January 8, 2008

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: REVISE TECHNICAL SPECIFICATIONS TO SUPPORT STEAM GENERATOR REPLACEMENT (TAC NOS. MD3992 AND MD3993)

Dear Mr. Keenan:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 198 to Facility Operating License No. DPR-80 and Amendment No. 199 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 January 11, 2007, as supplemented by letters dated August 9, and September 28, 2007.

The amendments revise the TS to support replacement of the steam generators. Revisions are proposed to TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator (SG) Tube Inspection Report."

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. 198 to DPR-80
2. Amendment No. 199 to DPR-82
3. Safety Evaluation

cc w/encls: See next page

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GHill, OIS (4)

ADAMS Accession Nos.: Pkg ML073240002 (Amdt./License ML073240006 TS Pgs ML073240008)
(*)SE input Memo (**) See previous concurrence

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	DSS/SRXB/BC	DCI/CSGB/BC	OGC - NLO	NRR/LPL4/BC
NAME	AWang	JBurkhardt	GCranston(*)	AHiser(*)	APHodgdon (**)	THiltz
DATE	1/8/08	1/8/08	10/31/07	10/26/07	12/10/07	1/8/08

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. 198
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 January 11, 2007, as supplemented by letters dated August 9, and September 28, 2007, 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. 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. This license amendment is effective as of its date of issuance and shall be implemented prior to entry into Mode 4 following the 15th refueling outage.

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 No. DPR-80
and Technical Specifications

Date of Issuance: January 8, 2008

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 199
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 January 11, 2007, as supplemented by letters dated August 9, and September 28, 2007, 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. 199, 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 prior to entry into Mode 4 following the 14th refueling outage.

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 No. DPR-82
and Technical Specifications

Date of Issuance: January 8, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 198

TO FACILITY OPERATING LICENSE NO. DPR-80 AND

AMENDMENT NO. 199 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
3.3-31	3.3-31
5.0-10	5.0-10
5.0-11	5.0-11
5.0-12	--
5.0-13	--
5.0-14	--
5.0-15	--
5.0-16	--
5.0-17	--
5.0-18	--
5.0-19	--
5.0-20	5.0-12
5.0-21	5.0-13
5.0-22	5.0-14
5.0-23	5.0-15
5.0-24	5.0-16
5.0-24a	5.0-17
5.0-25	5.0-18
5.0-26	5.0-19
5.0-27	5.0-20
5.0-27a	5.0-21
5.0-28	5.0-22
5.0-29	5.0-23
5.0-30	--
5.0-30a	--
5.0-30b	--
5.0-31	5.0-24
5.0-32	5.0-25
5.0-33	5.0-26

(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. 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

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. 199, 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. 198 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 199 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 January 11, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070190094), as supplemented by letters dated August 9, and September 28, 2007 (ADAMS Accession Nos. ML072260512 and ML072840047, respectively), Pacific Gas and Electric Company (PG&E or the licensee) requested changes to the Technical Specifications (TS, Appendix A to Facility Operating License Nos. DPR-80 and DPR-82) for the Diablo Canyon Power Plant, Units 1 and 2 (DCPP), respectively.

The proposed amendments would revise TS 3.3.2, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation,” TS 5.5.9, “Steam Generator (SG) Program,” and TS 5.6.10, “Steam Generator (SG) Tube Inspection Report.” Specifically, the proposed changes would revise TS 3.3.2 to change the Nominal Trip Setpoint (NTSP) and Allowable Value (AV) and clarify the surveillance requirements (SRs) associated with ESFAS function 5.b, “Feedwater Isolation SG Water Level-high High.” The TS 3.3.2 changes are consistent with TS Task Force (TSTF) Standard TS Change Traveler TSTF-493, “Clarify Application Setpoint Methodology for LSSS [Limiting Safety System Settings] Functions,” Revision 1. In addition, changes to TS 5.5.9 and TS 5.6.10 were proposed and the proposed changes are consistent with U.S. Nuclear Regulatory Commission (NRC)-approved TSTF Traveler, TSTF-449, “Steam Generator Tube Integrity,” Revision 4. The availability of this TS improvement was announced in the *Federal Register* on May 6, 2005, as part of the consolidated line item improvement process (CLIIP).

The supplemental letters dated August 9, and September 28, 2007, 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's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 13, 2007 (72 FR 6787).

2.0 REGULATORY EVALUATION

NRC Regulatory Issue Summary (RIS) 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006, discusses the requirements of Part 50, Section 36 of Title 10 of the *Code of Federal Regulations* (i.e., 10 CFR 50.36) related to Limiting Safety System Settings and provides an approach acceptable to the NRC to address LSSS issues. LSSS are settings for automatic protective devices related to those variables having significant safety functions.

RIS 2006-17 provides guidance on how to determine when as-found values are acceptable with respect to the NTSP and required actions to be taken when the as-found value is outside predefined acceptance limits or outside the AV. TSTF-493, Revision 1, incorporates this guidance by specifying the requirements for assessing whether an instrument channel is operable based on the as-found setpoint and describes the required actions before returning a channel to service. In addition, the NRC provided comments on TSTF-493, Revision 1, in a letter dated December 14, 2006. Since the SG replacement requires changes to the Feedwater Isolation SG Water Level-High High (P-14) ESFAS setpoint, the guidance of TSTF-493, Revision 1, and the NRC letter dated December 14, 2006, is applied to ESFAS Function 5.b, Feedwater Isolation SG Water Level-High High (P-14). The licensee has stated that the TSTF-493 changes to the remaining applicable Reactor Trip System (RTS) and ESFAS functions will be the subject of a separate license amendment request (LAR). That LAR will be submitted after TSTF-493 is approved by the NRC as part of a CLIIP. The NRC staff used the following references in its review of the SG Water Level-High High (P-14) setpoint change:

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 36, "Technical Specifications," states, "[e]ach applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section." Specifically, paragraph 50.36(c)(1)(ii)(a) states, "[w]here a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Furthermore, paragraph 50.36(c)(3) states, "[s]urveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 13, "Instrumentation and Control," requires that the instrumentation be provided to monitor variables and systems and that controls be provided to maintain these variables and systems within prescribed operating ranges.

- 10 CFR Part 50, Appendix A, Criterion 20, "Protection System Functions," requires that the protection system be designed to initiate operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded.

- Regulatory Guide (RG) 1.105, Revision 3, “Setpoints for Safety-Related Instrumentations,” describes a method acceptable to the NRC staff for complying with the NRC’s regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits. The RG endorses Part I of ISA-S67.04-1994, “Setpoints for Nuclear Safety Instrumentation,” subject to the NRC staff clarifications.

- Letter from Timothy J. Kobetz, NRC, to Technical Specifications Task Force (TSTF), TSTF Traveler 493, Revision 1, “Clarify Application of Setpoint Methodology for LSSS Functions,” dated December 14, 2006, available on the NRC public website under ADAMS Accession No. ML063450324.

- Letter from Patrick L. Hiland, NRC, to NEI [Nuclear Energy Institute] Setpoint Methods Task Force, "Technical Specification for Addressing Issues Related to Setpoint Allowable Values," dated September 7, 2005 (ADAMS Accession No. ML052500004). This letter addresses the footnotes that should be added to SRs related to setpoint verification surveillance for instrument functions on which a safety limit has been placed and the information to be included to ensure operability of the instruments following surveillance tests related to instrument setpoints.

- Letter from James A. Lyons, NRC, to Alexander Marion, NEI, "Instrumentation, Systems, and Automation Society S67.04 Methods for Determining Trip Setpoints and Allowable Values for Safety-Related Instrumentation," dated March 31, 2005 (ADAMS Accession No. ML051660447).

- Letter from Bruce A. Boger, NRC, to Alexander Marion, NEI, "Instrumentation, Systems, and Automatic Society (ISA) S67.04 Methods for Determining Trip Setpoints and Allowable Values for Safety-Related Instrumentation," dated August 23, 2005 (ADAMS Accession No. ML050870008).

In addition, TS 5.5.9 and TS 5.6.10 are being revised to delete the existing SG tube alternate repair criteria (ARC) and associated reporting requirements. The existing TS 5.5.9.b.1 reference to the ARC, the TS 5.5.9.b.1 structural integrity performance criteria for Tube Support Plate Voltage-Based Repair Criteria and Axial Primary Water Stress Corrosion Cracking (PWSCC) Depth-Based Repair Criteria, the TS 5.5.9.b.2 Tube Support Plate Voltage-Based Repair Criteria, W* Repair Criteria, and Axial PWSCC Depth-Based Repair Criteria, the TS 5.5.9.d tube inspection requirements for the ARC, and the TS 5.6.10.b through 5.6.10.g ARC reporting criteria, are deleted since they are not applicable to the replacement steam generators (RSGs). SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, tube integrity means that the tubes are capable of performing these functions in accordance with the plant design and licensing basis.

Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the fundamental regulatory requirements with respect to the integrity of the SG tubing. Specifically, the General Design

Criteria (GDC) in Appendix A to 10 CFR Part 50 state that the RCPB shall have "an extremely low probability of abnormal leakage...and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDCs 15 and 31), shall be of "the highest quality standards possible" (GDC 30), and shall be designed to permit "periodic inspection and testing ... to assess ... structural and leak tight integrity" (GDC 32). To this end, 10 CFR 50.55a specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Section 50.55a further requires, in part, that throughout the service life of a pressurized-water reactor (PWR) facility, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements, in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. Section XI requirements pertaining to inservice inspection of SG tubing are augmented by additional SG tube SRs in the TSs.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents such as an SG tube rupture and main steamline break. These analyses consider the primary-to-secondary leakage through the tubing which may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of 10 CFR Part 100 for offsite doses (or 10 CFR 50.67, as appropriate), GDC 19 criteria for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits).

The DCPP TSs are modeled after TSTF-449, "Steam Generator Tube Integrity," Revision 4. TS 5.5.9 for DCPP requires that an SG program be established and implemented to ensure that SG tube integrity is maintained. Tube integrity is maintained by meeting specified performance criteria for structural and leakage integrity consistent with the plant design and licensing bases. TS 5.5.9 requires a condition monitoring assessment be performed during each outage during which the SG tubes are inspected to confirm that the performance criteria are being met. TS 5.5.9 also includes provisions regarding the scope, frequency, and methods of SG tube inspections.

3.0 TECHNICAL EVALUATION

Each unit at DCPP currently has four Westinghouse Model 51 SGs with mill-annealed Alloy 600 tubes. In addition to a depth-based tube repair criteria, the licensee is authorized to apply the voltage-based tube repair criteria for predominantly axially-oriented outside diameter stress-corrosion cracking within the tube support plates. The licensee is also authorized to implement an ARC for PWSCC indications at the tube support plate elevations and to leave certain flaws within the tubesheet region in service, provided they satisfy the W* repair criterion.

The licensee currently plans to replace the SGs at both units. The RSGs are Westinghouse Model Delta 54 with Alloy 690 thermally treated tubes. The SGs for Unit 2 are scheduled to be replaced during the 14th refueling outage (2R14), in February 2008, and the SGs for Unit 1 are scheduled to be replaced during the 15th refueling outage (1R15), currently scheduled for January 2009. The licensee concluded that the existing SGs and RSGs are similar and, therefore, the SGs' replacement evaluation was performed under 10 CFR 50.59.

3.1 Steam Generator Replacement 10 CFR 50.59 Evaluation

Westinghouse performed a comprehensive review of the updated final safety analysis report (UFSAR) Chapter 15 accidents and transient analyses. Westinghouse performed loss-of-coolant accident (LOCA) and non-LOCA analyses and evaluations to demonstrate that the Nuclear Steam Supply System (NSSS) is in compliance with applicable licensing acceptance criteria and requirements at the current NSSS thermal power of 3425 megawatts thermal (MWt) (3411 MWt core power + 14 MWt reactor coolant pump net heat input) with the Model Delta 54 RSG design and operating parameters. The analyses or evaluations were performed using NRC-approved analytical methods to demonstrate compliance with the licensing acceptance criteria and standards. In the analysis of a few non-LOCA events, the secondary system was not modeled because the event is a fault occurring on the primary side and occurs too rapidly to be influenced by the secondary-side conditions. In this case, the analysis is insensitive to the specific design and operating properties of the SGs. Some transient events are particularly sensitive to the primary-to-secondary system heat transfer and SG design characteristics. These events have been reanalyzed to model the specific characteristics of the RSGs. Other analyses are not sensitive to the specific design characteristics of the SGs, and the current analysis of record was evaluated and determined to remain valid. The licensee noted that the NRC approval of this revised safety analyses is not required since the changes are being evaluated under 10 CFR 50.59.

DCPP implemented the Steam Generator Replacement Program (SGRP) to replace the Westinghouse Model 51 original steam generator (OSG) with Westinghouse Model Delta 54 as the RSG. The licensee stated that since the OSG and RSG are similar, the SG replacement can be evaluated under 10 CFR 50.59. As noted above, the Chapter 15 safety analyses for the RSGs were performed using NRC-approved methods and have demonstrated compliance with applicable acceptance criteria and standards. The NRC requested additional information regarding the licensee’s conclusion that the RSG could be evaluated under 10 CFR 50.59. In response to the NRC staff’s request for additional information, the licensee, by letter dated September 28, 2007, provided a comparison table listing all key design and operating parameters for both OSG and RSG to demonstrate that the SGs are similar. Based on a review of this table, the NRC staff concluded that the RSGs are designed and will operate similar to the OSGs. The NRC staff has also reviewed the licensee’s 10 CFR 50.59 analyses regarding the SGRP, and as part of the inspection effort related to the SGRP, NRC Inspection Manual, Inspection Procedure (IP) 50001, states the NRC staff will:

1. Verify that selected design changes and modifications to systems, structures, and components (SSCs) described in the Final Safety Analysis Report (FSAR) are reviewed in accordance with 10 CFR 50.59.

Therefore, as part of the NRC inspection of the SGs at DCPP, the NRC staff will confirm that the 10 CFR 50.59 analyses is correctly applied to the SGRP. Based on the above, the NRC staff agrees that the SG replacement effort does not meet any of the criteria in 10 CFR 50.59, and therefore, the reanalysis of the SGs does not need NRC staff review and approval, assuming a satisfactory completion of the IP 50001 inspection, except for the Feedwater Isolation SG Water Level-High High (P-14) ESFAS setpoint which was changed.

3.2 Effect of Feedwater Isolation SG Water Level-High High (P-14) Change on Accident Analysis

The OSGs and the RSGs by Westinghouse have two-stage moisture separation. The first stage uses centrifugal separators, and the second stage uses chevron-type separators. A mid-deck divider plate separates the two stages. The SG Water Level (SGWL) instrumentation uses differential pressure instruments with several ranges: a wide-range non-safety-related instrument and three or four narrow-range safety-related instruments. The wide-range instrument spans the entire length of the downcomer region, while the narrow-range instruments span only the upper 25 percent of the wide-range to cover the normal operating band. The upper taps for all four instruments are located above the mid-deck plate, while the lower taps are all located below this plate.

In addition, the OSGs and the RSGs have holes in the mid-deck, which were designed to allow moisture removed from the second-stage separators to flow back into the downcomers, act as orifices that restrict steam flow and allow pressure differences with water levels below the mid-deck region. At higher steam flow rates with a decreasing SGWL, steam exiting the first stage separators along with the moisture being separated is enough to build up pressure below the plate that is not acting above the plate. Since the upper SGWL instrument taps are connected above the plate, a pressure difference acts on the four instruments and provides a bias that causes the instruments to indicate a higher-than-actual water level. For the limiting safety setting of SG low-low water level setpoint, this bias acts in a non-conservative direction. The magnitude of the bias drops as the steam flow decreases.

Westinghouse Nuclear Safety Advisory Letter 02-4 identified that, due to the void content of the two-phase mixture above the mid-deck plate, the SGWL instrument channel will not indicate water level as accurately as presumed above the mid-deck plate. As a result, an SG high-high level trip (P-14) may not occur even though the two-phase mixture level may in reality be above the upper level tap. Due to instrument channel saturation, water mass above the upper level tap will not be reflected in the level measurement. SGWL is determined by the differential pressure between a reference column of water at ambient containment conditions and a head of fluid in the SG sensed via the lower level tap. Both columns of fluid are connected via the upper level tap to result in a common pressure at the top of each fluid column. As the SGWL rises, the differential pressure across the level transmitter decreases. Since the SGWL is determined from the differential pressure across the transmitter, the maximum SG high-high level Safety Analysis Limit (SAL) is limited. The maximum SAL is limited to be a value less than that resulting from when there is the minimum differential pressure across the transmitter to reliably perform the trip function with voids present. Westinghouse refers to this minimum differential pressure limit as the maximum reliable indicated level (MRIL). The SG high-high level trip setpoint is determined based on utilization of the MRIL as the SAL. This setpoint value is then reduced to address instrumentation uncertainties and arrive at an NTSP. The SG high-high level NTSP is provided to protect against a feedwater malfunction that results in an uncontrolled increase in water level.

The SGWL narrow-range (NR) span of the OSGs is different from that of the RSGs due to an expanded NR span's being incorporated as part of the RSGs design. The existing SGs have an SGWL NR span of 144 inches, while RSGs have an SGWL NR span of 212 inches. The revised SGWL NR span of 212 inches has been incorporated into the UFSAR Chapter 15 safety analyses for the RSGs. The Feedwater Isolation SGWL-High High (P-14) function is credited in the analysis of the Excessive Heat Removal due to Feedwater System Malfunction event. A change in SG feedwater conditions resulting in an increased feedwater flow could result in excessive heat removal from the RCS. Due to an expanded transmitter span of 212 inches for RSGs versus 144 inches span of existing SGs and an increase in the nominal control level setpoint, an increase in the trip setpoint is necessary to provide sufficient operating margin from the nominal control point to the trip setpoint. Therefore, the SGWL-High High trip setpoint is raised from 75 percent of existing SGs to 90 percent for the RSGs. Based on the setpoint analysis for the Feedwater Isolation SGWL-High High (P-14) setpoint, the MRIL is 98.8 percent span, the NTSP is 90.0 percent, and the allowable value (AV) is less than or equal to 90.2 percent span. Thus, the licensee will revise SGWL-High High (P-14) setpoint from 75 percent to 90.0 percent, and AV from 75.2 percent to 90.2 percent. The NRC staff has reviewed these TS changes and concluded that they are acceptable.

The existing SGWL-Low Low function TS values represent lower water levels in the RSGs compared to the existing SGs. This is accommodated in the RSG design by the location of the lower NR tap, the configuration of the SG tube bundle, and the revised UFSAR Chapter 15 safety analyses. Therefore, the TS values for SGWL-Low Low NTSP and AV are unchanged and no TS changes are required for the SGWL-Low Low NTSP and AV for the RSGs.

3.3 Setpoint Calculations

The licensee used the setpoint methodology provided in WCAP-11082, "Westinghouse Setpoint Methodology for Protection Systems, Diablo Canyon Units 1 & 2, 24-Month Fuel Cycle Evaluation," Revision 6, for the proposed AV and NTSP changes for Function 5.b, Feedwater Isolation SG Water Level-High High (P-14), in Table 3.3.2-1. By letter dated December 2, 2004, this WCAP was approved by the NRC for DCPP by Amendment Nos. 178 and 180, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment Re: Revised Technical Specifications 3.3.1, 'Reactor Trip System (RTS) Instrumentation' and 3.3.2, 'Engineered Safety Features Actuation System (ESFAS) Instrumentation' (TAC Nos. MC0893 and MC0894)."

The licensee derived the NTSP for the feedwater isolation SGWL-High High function by deducting Total Allowance (TA) from the MRIL. The licensee calculated the MRIL from the SAL for the feedwater isolation SGWL-High High (P-14) function assumed in the safety analysis. The licensee calculated the TA by adding a Margin to Channel Statistical Analysis Allowance (CSA). The CSA is comprised of process effects and the instrument loop tolerances. The licensee used non-instrument effects such as process pressure variation and mid-deck plate pressure loss as process tolerances and treated them as biases and combined them algebraically. The licensee statistically combined the various instrument loop tolerances, such as the transmitter and the rack tolerances, which are independent and random, using the square-root-of-the-sum-of-the- square (SRSS) technique. The licensee derived Acceptable As-Left tolerance span around the instrument setpoint using the rack calibration accuracy only. The NRC RIS 2006-17 permits the use of SRSS for reference accuracy, measurement and test equipment (M&TE) accuracy, and readability uncertainties for the Acceptable As-Left tolerance. The NRC staff has reviewed the value of the Acceptable As-Left tolerance in Westinghouse Proprietary version of WCAP-11082 and finds it consistent with the Acceptable As-Found tolerance and the CSA and, therefore, acceptable.

The licensee used only rack drift of +0.2 percent of the span in calculating Acceptable As-Found tolerance. The industry practice permits Acceptable As-Found tolerance as SRSS for reference accuracy, M&TE, and rack drift. Furthermore, the licensee used the Acceptable As-Found tolerance as the tolerance to calculate the AV, adding it algebraically to the NTSP. Therefore, the NRC staff finds the proposed AV and NTSP in TS Table 3.3.2-1 for Function 5.b conservative and acceptable.

3.4 Plant Surveillance Test Procedures

The licensee stated that SRs 3.3.2.5 and 3.3.2.9 are performed for ESFAS Function 5.b using surveillance test procedures (STP) I-4-L5xx series procedures (i.e., STP I-4-L517, I-4-L518, I-4-L519, I-4-L527, I-4-L528, I-4-L529, I-4-L537, I-4-L538, I-4-L539, I-4-L547, I-4-L548, and I-4-L549) that are controlled under 10 CFR 50.59. SR 3.3.2.5 is for performance of the channel operational test and SR 3.3.2.9 is for the performance of the channel calibration.

By letter dated September 7, 2005, the NRC recommended the addition of the following two footnotes for verification of setpoint surveillance for instrument functions on which a safety limit has been placed:

Note 1: If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Note 2: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the [Limiting Trip Setpoint*, or a value that is more conservative than the Limiting Trip Setpoint]; otherwise, the channel shall be declared inoperable. The [Limiting Trip Setpoint] and the methodology** used to determine the [Limiting Trip Setpoint], the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in the UFSAR [or Bases] [or a document incorporated into the UFSAR such as the technical requirements manual].

*Reviewers Note: the words "Limiting Trip Setpoint" are generic terminology for the setpoint value calculated by means of the plant-specific setpoint methodology documented in the UFSAR, or Bases, or a document incorporated into the UFSAR such as the technical requirements manual. The nominal Trip Setpoint (field setting) may use a setting value that is more conservative than the Limiting Trip Setpoint, but for the purpose of TS compliance with 10 CFR 50.36, the plant-specific setpoint term for the Limiting Trip Setpoint must be cited in Note 2. The brackets indicate plant-specific terms may apply, as reviewed and approved by the NRC staff.

**The NRC staff will review and approve the methodology supporting the requested changes in the LAR.

The licensee, by letter dated September 28, 2007, addressed this issue by providing the following as Regulatory Commitments:

> In order to provide compliance with the proposed notes to Surveillance Requirements (SR) 3.3.2.5 and 3.3.2.9 for Engineered Safety Feature Actuation System (ESFAS) Function 5.b in TS Table 3.3.2-1, and the proposed changes to the Technical Specification (TS) 3.3.2 Bases for SR 3.3.2.5 and SR 3.3.2.9 for ESFAS Function 5.b, the 10 CFR 50.59 controlled surveillance test procedures applicable to ESFAS Function 5.b will be updated as required as part of implementation of the amendment for each unit. The Actions for the various potential surveillance outcomes will be required as follows:
>
> > The instrument channel setpoint exceeds the as-left tolerance but is within the as-found tolerance:
> >
> > - Reset the instrument channel setpoint to within the as-left tolerance;
> > - If the instrument channel setpoint cannot be reset to a value that is within the as-left tolerance around the instrument channel setpoint at the completion of the surveillance, if not already inoperable, the instrument channel shall be declared inoperable.
> >
> > The instrument channel setpoint exceeds the as-found tolerance but is conservative with respect to the TS Allowable Value (AV):
> >
> > - Reset the instrument channel setpoint to within the as-left tolerance;
> > - If the instrument channel setpoint cannot be reset to a value that is within the as-left tolerance around the instrument channel setpoint at the completion of the Surveillance, if not already inoperable, the instrument channel shall be declared inoperable;
> > - Enter the channel's as-found condition in the Corrective Action Program for prompt verification that the instrument is functioning as required and further evaluation. Evaluate the channel performance utilizing available information to verify that it is functioning as required before returning the channel to service. The evaluation may include an evaluation of magnitude of change per unit time, response of instrument for reset, previous history, etc., to provide confidence that the channel will perform its specified safety function;
> > - Document the condition for continued OPERABILITY.
> >
> > The instrument channel setpoint is non-conservative with respect to the TS AV:
> >
> > - If not already inoperable, declare the channel inoperable;

- Reset the instrument channel setpoint to within the as-left tolerance;

- Enter the channel's as-found condition in the Corrective Action Program for evaluation. Evaluate the channel performance utilizing available information to verify that it is functioning as required before returning the channel to service.

- The evaluation may include an evaluation of magnitude of change per unit time, response of instrument for reset, previous history, etc., to provide confidence that the channel will perform its specified safety function.

The NRC staff finds the above plant surveillance procedures comply with the NRC RIS 2006-17 and the September 7, 2005, letter from Patrick L. Hiland to NEI Setpoint Methods Task Force.

3.5 Footnotes for Safety Limit Related Functions

By letter dated August 9, 2007, the licensee proposed the addition of the following two footnotes to SR 3.3.2.5 and SR 3.3.2.9 in TS Table 3.3.2-1:

Footnote (d): If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. Footnote (a) does not apply to this function.

Footnote (e): The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in the Equipment Control Guidelines. Footnote (a) does not apply to this function.

The NRC staff finds the licensee's proposed footnotes together with the commitments made in Section 3.4 complies with the NRC's letter dated September 7, 2005, and are acceptable to the NRC staff.

3.6 TSTF-449

The licensee is proposing to delete the TS requirements associated with alternate tube repair criteria applicable to their original SGs. These requirements include performance criteria (in TS 5.5.9.b), tube repair criteria (in TS 5.5.9.c), tube inspection criteria (in TS 5.5.9.d), and reporting requirements (in TS 5.6.10). In addition, the licensee is proposing to modify its inspection requirements to adopt those requirements applicable to SGs with thermally treated Alloy 690 tubes (i.e., the material used in its RSGs).

The alternate tube repair criteria (including the associated performance criteria, inspection requirements, and reporting requirements) were developed for the licensee's OSGs. With the planned replacement of the OSGs, these alternate tube repair criteria are no longer needed. In addition, given the design differences between the OSGs and RSGs, these repair criteria are not applicable to the RSGs. As a result, the NRC staff concludes that deletion of these requirements are acceptable.

With respect to modifying the inspection requirements to replace the current requirements, which are applicable to plants with mill-annealed Alloy 600 tubes, with those inspection requirements applicable to plants with thermally treated Alloy 690 tubes, the NRC staff finds these proposed changes acceptable since the licensee's RSGs have thermally treated Alloy 690 tubes and the proposed changes are consistent with TSTF-449.

In summary, the NRC staff finds that the proposed changes to the SG TS requirements are acceptable since the resultant TSs are consistent with TSTF-449.

4.0 LIST OF REGULATORY COMMITMENTS

In addition to the commitments discussed in Section 3.4 of this safety evaluation, the licensee has also the made the following list of regulatory commitments with respect to its LAR. These commitments, identified in Enclosure 5 to the licensee's application dated January 11, 2007, and Enclosure 1 to its supplemental letter dated August 9, 2007, are as follows:

1. The TSTF-493 changes will be made to the remaining applicable RTS and ESFAS functions in a separate LAR that will be submitted after TSTF-493 is approved by the NRC.

2. PG&E will include the methodologies used to determine the as-found and the as-left tolerance (including the as-found and as-left tolerance values) in the Equipment Control Guidelines, which is a 10 CFR 50.59 controlled document.

5.0 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.

6.0 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 February 13, 2007 (72 FR 6787). 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.

7.0 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.

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