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

December 31, 2008

Mr. John Conway
Senior Vice President - Station
Generation 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: INCREASE IN THE COMPLETION TIMES FOR REQUIRED ACTIONS RELATED TO TECHNICAL SPECIFICATIONS 3.5.2, REGARDING THE EMERGENCY CORE COOLING SYSTEM, AND 3.6.6, REGARDING THE CONTAINMENT SPRAY AND COOLING SYSTEMS (TAC NOS. MD7512 AND MD7513)

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. 202 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated December 17, 2007, as supplemented by letters dated October 2, and November 18, 2008.

The amendments increase the completion times for required actions related to Technical Specifications 3.5.2, regarding the Emergency Core Cooling System, and 3.6.6, regarding the Containment Spray and Cooling Systems from 72 hours to 14 days.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Alan Wang

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. 202 to DPR-80
2. Amendment No. 203 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. 202
License No. DPR-80

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

   A. The application for amendment by Pacific Gas and Electric Company (the licensee), dated December 17, 2007, as supplemented by letters dated October 2, and November 18, 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. 202, 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 180 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: December 31, 2008

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. 203
License No. DPR-82

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

   A. The application for amendment by Pacific Gas and Electric Company (the licensee), dated December 17, 2007, as supplemented by letters dated October 2, and November 18, 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. 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 180 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: December 31, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 202

TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 203 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
DPR-80 License Page 3	DPR-80 License Page 3
DPR-82 License Page 3	DPR-82 License Page 3

Technical Specifications

REMOVE	INSERT
3.5-3	3.5-3
3.6-13	3.6-13

(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. 202, 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 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. 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) 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.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

-----NOTE-----
In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valve(s) for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status	72 hours -----NOTE----- The Required Action A.1 Completion Time is to be used for planned maintenance or inspections. The Completion Times of Required Actions A.2.1, A.2.2, and A.2.3 are for unplanned corrective maintenance or inspections. -----
	OR	
	A.2.1 Verify only one subsystem in one ECCS train is inoperable	72 hours
	AND	
	A.2.2 Determine there is no common cause failure in the same subsystem in the OPERABLE ECCS train	72 hours
	AND	
	A.2.3 Restore train to OPERABLE status	14 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 The containment fan cooling unit (CFCU) system and two containment spray trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status. OR	72 hours AND 10 days from discovery of failure to meet the LCO ----------NOTE----------- For planned maintenance or inspections, the Completion Time is 72 hours. The Completion Times of Required Action A.2 are for unplanned corrective maintenance or inspections. ------------------------------
	A.2 Restore containment spray train to OPERABLE status	14 days <u>AND</u> 14 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 Be in MODE 5.	84 hours
C. One required CFCU system inoperable such that a minimum of two CFCUs remain OPERABLE.	C.1 Restore required CFCU system to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 203 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 17, 2007, as supplemented by letters dated October 2, and November 18, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML073650012, ML082890537, and ML083370199, respectively), Pacific Gas and Electric Company (PG&E, the licensee) requested changes to the Technical Specifications (TSs), Appendix A to Facility Operating License Nos. DPR-80 and DPR-82, respectively, to the U.S. Nuclear Regulatory Commission (NRC) for the Diablo Canyon Power Plant, Units 1 and 2 (DCPP). The supplements dated October 2, and November 18, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff original proposed no significant hazards consideration determination as published in the *Federal Register* on January 29, 2008 (73 FR 5227).

The proposed changes would extend the completion times (CTs) applicable to a single train inoperability in either the Emergency Core Cooling System (ECCS) or the Containment Spray (CS) System. Specifically, TS 3.5.2, "ECCS - Operating," would be modified to include a new set of required actions, A.2.1 through A.2.3, which would permit 14 days to restore a single inoperable ECCS subsystem to operable status, and TS 3.6.6, "Containment Spray and Cooling Systems," would be modified to include new required action A.2, which would permit 14 days to restore a single inoperable ECCS subsystem to operable status. A new note is added to state that the proposed TS changes are applicable only for unplanned repair activities, and the TS retain the current 72 hour CT for planned maintenance. In addition, the second CT associated with contiguous application of the actions of the TS 3.6.6 is proposed to be 14 days. Finally, a note in each TS, applicable to a previous temporary change, is deleted from the TS actions.

1.1 Background

1.1.1 Emergency Core Cooling System

In the PG&E letter dated December 17, 2007, the licensee stated that:

> The ECCS functions to provide core cooling and negative reactivity to ensure that the reactor core is protected after a design basis accident. The ECCS consists of 3 separate subsystems: centrifugal charging, SI [safety injection], and RHR [residual heat removal]. Each subsystem consists of two 100-percent capacity trains that are interconnected and redundant such that either train is capable of taking suction from the refueling water storage tank (RWST) and supplying 100 percent of the flow to the reactor core required to mitigate the accident consequences. The interconnecting and redundant subsystems design provides the operators with the ability to utilize components from opposite trains to achieve the required 100 percent flow to the core. Each ECCS train consists of an ECCS centrifugal charging pump (CCP), an SI pump, an RHR pump, piping, valves and heat exchangers. The ECCS pumps are normally in standby mode. In Modes 1, 2 and 3, two independent and redundant ECCS trains are required by the TSs to be OPERABLE to protect against a single failure, which could affect either train.

Requirements for the ECCS are contained in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" (Reference 4). The NRC staff evaluated the licensee's request in accordance with criteria in 10 CFR 50.46 and Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants." The applicable General Design Criteria (GDC) and regulatory requirements are discussed in the NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 6.3, "Emergency Core Cooling System," and Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary " (Reference 5).

The regulations in 10 CFR 50.46(a)(1)(i) requires, in part, that each pressurized light-water nuclear power reactor must be provided with an ECCS that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents (LOCAs) conforms to criteria set forth in 10 CFR 50.46(b)(1) through (b)(5), including requirements for peak fuel cladding temperature, maximum fuel cladding oxidation, maximum hydrogen generation, coolable core geometry, and long-term cooling.

1.1.2 Containment Spray System

In the PG&E letter dated December 17, 2007, the licensee stated that:

> The CS system is designed to provide containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. During a design basis accident inside containment, the CS system sprays refueling water storage tank (RWST) water, mixed with sodium hydroxide from the spray additive tank, into the upper region of containment. The CS system, together with the containment fan cooler units, provides the heat removal capability to reduce the containment pressure and temperature. The CS system is also credited to reduce fission products from the containment atmosphere. The CS system consists of two separate trains of containment spray pumps, spray headers, nozzles, valves and piping, and a common spray additive tank. Each train of the CS is capable of providing the necessary spray to fulfill the design function required containment atmospheric heat removal. The CS system takes suction from the RWST during the injection phase of operation. In the

recirculation phase of operation, CS is supplied by manual realignment of the RHR pumps to supply the CS header from the containment sump after the low water level is reached in the RWST.

The CS system is required to maintain a suitable containment environment such that the capability for long-term core cooling following a LOCA is preserved. The containment spray system is, therefore, considered a part of the ECCS, to the extent that it is a system required to function in order to meet the requirements of 10 CFR 50.46(b)(5). During the review of the licensee's amendment request, the NRC staff also considered the GDC applicable to the Containment Spray system, as discussed in the NUREG-0800, Chapter 6.2.2, "Containment Heat Removal Systems."

2.0 REGULATORY EVALUATION

In Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, the Commission established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. Of pertinence to this license amendment request, 10 CFR 50.36(c)(2) describes limiting conditions for operation (LCOs).

LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. The remedial actions in the TSs are specified in terms of LCO conditions, required actions, and CTs to complete the required actions. When an LCO is not being met, the CTs specified in the TSs are the time allowed in the TSs for completing the specified required actions. The conditions and required actions specified in the TSs must be acceptable remedial actions for the LCO not being met, and the CTs must be reasonable time for completing the required actions while maintaining the safe operation of the plant. The remedial actions allow the licensee to restore an inoperable train within a specified completion time, in order to avoid shutting down the plant as required by 10 CFR 50.36(c)(2). The license amendment request proposes, on a risk-informed basis, to extend the specified completion time. The licensee is not proposing any system or hardware configuration changes to the ECCS or CS system associated with the proposed TS amendments.

The DCPP TSs contain remedial actions when an ECCS train or CS subsystem is inoperable. Criterion 3 of 10 CFR 50.36(c)(2)(ii) require that a TS LCO must be established for: "[a] structure, system, or component that is part of the primary success path and which function or actuate to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

The licensee submitted its request, in part, based on the risk-informed guidance contained in References 2 and 3. Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," provides a general approach to evaluate and submit a proposal to change the licensing basis based on risk-informed decisions (Reference 2). RG 1.177, "An Approach for Risk-Informed, Plant-Specific Decisionmaking: Technical Specifications," contains more specialized guidance for evaluating changes that increase allowed outage times (AOTs) (Reference 3). The guidance

documentation is applicable, because the licensee's proposed increases to the required CT effectively increase the AOTs for portions of the ECCS and CS system.

The NRC staff reviewed DCPP's request in accordance with RG 1.174 and RG 1.177.

- RG 1.174, describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

- RG 1.177, describes an acceptable risk-informed approach specifically for assessing proposed permanent TS changes in allowed outage times. This RG also provides risk acceptance guidelines for evaluating the results of such assessments. RG 1.177 identifies a three-tiered approach for the licensees evaluation of the risk associated with a proposed CT TS change, as discussed below.

    - Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. The first tier assesses the impact on operational plant risk based on the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out-of-service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). Tier 1 also addresses Probabilistic Risk Assessment (PRA) quality, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Cumulative risk of the present TS change in light of past related applications or additional applications under review are also considered along with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

    - Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out-of-service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that there are appropriate restrictions in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed CT is implemented.

    - Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk significant configurations that may not have been considered when the Tier 2 evaluation was performed. Compared with Tier 2, Tier 3 provides additional coverage to ensure risk-significant plant equipment outage configurations are identified in a timely manner and that the risk impact of out of service equipment is appropriately evaluated prior to performing any maintenance activity over

extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application. The CRMP is to ensure that equipment removed from service prior to or during the proposed extended CT will be appropriately assessed from a risk perspective.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in SRP Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance." Guidance on evaluating PRA technical adequacy is provided in SRP Section 19.1, Rev. 2, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 5). More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, Rev.1, "Risk-informed Decision Making: Technical Specifications," (Reference 5), which includes CT changes as part of risk-informed decision making. SRP Section 19.2 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

1. The proposed change meets the current regulations, unless it explicitly relates to a requested exemption.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. When proposed changes increase core damage frequency or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
5. The impact of the proposed change should be monitored using performance measurement strategies.

3.0 TECHNICAL EVALUATION

The proposed changes are intended to allow a longer CT for the ECCS and CS system to accommodate unplanned corrective maintenance and inspections. A new note in the TS actions identify that the longer CTs do not apply to planned maintenance or inspection. The licensee has stated that it does not intend to increase planned maintenance activities during plant operation.

3.1 Detailed Description of the Proposed Change

The proposed TS change would modify TS 3.5.2 Condition A by providing three new Required Actions. Required Action A.2.1 would verify that only one subsystem in one ECCS train was inoperable; Required Action A.2.2 would determine that no common cause failure exists in the same redundant subsystem in the opposite train. If these two actions are satisfied within 72 hours, then Required Action A.2.3 could be applied to permit 14 days to restore the inoperable

ECCS subsystem, provided that the cause of the inoperability was not for planned maintenance or inspections. By letter dated November 18, 2008, the licensee proposed adding a new Note to TS 3.5.2 to specify the limitations on the use of the 14-day extended CT. In addition, a note in TS 3.5.2 Condition A, applicable to allow a one-time CT extension for DCPP, Unit 1 Cycle 12 for a centrifugal charging pump seal replacement, is being deleted. The work has been completed and the note is no longer applicable.

The proposed changes would modify TS 3.6.6 Condition A by providing a new Required Action A.2. Required Action A.2 would permit up to 14 days to restore the inoperable CS subsystem, provided that the cause of the inoperability was not for planned maintenance or inspections. By letter dated November 18, 2008, the licensee proposed adding a new Note to TS 3.6.6 to specify the limitations on the use of the 14-day extended CT. The proposed change also adds a second CT for Required Action A.2 which limits its use to 14 days from discovery of the failure to meet the LCO. The second CT establishes a limit on the maximum time allowed for any combination of CS pumps and containment fan coolers to be inoperable during any single contiguous occurrence of failing to meet the limiting condition for operation (LCO). In addition, a note in TS 3.6.6 Condition A, applicable to allow a one-time CT extension for DCPP Unit 2 for CS pump control circuit repair, is being deleted. This work was completed and the note is no longer applicable.

3.1.1 TS 3.5.2 Note

The licensee stated it is deleting a Note in TS 3.5.2, Condition A, that was applicable to allow a one-time CT extension for DCPP, Unit 1, during Cycle 12 for component cooling pump seal replacement. This work was completed and the note is no longer applicable. The NRC staff concludes that this Note can be deleted as this proposed TS change is editorial-in-nature and therefore, is acceptable.

3.1.2 TS 3.6.6 Note

The licensee stated it is deleting a Note in TS 3.6.6, Condition A, that was applicable to allow a one-time CT extension for DCPP, Unit 2, for CS pump control circuit repair. This work was completed and the note is no longer applicable. The NRC staff concludes that this Note can be deleted as this proposed TS change is editorial-in-nature and therefore, is acceptable.

TS 3.6.6 Condition A

This proposed TS change adds a second CT for new Required Action A.2 which limits its use to 14 days from discovery of the failure to meet the LCO. The second CT establishes a limit on the maximum time allowed for any combination of CS pumps and containment fan coolers to be inoperable during any single contiguous occurrence of failing to meet the limiting condition for operation (LCO). This logic connector is typically used in TS sections with more than one function. In TS 3.6.6, the two functions are CS and containment cooling systems and this TS also addresses a Condition D when one train of CS and one train of containment cooling system are inoperable. The purpose of the logical connector is to prevent the situation where switching between Condition A and Condition D or Condition C and Condition D would allow indefinite continued operation while not meeting either of the LCO of Conditions A or C. The "14 days" is based on the CT of Condition A.2. This is the least amount of time Condition A.2 is allowed in this situation and is conservative. For instance, the LCO for TS 3.6.6, Condition C is 7 days and

10 days from discovery of failure to meet the LCO. The NRC staff agrees that this change is consistent with the CT extension and is conservative and, therefore, is acceptable.

3.2 Review Methodology

In accordance with SRP Section 19.2 and Section 16.1, the NRC staff reviewed the TS amendment regarding the extension of the ECCS and CS system CTs using the three-tiered approach and the five key principles of risk-informed decision-making presented in RG 1.174 and RG 1.177.

3.3 Key Information Used in the Review

The key information used in the NRC staff review is contained in Section 4.2 of Enclosure 1 of the license amendment request dated December 17, 2007 (Reference 1), as modified by requests for additional information (RAI) responses dated October 2, 2008 (Reference 7), and November 18, 2008 (Reference 8).

3.4 Comparison Against Regulatory Criteria/Guidelines

The NRC staff evaluation of the licensee's proposed changes to TS 3.5.2, "ECCS - Operating," and TS 3.6.6, "Containment Spray and Cooling Systems," using the three-tiered approach and the five key principles outlined in RG 1.174 and RG 1.177, are presented in the following sections.

3.4.1 Traditional Engineering Evaluation

The traditional engineering evaluation addresses key principles 1, 2, 3, and 5 in RG 1.177, which concerns compliance with current regulations, evaluation of defense-in-depth, evaluation of safety margins, and performance monitoring strategies.

Key Principle 1: Compliance With Current Regulations

In the PG&E letter dated December 17, 2007, the licensee stated that no system or hardware configuration changes are proposed as a part of this request. Therefore, the NRC staff finds that an ECCS and CS system of suitable redundancy will still be provided for compliance with 10 CFR 50.46(a)(1)(i) following the proposed change.

The regulations in 10 CFR 50.46(a)(1)(i) states that each pressurized ligh-water nuclear power reactor must be provided with an ECCS that conforms to criteria set forth in 10 CFR 50.46(b). The NRC staff has found previously that the analysis of record at DCPP demonstrates suitable ECCS performance to comply with these criteria. The NRC staff's review of the analysis of record determined that it contains conservative assumptions regarding limiting single failures, initial conditions, and equipment operability.

The TSs in place at DCPP assure that the analytic assumptions remain conservative based on requirements for equipment operability pertaining to the ECCS and CS system. As required by the TSs, both trains of the ECCS and CS system must be operable. This requirement will not change following implementation of the proposed amendment. Therefore, the NRC staff finds that the licensee may implement the proposed TS modifications without revising LOCA analyses

to demonstrate compliance with the 10 CFR 50.46(b) LOCA criteria. The LOCA analyses demonstrate the predicted capability of the ECCS and CS system, such that there is reasonable assurance that the licensee will remain in compliance with 10 CFR 50.46(a)(1)(i) following implementation of the proposed license amendment.

Because the proposed changes provide a risk-informed basis to increase the required CTs, and the licensee does not propose to change the LCOs associated with the ECCS and CS system, the NRC staff finds that the licensee remains in compliance with 10 CFR 50.36(c)(2)(ii). The NRC staff evaluation of the risk analysis demonstrated that the proposed increases in CTs are acceptable to satisfy the intent of 10 CFR 50.36(c)(2)(ii).

Based on the above, the NRC staff has determined that the licensee has adequately met the intent of Regulatory Position 2.1 of RG 1.177 for the proposed TS modifications and therefore, is acceptable.

Key Principle 2: Evaluations of Defense-in-Depth

In the enclosure to the PG&E letter dated December 17, 2007, the licensee provided an evaluation that concludes that the proposed increase in CT is consistent with the defense-in-depth philosophy, as recommended in Regulatory Position 2.2.1, "Defense-in-Depth," of RG 1.177. The evaluation is needed based on the added risk due to the proposed extension of the CTs.

The NRC staff reviewed the licensee's evaluation for consistency with defense-in-depth philosophy which considered each of the points stipulated in Regulatory Position 2.2.1 of RG 1.177. One of these points is that since the proposed extension contributes added risk by increasing the CTs a PRA analysis may be used to help determine the appropriate extent of defense-in-depth. The NRC staff has determined that the licensee has adequately met the intent of Regulatory Position 2.2.1 of RG 1.177 for the proposed TS modifications and, therefore, is acceptable.

Key Principle 3: Evaluation of Safety Margins

In the PG&E letter dated December 17, 2007, the licensee considered Regulatory Position 2.2.2, "Safety Margins," of RG 1.177. This position directs the licensee to consider whether sufficient safety margins are maintained in light of the proposed change. The maintenance of sufficient safety margins is demonstrated by compliance with applicable codes and standards and NRC safety analysis acceptance criteria. The licensee stated that, because the proposed TS revisions increase the CTs, and involves no design changes, the revisions do not conflict with applicable codes and standards, nor do they adversely affect assumptions or inputs to the safety analysis.

The NRC staff reviewed the licensee's evaluation of consistency with safety margins. The NRC staff has determined that the licensee has adequately met the intent of Regulatory Position 2.2.2 of RG 1.177 for the proposed TS modifications and, therefore, is acceptable.

Key Principle 5: Performance Measurement Strategies - Implementation and Monitoring Program

The licensee considered Regulatory Position 3.0, "Define Implementation and Monitoring Program," of RG 1.177. This position directs the licensee to use a three-tiered approach for implementation, and 10 CFR 50.65 (the Maintenance Rule) for monitoring. The licensee is using the three-tiered approach as discussed below. RG 1.174 and RG 1.177 establish the need for an implementation and monitoring program to ensure that extensions to TS CTs do not degrade operational safety over time, and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms. An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of systems, subsystems, and components (SSCs) impacted by the change. RG 1.174 states that monitoring performed in conformance with the Maintenance Rule can be used when the monitoring performed is sufficient for the SSCs affected by the risk-informed application. In the PG&E letter dated December 17, 2007, the licensee stated that:

> With the implementation of 10 CFR 50.65, "Maintenance Rule," risk with on-line maintenance activities is assessed and managed. This ensures that multiple safety systems will not be taken out-of-service simultaneously during extended CTs that could lead to degradation of these barriers and an increase in risk to the public.

The NRC staff has determined that the licensee has adequately met the intent of Regulatory Position 3.0 of RG 1.177 for the proposed TS modifications and, therefore, is acceptable.

3.4.2 PRA Evaluation

The evaluation presented below addresses the NRC staff philosophy of risk-informed decision making, that when the proposed changes result in a change in Core Damage Frequency (CDF) or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (Key Principle 4: Assessment of Impact on Risk)

3.4.2.1 Tier 1: PRA Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk. The Tier 1 NRC staff review involves two aspects: (1) evaluation of the validity of the DCPP PRA models and their application to the proposed changes, and (2) evaluation of the PRA results and insights based on the licensee's proposed application.

PRA Quality

The objective of the PRA quality review is to determine whether the DCPP PRA used in evaluating the proposed changes to TS 3.5.2 and 3.6.6 CTs is of sufficient scope, level of detail, and technical adequacy for this application. The NRC staff review evaluated the PRA quality information provided by the licensee in their submittals, including industry peer review results.

The DCNP PRA model is a full-scope model addressing both level one (core damage) and level two (containment performance and large early release) for internal events, seismic events, and

fire events at full power. The DCNP PRA model is performed for Unit 1, but the results are equally applicable to Unit 2, because the two units are essentially identical.

The PRA model is based on the original 1988 DCPP PRA performed as part of the long-term seismic program, which was a full scope PRA. The model was updated to support the individual plant examination (IPE), and the IPE for external events (IPEEE). Since the IPEEE, several model updates have been made to incorporate plant and procedure changes, update plant-specific reliability and unavailability data, improve the fidelity of the model, incorporate peer review comments, and to support other applications. The DCPP PRA model was peer-reviewed in May 2000, using the Westinghouse Owners' Group Peer Review Certification Guidelines (Reference 11). The significant findings from this review have been dispositioned.

The human reliability analysis (HRA) underwent a major enhancement, and was subjected to a focused peer review. In addition, three limited scope independent assessments of the PRA model were recently performed for the internal events, level two, and flooding portions of the PRA model. The licensee provided its disposition of the findings (Reference 8) from these evaluations with regard to this application. Specifically, (1) none of the level two or internal flooding items were judged to have any significant impact on the conclusions of the application, (2) documentation issues identified with the HRA items were stated not to affect the results, and (3) remaining HRA issues were either unrelated to the application or were shown to be insignificant using a sensitivity evaluation, which the licensee also provided.

The NRC staff reviewed the licensee's evaluation of these deficiencies identified by these reviews, and concluded that the results of the risk analyses supporting this application are insensitive to the items identified. Specifically:

- The ECCS and CS systems would not be relied upon to provide significant mitigation of flooding events, and so deficiencies in this portion of the PRA model would not impact this application.

- The issues identified for the level two analyses were either documentation issues, issues not related to LERF aspects of the level two analyses, or model conservatisms which would tend to overestimate the LERF for this application.

- The human error events applicable to the risk evaluations supporting this license application request were increased by a factor of five as a bounding sensitivity assessment of the potential impact of the unresolved HRA issues. The sensitivity analysis results show only a small effect on the calculated CDF and LERF, and so the HRA issues identified were judged not to impact this application.

The licensee identified the truncation level applied for this application as being in the range of 1E-12/year to 1E-15/year depending upon the specific initiating event being evaluated. This value is acceptably low to assure that significant sequences are sufficiently accounted for in the risk evaluation.

While the licensee has stated that it does not intend to increase planned maintenance activities, an evaluation of increased unavailability was performed to address the increased CT request. The TS changes preclude the application of the extended CTs for planned activities, and are instead limited to emergent repairs.

Based on review of the above information, the NRC staff concludes that the licensee has satisfied the intent of RG 1.177 (Sections 2.3.1, 2.3.2, and 2.3.3), RG 1.174 (Section 2.2.3 and 2.5), and SRP Chapter 19.2, and that the quality of the DCPP PRA is sufficient to support the risk evaluation provided by the licensee in support of the proposed license amendment.

PRA Results and Insights

The licensee calculated the risk impacts of postulated outages of one train of CS, centrifugal charging, high pressure SI, or RHR. This is consistent with the proposed TS changes, which limit the extended CT for ECCS systems to a single subsystem train being inoperable. The licensee's calculations assumed that the train was inoperable due to emergent failure, and so the CCF probability of the redundant equipment was assumed to be increased.

For ΔCDF and ΔLERF, the licensee stated that the intended use of the extended CTs are for emergent repairs only, and the proposed TSs directly restrict planned unavailability to the existing 72-hour CTs. Therefore, the anticipated long-term change in CDF and LERF is minimal, potentially increasing only to the extent that emergent failures of the ECCS or CS systems occur and require extended repair times. The licensee assessed the ΔCDF and ΔLERF based on an increase in train unavailability by a factor of 14/3 (14 days versus 3 days) to reflect the increased CT as a sensitivity study, and these values are judged by the NRC staff to bound any potential risk increase associated with this change.

The risk assessment calculations assume that the probability of common cause failures (CCF) on the operable redundant train is elevated, based on an emergent failure having occurred on the out-of-service train.

The licensee's methodology is consistent with the guidance of RG 1.177, Section 2.3.4 and Section 2.4 and is, therefore, acceptable to the NRC staff.

The results of the licensee analyses are shown in Table 1.

Table 1: Configuration-Specific Risk Results

Case	Level One	Level Two
Baseline	CDF= 6.556E-5/year	LERF = 3.489E-6/year
Acceptance Guidelines[1]	ICCDP < 5E-7 ΔCDF < 1E-6/year	ICLERP < 5E-8 ΔLERF < 1E-7/year
One CS Train	N/A[3]	ICLERP = 1.21E-9 ΔLERF = 3.91E-10[2]
One RHR Train	ICCDP = 7.33E-7 ΔCDF = 4.72E-8/year[2]	ICLERP = 1.55E-7 ΔLERF = 5.64E-9/year[2]
One CCP Train	ICCDP = 1.93E-7 ΔCDF = 2.73E-8/year[2]	ICLERP = 4.99E-9 ΔLERF = 7.39E-10/year[2]
One High Pressure SI Train	ICCDP = 8.16E-8 ΔCDF = 1.87E-8/year[2]	ICLERP = 2.24E-9 ΔLERF = 5.13E-10/year[2]

1 Acceptance Guidelines from RG 1.174 for very small changes for ΔCDF and ΔLERF, and RG 1.177 for ICCDP and ICLERP.
2 Results from sensitivity analyses assuming maintenance unavailability increases by 14/3 for each affected ECCS and CS train.
3 The CS system provides no mitigation of core damage.

The risk metrics associated with a CS train, CCP train, and High Pressure SI train are within the guidelines of RG 1.174 and RG 1.177. The metrics for one RHR train are within the RG 1.174 metrics for ΔCDF and ΔLERF, but slightly above the ICCDP and ICLERP guidelines of RG 1.177. These calculations assume an elevated CCF probability of the operable redundant components, and the numerical result is only slightly above the guidance. The licensee also evaluated the ICCDP and ICLERP over the 14-day period assuming no CCF mechanism, and for the RHR train, the results were 1.60E-7 and 1.91E-8, respectively, which are within the RG 1.177 guidance. The licensee has proposed to include a new action requirement, prior to exceeding the existing 72-hour CT, to determine there is no CCF mechanism in the redundant ECCS subsystem. The new action requirement is applicable to all three ECCS subsystems, including inoperability of an RHR train. This has the effect of reducing the risk impact since the likelihood of a common cause failure mechanism is lessened. Assuming that the CCF mechanism may exist for the first 72-hours of the 14-day CT, then the risk of the RHR train outage was determined by the NRC staff to be as follows:

ICCDP: (3/14)*(7.33E-7) + (11/14)*(1.60E-7) = 2.83E-7
ICLERP: (3/14)*(1.55E-7) + (11/14)*(1.91E-8) = 4.82E-8

These values are within the acceptance guidelines of RG 1.177.

The NRC staff finds that the licensee has satisfied the intent of RG 1.177 (Sections 2.4), RG 1.174 (Section 2.2.4 and 2.2.5), and SRP Chapter 19.2 and, therefore, is acceptable.

External Events

The DCPP PRA model is a full scope model which includes contributions from internal fires, internal floods, seismic events, or other external events. The risk impact of the proposed change from these events was therefore included in the risk metric calculations. The licensee also presented the separate results for seismic, fire, and other external events, and demonstrated quantitatively that the ECCS and CS system CT extensions are not significant to these events.

Shutdown and Transition Risk

The licensee did not provide an assessment of shutdown or transition risk. Because the proposed TS changes are not applicable in Modes 5 and 6, shutdown risk is not relevant to the proposed change. The risk analysis presented used an at-power PRA model, which assumes Mode 1 operation. Although the proposed changes to TS are applicable in Modes 2 - 4, the plant does not typically operate in these modes for extended periods, and is in these modes only during transition from power operations to outage conditions, and return to service following outages. Based on the reduced time spent in these transition modes, a detailed transition risk analysis is not required.

Uncertainty

The licensee's risk analysis results are based on point estimates of the mean values. The licensee did not provide an analysis which used a quantification method which propagates uncertainties within the solution. RG 1.177 identifies that the change in risk from TS CT extensions is relatively insensitive to uncertainties because they affect both the base case and the changed case.

The licensee identified the key contributors to modeling uncertainty for this application as seismic event frequency and impacts resulting from uncertainty in the seismic site hazard data and the plant equipment fragility data. These sources of uncertainty are the same as the base model, and the application has not introduced any new sources of uncertainty to be addressed.

The licensee also identified the total unavailable times for the affected equipment as a source of uncertainty. The restricted use of the extended CTs to support corrective repairs only, rather than to increase planned maintenance activities conducted on line, should result in no significant increase in the unavailability of the affected ECCS and CS components. This uncertainty has been satisfactorily addressed by the licensee's use of conservative assumptions in evaluating the risk associated with implementation of this change, specifically in assuming an increase in train unavailability by a factor of 14/3.

3.4.2.2 Tier 2 - Avoidance of Risk-Significant Plant Configurations

The second tier requires a licensee to provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out of service in accordance with the proposed TS change. In the PG&E letter dated December 17, 2007, the licensee identified both TS and administrative controls applicable upon entry into plant conditions which use an extended CT.

The proposed TS action in TS 3.5.2 would prohibit more than one ECCS subsystem being inoperable during the extended CT. This assures that a higher risk configuration involving multiple ECCS subsystems (i.e., both a high and low pressure SI pump) is not permitted. The existing TS LCO 3.6.6 Condition D requires operability of the containment fan cooling unit system while a CS train is inoperable. This assures no significant loss of capability for the containment cooling function during an extended CS pump outage.

TS 5.5.15, "Safety Function Determination Program (SFDP)," requires verification that a loss of capability to perform a safety function is not undetected, by performing cross-train checks during inoperability of safety-related support systems.

The licensee identified that the risk associated with inoperability of one train of the ECCS or CS systems is addressed by DCPP procedure AD7.DC6, "On-Line Maintenance Risk Management." Risk management actions and restrictions, such as around-the-clock maintenance to minimize the time spent with equipment unavailable, redundant equipment operability walkdowns, verifications and postings, development of criteria and procedures for restoration of inoperable equipment, and other administrative controls are addressed by this procedure, which would apply to the proposed extended CTs.

Based on the above, and considering the small risk increase noted for the proposed TS change, the NRC staff concludes that the licensee's Tier 2 evaluation of potential risk significant configurations support the implementation of changes to TS 3.5.2 and 3.6.6, and is acceptable.

3.4.2.3 Tier 3 - Risk-Informed Configuration Risk Management

The third tier requires a licensee to develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity.

The licensee identified its process for online risk assessment and management, procedure AD7.DC6, as assuring this requirement is met during any use of the proposed extended CTs. This procedure addresses risk management in the planning and execution phase of maintenance activities during plant operational modes, and includes real-time evaluation of risk. The process minimizes the total number of plant components out-of-service simultaneously, and avoids higher risk combinations of out-of-service components based on PRA insights. Defense-in-depth is maintained by avoiding combinations of out-of-service components that are related to similar safety functions or which affect multiple safety functions.

Based on the licensee's conformance to the requirements of the guidelines of RG 1.177 for managing configuration risk during an extended CT, the NRC staff finds the licensee's Tier 3 program is acceptable and supports the proposed changes to TS 3.5.2 and 3.6.6, and is acceptable to the NRC staff.

Comparison With Regulatory Guidance

The proposed changes to TS 3.5.2 and 3.6.6 to extend the CTs for one inoperable ECCS subsystem or one CS train are reasonably consistent with the acceptance guidance of RG 1.174 and RG 1.177, and the guidance outlined in SRP Chapter 19.0, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," and SRP Chapter 16.1, "Risk-Informed Decisionmaking: Technical Specifications," of NUREG-0800.

3.5 PRA Findings

The risk impacts for ΔCDF, ΔLERF, ICCDP, and ICLERP, as estimated by the licensee, are consistent with the acceptance guidelines for RG 1.174 and RG 1.177 for the proposed changes to TS 3.5.2 and 3.6.6 to extend the CTs for one inoperable ECCS subsystem and one CS train. The licensee's Tier 2 analysis provides reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific-plant equipment is taken out of service in accordance with the proposed TS change. The licensee's Tier 3 configuration risk management program (CRMP) was found to be consistent with the RG 1.177 CRMP guidelines.

3.6 Conclusions

Based on the traditional engineering considerations discussed in Regulatory Positions 2.1 of RG 1.177, the licensee has demonstrated continued compliance with the applicable NRC regulations identified in Section 3.4.1 of this Safety Evaluation. Also, the licensee's evaluations demonstrate acceptable compliance with the intent of Regulatory Positions 2.2.1, 2.2.2 and 3.0 of RG 1.177. These evaluations concluded that the proposed TS changes have no impact on the redundancy, independence, or diversity of these systems and subsystems and all elements

of the defense-in-depth principle and safety margins are met. The risk assessment concluded that the increase in plant risk is small and consistent with the acceptance guidelines in RGs 1.174 and 1.177. Based on the above, the NRC staff concludes that the TS changes regarding the extended CTs for the ECCS and CSS meet the requirements of 10 CFR 50.36(c)(2) and are acceptable.

3.7 TS Bases

The licensee identified changes to the TS Bases for the proposed amendment. The TS Bases are licensee controlled and governed by TS 5.5.14, "TS Bases Control Program." The NRC staff has reviewed the proposed changes to the TS Bases for technical correctness and we have no objections.

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 (73 FR 5227). 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.

REFERENCES

1. Letter from J. R. Becker (PG&E), to NRC, dated December 17, 2007 (ADAMS Accession No. ML073650012).

2. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002 (ADAMS Accession No. ML023240437).

3. Regulatory Guide 1.177, “An Approach for Risk-Informed, Plant-Specific Decision Making: Technical Specifications,” August 1998 (ADAMS Accession No. ML003740176).

4. 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.”

5. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants”.

6. Diablo Canyon Power Plant Final Safety Analysis Report Update, Revision 18, October 2008.

7. Letter from J. R. Becker (PG&E), to NRC dated October 2, 2008 (ADAMS Accession No. ML082890537).

8. Letter from J. R. Becker (PG&E), to NRC dated November 18, 2008 (ADAMS Accession No. ML083370199).

9. Regulatory Guide 1.200 for Trial Use, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” February 2004.

10. ASME RA-Sb-2005, Addenda to ASME RA-S-2002, “Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,” December 2005.

11. NEI 00-02, Probabilistic Risk Assessment (PRA) Peer Review Process Guidance, 2000.

12. NEI 04-10, Risk-Informed Technical Specifications Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies, Revision 1, April 2007.

13. ANSI/ANS-58.23-2007, “Fire PRA Methodology,” November 2007.

14. NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” September, 2005.

Principal Contributors: A. Howe
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Date: December 31, 2008

December 31, 2008

Mr. John Conway
Senior Vice President - Station
Generation and Chief Nuclear Officer
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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: INCREASE IN THE COMPLETION TIMES FOR REQUIRED ACTIONS RELATED TO TECHNICAL SPECIFICATIONS 3.5.2, REGARDING THE EMERGENCY CORE COOLING SYSTEM, AND 3.6.6, REGARDING THE CONTAINMENT SPRAY AND COOLING SYSTEMS (TAC NOS. MD7512 AND MD7513)

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. 202 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated December 17, 2007, as supplemented by letters dated October 2, and November 18, 2008.

The amendments increase the completion times for required actions related to Technical Specifications 3.5.2, regarding the Emergency Core Cooling System, and 3.6.6, regarding the Containment Spray and Cooling Systems from 72 hours to 14 days.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next 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. 202 to DPR-80
2. Amendment No. 203 to DPR-82
3. Safety Evaluation

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