UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 15, 2009

Mr. John T. Conway
Senior Vice President and
Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 3, Mail Code 104/6/601
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION TO TECHNICAL SPECIFICATION 5.5.16, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM," (TAC NOS. MD8042 AND MD8043)

Dear Mr. Conway:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 203 to Facility Operating License No. DPR-80 and Amendment No. 204 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant (DCPP), Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 1, 2008, as supplemented by letter dated August 20, 2008.

The amendments revise TS 5.5.16.b, "Containment Leakage Rate Testing Program," to specify a lower peak calculated containment internal pressure following a large-break loss-of-coolant accident and the containment design pressure at the DCPP, Units 1 and 2. By letter dated August 20, 2008, the licensee withdrew its request to use the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002, "Containment System Leakage Testing," in lieu of the 1994 Edition.

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,

Alan Wang

Alan B. 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. 203 to DPR-80
2. Amendment No. 204 to DPR-82
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 203
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 February 1, 2008, as supplemented by letter dated August 20, 2008, 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-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. 203, 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. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. DPR-80
and Technical Specifications

Date of Issuance: January 15, 2009

**UNITED STATES**
**NUCLEAR REGULATORY COMMISSION**
**WASHINGTON, D.C. 20555-0001**

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 204
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 February 1, 2008, as supplemented by letter dated August 20, 2008, 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) 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. 204, 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. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. DPR-82
and Technical Specifications

Date of Issuance: January 15, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 203

TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 204 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.

Facility Operating License Nos. DPR-80 and DPR-82

| REMOVE | INSERT |
|---|---|
| -3- | -3- |

Technical Specifications

| REMOVE | INSERT |
|---|---|
| 5.0-16 | 5.0-16 |

(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) <u>Maximum Power Level</u>

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) <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. 203, 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) <u>Initial Test Program</u>

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 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 (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. 204, 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.

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

   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 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 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.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 43.5 psig. The containment design pressure is 47 psig.

c. The maximum allowable containment leakage rate, $L_a$, at $P_a$, shall be 0.10% of containment air weight per day.

(continued)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 203 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 204 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 February 1, 2008, as supplemented by letter dated August 20, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML080390454 and ML082410418, respectively), Pacific Gas and Electric Company (PG&E, the licensee) requested changes to the Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP), Units 1 and 2. The supplemental letter dated August 20, 2008, provided additional information that clarified the application, 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 March 25, 2008 (73 FR 15787).

The proposed amendments will revise TS 5.5.16.b, "Containment Leakage Rate Testing Program," to specify a lower peak calculated containment internal pressure following a large-break loss-of-coolant accident (LOCA) and the containment design pressure at the DCPP, Units 1 and 2. By letter dated August 20, 2008, the licensee withdrew its request to use the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002, "Containment System Leakage Testing," in lieu of the 1994 Edition.

## 2.0 REGULATORY EVALUATION

Section 50.54(o) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J, "Leakage Rate Testing of Containment of Water Cooled Nuclear Power Plants," to 10 CFR Part 50. Appendix J specifies containment leakage testing requirements, including the types of tests required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing, and reporting requirements for each type of test.

10 CFR 50 Appendix A, General Design Criterion (GDC) 50, Containment design basis, requires that the containment, shall be designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated conditions from any LOCA.

10 CFR 50 Appendix A, GDC 16, Containment design, requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment, and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

NUREG-1431, "Standard Technical Specifications [STS], Westinghouse Plants Specifications," Volume 1, Revision 3, dated June 2004, is the latest approved version of STS available for use.

## 3.0 TECHNICAL EVALUATION

The current DCPP TS 5.5.16.b states that the peak calculated containment internal pressure for the design-basis LOCA, $P_a$, is 47 pounds per square inch gauge (psig), which is the containment design pressure value. The current Final Safety Analysis Report Update (FSARU) analysis of record for the peak containment internal pressure following the design-basis LOCA with the original Model 51 steam generators (SGs) is contained in Appendix 6.2C of the DCPP FSARU. Using the COCO digital computer code, the peak containment internal pressure was calculated to be 41.53 psig. The use of the containment design pressure of 47 psig as $P_a$ in TS 5.5.16.b is conservative since it results in containment leak-rate testing being performed at a pressure well above the current peak calculated containment internal pressure of 41.53 psig following the design-basis LOCA. In accordance with Section 3.2.11 of ANSI/ANS 56.8-1994, compliance with the current TS for an integrated leak-rate test (ILRT) requires that the ILRT test pressure shall not be less than 0.96 $P_a$ (47 psig) or 45.12 psig nor exceed P-design (47 psig).

Current TS 5.5.16.b states:

> The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 47 psig.

The proposed change to TS 5.5.16.b would state:

> The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 43.5 psig. The containment design pressure is 47 psig.

The licensee performed a new containment integrity analysis as part of the DCPP replacement SG program. The reanalysis was performed to reevaluate the bounding peak pressure and temperature of design-basis LOCA inside containment and to demonstrate the ability of the containment heat removal systems to mitigate accidents. Calculation of the containment response following a postulated LOCA was analyzed by use of the digital computer code GOTHIC Version 7.2. This code version was used to take advantage of the diffusion layer model heat transfer option.

The GOTHIC modeling for DCPP is consistent with the NRC-approved Kewaunee Power Station evaluation model. Kewaunee and DCPP both have large dry containment designs with similar active heat removal capabilities. The heat transfer option was approved by the Nuclear Regulatory Commission (NRC) for use in the Kewaunee analyses with the condition that the effect of mist be excluded from what was earlier termed as the mist diffusion layer model. The GOTHIC containment modeling for DCPP has followed the conditions of acceptance placed on the Kewaunee evaluation model.

The NRC staff reviewed the input parameters and assumptions used for the DCPP GOTHIC model for containment integrity analysis for a large-break LOCA. The NRC staff found the input parameters and assumptions to be conservative and consistent with the input parameters and assumptions used for the NRC-approved Kewaunee evaluation model. The differences in GOTHIC code versions are documented in Appendix A of the GOTHIC User Manual Release Notes. Version 7.2 of GOTHIC is used consistently with the restrictions identified in the NRC-approved evaluation model and none of the user-controlled enhancements added to Version 7.2 were implemented in the DCPP containment model.

According to the licensee's analysis, the actual peak calculated containment internal pressure is 41.2 and 41.4 psig, for DCPP, Units 1 and 2, respectively. The licensee has proposed to use a conservative value of 43.5 psig as the peak calculated containment internal pressure in TS 5.5.16.b for DCPP. This ensures that the value specified in TS 5.5.16.b contains adequate margin to the FSARU containment internal pressure analysis values of 41.2 and 41.4 psig for the DCPP, Units 1 and 2, respectively. The containments for DCPP, Units 1 and 2, are designed and constructed to withstand a maximum internal pressure of 47 psig, which is specified as the containment design pressure in the proposed TS 5.5.16.b. Therefore, the licensee proposes to change TS 5.5.16.b, to specify the peak calculated containment internal pressure, $P_a$, as 43.5 psig following a LOCA and to specify the containment design pressure as 47 psig. The revised TS 5.5.16.b will allow Type A, B, and C leak-rate tests at the DCPP to be performed at a lower pressure and provide a wider acceptable pressure test range for containment ILRTs without exceeding the containment design pressure.

As noted above, the current TS is conservative as it uses the containment maximum design pressure of 47 psig for $P_a$, the peak calculated containment internal pressure for the design basis loss-of-coolant accident. As allowed by NUREG-1431, Revision 3, STS 5.5.16.b, Option B, the licensee has proposed that the TS 5.5.16.b $P_a$ value (43.5 psig) be based on the FSARU containment internal pressure analysis calculated pressure with margin included. DCPP converted to the NUREG-1431, STS and did not adopt STS 5.5.16.b, Option B change at the time of conversion. This change brings the DCPP TS into conformance with NUREG-1431, Revision 3, STS 5.5.16.b, Option B. The value of 43.5 psig is bounding for both units. The containment design pressure has not changed. Based on the above, the NRC staff concludes that the licensee's proposed change is 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 published in the *Federal Register* on March 25, 2008 (73 FR 15787). 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: B. Lee

Date: January 15, 2009

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 B. 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. 203 to DPR-80
2. Amendment No. 204 to DPR-82
3. Safety Evaluation

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