16, “Containment Leakage Rate Testing Program”, to allow the performance of visual examinations of the containment pursuant to ASME Code, Section XI, Subsections IWE and IWL in lieu of the visual examinations required by RG 1.163. The NRC staff determined that the proposed change: (1) is consistent with the requirements of 10 CFR Part 50, Section 50.55a, (2) meets requirements of 10 CFR Part 50 Appendix J, Option B, because the visual examination of containment as per ASME Code, Section XI, Subsections IWE and IWL, are more rigorous than the examinations recommended in regulatory position C.3 of RG 1.163, and (3) meets the requirements of 10 CFR 50, Appendix A, GDC 16, because the proposed change in the visual examination requirements continues to assure the leak-tight integrity of the containment. Therefore, the licensee’s proposed changes in this license amendment request are more restrictive in nature, reflect the current configuration of the plant, and increase the TS requirements. The NRC staff finds the proposed changes are technically justified, comply with 10 CFR 50.36 and 10 CFR 50.55a, and are consistent with the STS. On this basis, the NRC staff concludes that the proposed changes to the TS of the DCPP are acceptable. In addition, as a result of the change, TS Sections 5.5.16.d, e, and f have been moved to page 5.0-24a. This NRC staff has determined that this change is editorial in nature and acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 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 and there has been no public comment on such finding (72 FR 6785, published on February 13, 2007). 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.

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

Principal Contributor: P. Hearn
A. Sallman
A. Wang

Date: June 26, 2007