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

September 17, 2009

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

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NO. 2 - ISSUANCE OF EXIGENT AMENDMENT RE: REQUEST FOR ONE-TIME CHANGE TO TECHNICAL SPECIFICATION (TS) 3.7.1, "MAIN STEAM SAFETY VALVES (MSSVS)," TABLE 3.7.1-1, "MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE MSSVS" (TAC NO. ME2176)

Dear Mr. Conway:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 208 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant (DCPP), Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 3, 2009, as supplemented on September 8, 2009. The NRC staff has reviewed your request for an emergency amendment and concluded that the request does not meet the standard in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.91(a)(5) for emergency circumstances, but does meet the standard in 10 CFR 50.91(a)(6) for exigent circumstances.

The amendment revises DCPP, Unit No. 2 TS 3.7.1, "Main Steam Safety Valves (MSSVs)," Table 3.7.1-1, "Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable MSSVs." DCPP, Unit No. 2 currently has an MSSV inoperable (MS-2-RV-224) and per TS 3.7.1, Required Action A.1, is operating at approximately 80 percent rated thermal power (RTP). The amendment increases the Power Range Neutron Flux High setpoint in TS Table 3.7.1-1 from 87 percent RTP to 106 percent RTP with MS-2-RV-224 inoperable for the remainder of Cycle 15, which is scheduled to end in October 2009. Approval of this request will allow DCPP, Unit No. 2 to operate at 100 percent RTP for the remainder of Cycle 15 while in TS 3.7.1, Required Action A.1; then, the unit will be shut down for refueling and corrective action can be taken to repair or replace the inoperable MS-2-RV-224.

This amendment is being issued under exigent circumstances in accordance with 10 CFR 50.91(a)(6). The exigent circumstances and final no significant hazards considerations are addressed in Sections 4.0 and 5.0 of the enclosed Safety Evaluation.

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 No. 50-323

Enclosures:
1. Amendment No. 208 to DPR-82
2. Safety Evaluation

cc w/encl: Distribution via Listserv

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. 208
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 September 3, 2009, as supplemented on September 8, 2009, 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. 208, 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 30 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: September 17, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 208

TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NO. 50-323

Replace the following pages of the Facility Operating License No. 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 No. DPR-82

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
Page 3.7-2	Page 3.7-2

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

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

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

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

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

(2) <u>Technical Specifications (SSER 32, Section 8)* and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 208, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) <u>Initial Test Program (SSER 31, Section 4.4.1)</u>

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.

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*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.

Table 3.7.1-1 (page 1 of 1)
Maximum Allowable Power Range Neutron Flux High Setpoint With Inoperable MSSVs

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT %RTP
4	87* **
3	47*
2	29*

* Unless the reactor trip system breakers are in the open position.

** For Unit 2 Cycle 15 with only MS-2-RV-224 inoperable, a Maximum Allowable Power Range Neutron Flux High Setpoint of 106% RTP may be used. 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. 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.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNIT NO. 2

DOCKET NO. 50-323

1.0 INTRODUCTION

By application dated September 3, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092580627), as supplemented on September 8, 2009 (ADAMS Accession No. ML092530534), Pacific Gas and Electric Company (PG&E, the licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License No. DPR-82) for the Diablo Canyon Power Plant (DCPP), Unit No. 2.

The proposed amendment requests a one-time change to the DCPP, Unit No. 2 TS 3.7.1, "Main Steam Safety Valves (MSSVs)," Table 3.7.1-1, "Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable MSSVs." DCPP, Unit No. 2 currently has an MSSV inoperable (MS-2-RV-224) and per TS 3.7.1, Required Action A.1, is operating at approximately 80 percent rated thermal power (RTP). The amendment will increase the Power Range Neutron Flux High setpoint in Table 3.7.1-1 from 87 percent RTP to 106 percent RTP with MS-2-RV-224 inoperable for the remainder of Cycle 15, which is scheduled to end in October 2009. Approval of this request would allow DCPP, Unit No. 2 to operate at 100 percent RTP for the remainder of Cycle 15 while in TS 3.7.1, Required Action A.1; then, the unit will be shut down for refueling and corrective action can be taken to repair or replace the inoperable MS-2-RV-224. DCPP Unit No. 2 is operated at a power level below the Power Range Neutron Flux High Setpoint to ensure that nuclear instrumentation and channel uncertainties do not result in an inadvertent Power Neutron Flux High reactor trip signal.

The specific change proposed is to add a footnote to TS Table 3.7.1-1, as follows:

> For Unit 2 Cycle 15 with only MS-2-RV-224 inoperable, a Maximum Allowable Power Range Neutron Flux High Setpoint of 106% RTP may be used. 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. 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.

Pursuant to Section 50.91(a)(5) of Title 10 of the *Code of Federal Regulations* (10 CFR), the licensee requested that the proposed amendment be issued under emergency circumstances to allow DCPP, Unit No. 2 to operate at 100 percent RTP until the end of Cycle 15. The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed PG&E's request for an emergency amendment and concluded that the request does not meet the standard in 10 CFR 50.91(a)(5) for emergency circumstances, but does meet the standard in 10 CFR 50.91(a)(6) for exigent circumstances. A detailed explanation of the exigent circumstances of this issue is contained in Section 5.0 of this safety evaluation.

The supplemental letter dated September 8, 2009, 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 Tribune* newspaper, located in San Luis Obispo, California, on September 11 and 12, 2009.

The basis for PG&E's request is a thermal hydraulic analysis, performed in a manner largely consistent with the licensing basis analyses, which demonstrates an acceptable response to a postulated turbine trip/load rejection transient. This transient is the limiting transient for which the MSSVs provide pressure relief. The key feature of the degraded condition analysis is crediting the high neutron flux trip instead of the high pressurizer pressure trip, which will terminate the transient earlier. This difference analytically recaptures the safety margin lost by the assumption that the lowest setpoint MSSVs are out of service.

2.0 REGULATORY EVALUATION

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for removal of energy from the reactor coolant system (RCS) if the preferred heat sink, provided by the condenser and circulating water system, is not available.

Five MSSVs are located on each main steam header, outside of containment. The MSSVs must have sufficient capacity to limit the secondary system pressure to less than or equal to 110 percent of the steam generator (SG) design pressure. The design of the MSSVs includes staggered setpoints, as specified in TS 3.7.1, Table 3.7-2, so that only the needed valves will actuate. The staggered setpoints reduce the potential for valve chattering (rapid opening and closing) during a transient. The accident analysis assumes that five MSSVs per SG are operable to provide overpressure protection for design-basis transients that are initiated with the reactor at 102 percent of RTP. The unit is operated at a power level below the Power Range Neutron Flux Setpoint to ensure that nuclear instrumentation instrument and channel uncertainties do not result in an inadvertent Power Range Neutron Flux High reactor trip signal.

The NRC staff considered the following regulations and guidance in its review of the amendment request:

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities;"

- 10 CFR 50.36, "Technical specifications," which 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 50.36 further states, in part, that a limiting condition for operation (LCO) of a nuclear reactor must be established for each structure, system, or component that is part of the primary success path and which functions or actuates 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;

- 10 CFR 50.55a, "Codes and standards," which states in section (a)(2) that systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code specified in paragraphs(b) through (g) of that section;

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 10, "Reactor design," which requires that the reactor core and associated coolant, control and protection systems be designed with appropriate margin to ensure that specified acceptable fuel design limits [SAFDLs] are not exceeded during normal operations, including the effects of anticipated operational occurrences [AOOs];

- 10 CFR Part 50, Appendix A, Criterion 13, "Instrumentation and control," which 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 15, "Reactor coolant system design," which requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design condition of the RCPB are not exceeded during any condition of normal operation including AOO's;

- 10 CFR Part 50, Appendix A, Criterion 20, "Protection system functions," which requires that the protection system be designed to initiate operation of appropriate systems to ensure that SAFDLs are not exceeded;

- 10 CFR Part 50, Appendix A, Criterion 26, "Reactivity control system redundancy and capability," which requires, in part, that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to assure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded;

- NRC Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," dated December 1999, which 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 Instrument Society of America ISA-S67.04-1994, "Setpoints for Nuclear Safety Instrumentation," subject to the NRC staff clarifications;

- 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, which discusses the requirements of 10 CFR 50.36 related to limiting safety system settings (LSSS) 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 allowable value; and

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 15, "Transient and Accident Analysis," which describes the spectrum of transients and accidents to be analyzed for nuclear power plants and also provides review criteria for the NRC staff.

3.0 TECHNICAL EVALUATION

The MSSVs provide pressure relief that is essential to maintaining the RCPB within its ASME Code limit of 1210 pounds per square inch gauge (psig) under postulated load rejection or turbine trip conditions. Therefore, the MSSVs are part of the primary success path to mitigate a design-basis transient that, by degrading secondary plant heat transfer capabilities, presents a challenge to the integrity of the RCPB.

The NRC staff evaluated the licensee's request to ensure that the proposed modification to LCO 3.1.7.A.1 of the DCPP TS is consistent with 10 CFR 50.36 in that the degraded condition analysis (1) is performed acceptably, and (2) demonstrates that, even at the requested power level while in the degraded condition, the revised LCO preserves acceptable quality of the RCPB.

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events cause a sudden reduction in steam flow and result in pressurization events. Reactor protection and safety systems are actuated to mitigate these transients. The NRC staff's review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses.

A major loss of load can result from either a loss-of-external electrical load or from a turbine trip from full power without a direct reactor trip. These events result in a sudden reduction in steam flow. The loss of heat sink leads to pressurization of the RCS and Main Steam System (MSS). The AOO acceptance criteria applicable to this event are that (1) departure from nucleate boiling margin on the hot fuel pins is maintained, (2) pressure in the RCS and MSS are maintained below 11 percent of the design pressure values, and (3) the event does not develop into a more serious plant condition without the occurrence of another independent fault.

3.1 Summary of Technical Information Provided by the Licensee

The licensee's analyses of this event focused on the degraded condition; namely, that selected MSSVs are unavailable. Because of this, PG&E performed a limited analysis, using the licensing basis analyses for reference purposes. The licensing basis presents analyses of several cases; while some are designed to provide limiting results for primary system pressurization and departure from nucleate boiling, others are designed to yield the peak MSS pressure.

3.1.1 Description of Actual Condition

The degraded condition at DCPP, Unit No. 2 involves a faulty MSSV, MS-2-RV-224, which has been gagged closed. It has the highest specified lift setting of the five safety valves installed on its steam line. Therefore, during an actual transient demanding main steam pressure relief, this valve would normally open only under a pressurization scenario so severe that the opening of the four other relief valves would not provide adequate secondary system pressure relief.

3.1.2 Thermal Hydraulic Analytic Method

PG&E performed a loss of load analysis in a manner largely consistent with that described in the DCPP Final Safety Analysis Report (FSAR). The licensee employed conservative assumptions and used the RETRAN-02 systems computer code to perform the evaluation for MSS pressure boundary integrity. The program simulates the neutron kinetics, and the behavior of the RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, SG, and SG safety valves. The program computes pertinent variables, including temperatures, pressures, and power level.

The licensee noted that the MSSVs were modeled using an improved accumulation to full open. The licensing basis models of the reduction in secondary heat transfer transients model a 3 percent lift setpoint uncertainty and an additional 3 percent increase in accumulation pressure until the valves reach full open position. The analysis employs a modeling improvement that assumes a 5 pounds per square inch (psi) accumulation to full open, rather than 3 percent. The licensee stated that the adequacy of this modeling improvement has been demonstrated through extensive valve testing.

3.1.3 Description of Licensee's Thermal Hydraulic Model

The present evaluation focuses on those cases designed to yield the peak MSS pressure, since the unavailability of an MSSV would affect the MSS most significantly. The licensee also provided a disposition for the remaining cases to support a conclusion that the primary system response need not be re-analyzed for the degraded condition.

The licensee stated that the current licensing basis analysis evaluates several transients at each unit, and based on these evaluations, the DCPP, Unit No. 2 loss of load/turbine trip (LOL/TT) analysis assuming zero percent SG tube plugging is the limiting case for secondary pressurization. This is because Unit No. 2 has a slightly higher RCS temperature and the minimized SG tube plugging will maximize secondary heat transfer.

The licensee identified the following major assumptions regarding this event:

- RCS parameters (power, temperature, pressure) were assumed at nominal values consistent with steady-state, full-power operation. Core thermal power level was increased by 2 percent to account for calorimetric uncertainty.

- Only the four highest setpoint MSSVs are credited for pressure relief on the affected SG. The remaining SGs are modeled with full pressure relief capability.

- Neutronic design parameters were evaluated in two cases, one simulating beginning of core life (BOL) conditions with a positive moderator temperature coefficient, and one at end of core life (EOL) conditions with a more negative moderator temperature coefficient to assure that the high flux trip setpoint was appropriately determined to bound the remainder of the DCPP, Unit No. 2 fuel cycle.

- An additional case simulating an inadvertent main steam isolation valve (MSIV) closure was modeled to ensure that the limiting secondary pressurization transient remained the LOL/TT event.

- The transient is terminated by the high neutron flux trip. The licensing basis analyses, by contrast, credit the high pressurizer pressure trip, which is expected to occur after the receipt of the high neutron flux trip signal.

3.1.4 Results

The licensee's analysis demonstrates that the MSS pressure does not exceed 1210 psig at any point during the transient. The limiting case was the BOL analysis, which indicated a peak pressure of 1204.2 pounds per square inch absolute (psia). Neither the EOL case nor the MSIV closure case resulted in secondary pressures as high.

3.2 NRC Staff Evaluation

The NRC staff's review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses.

The sequence of events modeled for this transient is acceptable because the licensee modeled additional transient cases, including an inadvertent MSIV closure, to establish that the limiting secondary pressurization transient had been identified. Although the licensee did not model limiting departure from nucleate boiling ratio (DNBR) cases, the NRC staff concludes that the licensee's disposition is acceptable because the degraded secondary heat transfer condition caused by the inoperable MSSV would cause additional pressurization in the primary system, likely resulting in an increase in DNBR margin.

The NRC staff's acceptance of the presently reviewed analyses is based on two considerations: (1) the requested analysis supports full-power operation in a degraded condition for a reasonably small amount of time, and (2) RETRAN-02, whether applied in accordance with generically approved methods or used on an application-specific basis, has been accepted by the NRC previously for use in licensing applications at DCPP. The NRC staff did not, in its review of this license amendment request, establish whether the RETRAN modeling was performed in accordance with NRC-approved or otherwise acceptable methodology (i.e., WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis").

The parameters used in the model are acceptable to the NRC staff. The neutronic parameters are acceptable because both BOL and EOL conditions are modeled, and since the BOL condition is limiting and DCPP, Unit No. 2 is near the end of its cycle, the BOL case is conservative relative to the plant actual condition. The valve modeling parameters are acceptable because the licensee stated that the safety valve accumulation modeling is supported by extensive valve testing, and because the valve with the lowest setpoint is modeled to be out of service. This approach is conservative because the inoperable valve is the one with the highest lift setpoint, meaning the model provides less pressure relief capacity than actually exists currently at DCPP, Unit No. 2.

The NRC staff requested additional information concerning the licensee's selection of modeling parameters. The licensee chose to increase conservatively the core power level by 2 percent to provide thermal power uncertainty. Because the reactor trip being modeled is the high neutron flux trip, the NRC staff was concerned that, if initiated at a lower power level, the power ascension during the initial part of the load reject transient could cause greater pressurization.

In response to the NRC staff's concern, the licensee provided information about the differences between the BOL core neutronics that were modeled from the limiting case, and the EOL core neutronics that currently exist. The licensee provided sufficient information to demonstrate that, while the BOL case would increase in power during the loss of secondary heat transfer associated with the turbine trip, the EOL case, due to a strongly negative moderator temperature coefficient of reactivity, would actually result in a decrease in power and a trip on low pressurizer pressure. The information provided by the licensee demonstrates that increasing the core power level for the modeled case is a conservative treatment with regard to EOL conditions, and reaffirmed that the EOL case that was actually modeled was less severe than the BOL case. The NRC staff concludes that the licensee's supplemental information confirms that the parameters modeled in the licensee's analyses are acceptable.

The analytic results indicate acceptable performance at 100 percent power in the degraded condition. The limiting analysis result is 1204.2 psia, which is within the 1210 psia limit, and is, therefore, acceptable. As noted in the second paragraph of Section 3.2 of this safety evaluation, the licensee's decision not to model DNBR-limiting cases is acceptable.

Based on the above, the NRC staff concludes that: (1) the model was developed using approved/acceptable methodology, (2) the limiting transient sequence of events has been confirmed and modeled, and (3) the model was accomplished using conservative assumptions to yield a limiting pressure rise. The transient analysis demonstrates that full-power operation with the inoperable MSSV gagged closed is acceptable for the remainder of the current fuel cycle for Unit No. 2. Therefore, the licensee's results are acceptable.

The NRC staff notes that the TS Bases for Westinghouse plants indicate that reactivity and power distribution anomalies may also challenge the pressure relief capability of the MSS. For this reason, the NRC staff also reviewed DCPP's licensing basis analysis for the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power event. The NRC staff concludes that this transient is insignificant with respect to DCPP's current degraded condition, because in cases where the transient is terminated by a high neutron flux trip, the transient would be less severe, since the trip setpoint has been lowered to limit the peak achievable power increase prior to a reactor trip.

The NRC staff also reviewed the licensee's analysis of the setpoint methodology to be applied and the revised footnote to be added to TS Table 3.7.1-1. 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 determination of the RTP uncertainty for the Power Range Neutron Flux High Setpoint. By letter dated December 2, 2004, NRC approved the use of this WCAP for DCPP in Amendment Nos. 178 and 180, for Unit Nos. 1 and 2, respectively. This methodology (WCAP-11082) determined that a 6 percent RTP uncertainty should be applied for the Power Range Neutron Flux High Setpoints. The licensee arrived at the 106 percent Maximum Allowable Power Range Neutron Flux High Setpoint by reducing 6 percent RTP uncertainty from 112 RTP value from Case 1 analysis. The Case 1 analysis evaluates a loss of extended LOT/TT event with the inoperable MSSV assuming the most positive moderator temperature coefficient to determine the reduction in the Power Range Neutron Flux High Setpoint required to bound any potential overpower transient. The licensee stated that the 6 percent total uncertainty conforms to the guidance in RIS 2006-17. Furthermore, in its letter dated September 8, 2009, the licensee committed to calculate the required as-found and as-left tolerances in accordance with the guidance in RIS 2006-17 prior to the adjustment of the DCPP Unit 2 Maximum Allowable Power Range Neutron Flux High Setpoint from 87 percent to 106 percent during DCPP, Unit No. 2 Cycle 15.

The proposed footnote to TS Table 3.7.1-1 is similar to that for the engineered safety feature actuation system Steam Generator Water Level-High High Feedwater Isolation Nominal Trip Setpoint previously approved by the NRC staff in Amendment Nos. 198 and 199 for the DCPP, Unit Nos. 1 and 2, dated January 2, 2008. The additional footnote to be added by this amendment will ensure that the licensee will follow the guidance provided in RIS 2006-17 while operating with one inoperable MSSV for the remainder of Unit No. 2 Cycle 15.

3.3 Summary of NRC Staff Evaluation

The NRC staff has reviewed the licensee's analysis of the decrease in heat removal events, described above, for the degraded condition, and concludes that the licensee's analyses have adequately accounted for full-power operation of the plant with MS-2-RV-224 declared inoperable and gagged closed.

The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the RCPB pressure limits will not be exceeded as a result of these events while operating with the inoperable MSSV. The NRC staff concludes that the plant will continue to meet the requirements of 10 CFR 50.36 and 10 CFR Part 50, Appendix A, Criteria while operating within the revised LCO, which will apply only for the limited period of the remainder of Cycle 15 for Unit No. 2. Therefore, the NRC staff concludes that the requested TS change will allow facility operation with an acceptable level of quality of the main steam pressure relief system and the requested TS change is acceptable on an exigent basis.

The NRC staff concludes that the proposed license amendment request for a one-time change complies with the regulatory requirements specified in Section 2.0 of this safety evaluation, specifically the requirements in RIS 2006-17 and is, therefore, acceptable to the NRC staff.

The NRC staff concludes that the licensee's proposed revision to the footnotes in TS Table 3.7.1-1, to allow an increase in the Power Range Neutron Flux High Setpoint from 87 percent RTP to 106 percent RTP for DCPP, Unit No. 2, Cycle 15, with MSSV MS-2-RV-224 inoperable, together with the commitments made by the licensee, meets the regulatory requirements specified in Section 2.0 of this safety evaluation, and is acceptable.

4.0 PUBLIC COMMENT

During the comment period, the Commission received an e-mail from an individual with comments on the amendment request. The individual had a general comment regarding the proposed no significant hazards considerations (NSHC) finding and how the NRC staff makes that finding. The individual noted that the NRC staff needed to clarify the nature of valve failure in order for a citizen group to be able to offer meaningful comments. In addition, the individual had the following questions:

1. What is the history of the failed valve? Was it newly installed when the steam generators were replaced?

2. Does the history of the functioning valves match that of the failed valve?

3. How long had the failed valve and the other 19 valves been in use? What is the expected length of service?

4. Has the root cause of the valve failure been determined?

    a. Were the valves defective and, if so, will the manufacturer be held responsible? Who was the manufacturer?

    b. Have any other steam generators manufactured in Spain had similar problems? If so, how many?

    c. Has it been determined if PG&E and/or its contractors installed the valves incorrectly?

    d. If human error contributed to the failure, will the resulting expenses be charged to the steam generator project or will they be reflected in the next general rate case?

The approval of the amendment is based on the review of the licensee's analysis of the decrease in heat removal events, as previously discussed above, for the degraded condition. The NRC staff concluded that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the RCPB pressure limits will not be exceeded as a result of these events while operating with the inoperable MSSV. The reliability and failure mechanism of the MSSV are not pertinent to our approval of the amendment as the valve had failed. However, Questions 1 through 4 will be addressed as part of the corrective action program by the licensee. The corrective action program will include a root cause analysis. The root cause analysis will be performed following removal of the valve in October 2009. The results of the root cause analysis and any corrective actions to prevent recurrence will be documented in the associated Licensee Event Report (LER) to be submitted to the NRC within 60 days of the event. A supplemental LER may be necessary due to the time delay in completing a thorough root cause analysis and the inability to start the forensics on the valve until it is removed in October 2009. During the refueling outage, the licensee will determine if the valve can be repaired or needs to be replaced.

The NRC resident inspectors at DCPP plan to perform an "in-depth" follow-up of the failed valve per Inspection Procedure 7152, "Identification and Resolution of Problems." The results of this inspection are expected to be included in the fourth quarter DCPP Integrated Inspection Report. During the in-depth follow-up, the inspectors will:

- Review the maintenance history of the failed valve.

- Look for commonalities between the failed valve and the other steam generator safety valves.

- Determine how long the steam generator safety valves had been in use.

- Determine the expected service life of the valve.

- Review the root cause of the valve failure.

- Review the failure mechanism for applicability under 10 CFR Part 21 Notification requirements.

- Determine the valve manufacturer and review any vendor recommendations.

- Review industry operational experience related to failed steam generator safety valves.

- Review the maintenance procedures used on the failed valve.

- Determine if human error contributed to the failure.

5. Have there been similar cases at Diablo or other plants in which such a valve failed and such an amendment was granted? In other words, is this PG&E request in accordance with past practices?

The current TS Table 3.7.1-1 allows for the continued operation with up to three inoperable MSSVs at various reduced power levels and is consistent with the Standard Technical Specifications. While we do not have an example of an amendment for an MSSV failure, the NRC staff has many examples of where the pressure relief capacity was revised based on improved safety analysis. The licensee has stated that the plant has excess relief capacity due to conservatisms for which they had not previously taken credit for. The approval of this amendment was based on the demonstration of this excess relief capacity by the reanalysis of the decrease in heat removal events.

6. Or is this a unique circumstance and might it set a precedent for other plants or other incidents at Diablo?

This amendment is unique in the sense that not all plants have the excess relief capacity and could operate at 100 percent power with an inoperable MSSV. In addition, it should be noted that this amendment will be approved for the failure of a specific MSSV, having the highest specified lift setting, and for less than 4 weeks. This approval would set a precedent for another plant that can demonstrate the same conditions as the DCPP scenario (specific valve and limited time of approval).

7. On what date was the defective valve discovered? Does that date match the date of failure or was it after the fact?

The broken spring was found during valve testing on August 26, 2009, and the condition was entered into the DCPP Corrective Action Program on the same day. The root cause analysis will investigate a number of conditions including the time of service failure.

8. How many failed valves would it take to trigger a complete shut-down of the unit?

The current TS Table 3.7.1-1 would require a shutdown with four inoperable MSSVs on one SG.

9. The date for comments should be extended to a week past the date of PG&E's filing being made available to us. A mishandling of documents prevented the individual's organization from having access to the PG&E request for a license amendment until the very hour that comments on the same were due. The individual's request for a full week to review that request was denied.

The NRC published the NSHC in *The Tribune* on September 11 and 12, 2009. The NRC normally issues exigent amendments of this kind within three days of publication of the NSHC in local media. The NRC transmitted the licensee's submittals to the individual and allowed three days for additional comments beyond the date identified in *The Tribune*.

The questions raised above pertain, generally, to DCPP maintenance and NRC amendment review process and are not within the scope of the NRC staff's NSHC determination for the proposed amendment at DCPP. While the NRC staff did provide responses to these questions above, the questions and comments did not change the NRC staff's original proposed NSHC. As such, the NRC staff has made a final NSHC finding below in Section 6.0. In addition, the requested information did not provide concerns that would have caused the NRC staff to reconsider the conclusions in its technical review.

5.0 EXIGENT CIRCUMSTANCES

The Commission's regulations, 10 CFR 50.91, contain provisions for issuance of amendments when the usual 30-day public notice period cannot be met. Pursuant to 10 CFR 50.91(a)(5), the licensee requested that the proposed amendment be issued under emergency circumstances to allow DCPP, Unit No. 2 to operate at 100 percent RTP until the end of Cycle 15. The NRC staff has reviewed PG&E's request for an emergency amendment and concluded that the request does not meet the standard in 10 CFR 50.91(a)(5) for emergency circumstances, but does meet the standard 10 CFR 50.91(a)(6) for exigent circumstances. An exigency is a case where the staff and licensee need to act promptly.

DCPP, Unit No. 2 entered TS 3.7.1, Required Action A.1, at 12:45 p.m. on August 26, 2009, when the spring on MSSV MS-2-RV-224 in DCPP, Unit No. 2 was found broken. The valve was declared inoperable and power was reduced to approximately 80 percent RTP and the Power

Range Neutron Flux Setpoint was reduced to 87 percent RTP as required by TS Table 3.7.1-1. The unit is operated at a power level below the Power Range Neutron Flux Setpoint to ensure that nuclear instrumentation instrument and channel uncertainties do not result in an inadvertent Power Range Neutron Flux High reactor trip signal.

The broken spring was found during valve testing on August 26, 2009, and the condition was entered into the DCPP Corrective Action Program. MS-2-RV-224 was last replaced with a refurbished relief valve in March 1998. Testing is performed every other cycle, and is normally performed as pre-outage work. MS-2-RV-224 was last tested per TS Surveillance Requirement (SR) 3.7.1.1 in 2006 and the as-found and as-left setpoints were within specification with no adjustments required.

TS Table 3.7.1-2 specifies the lift setting for MS-2-RV-224 as 1115 psig plus 3 percent. MS-2-RV-224 has the highest specified lift setting of the five MSSVs per SG. During the August 26, 2009, testing, the valve initially lifted 7 percent low and was adjusted. In the next test, it lifted 4.5 percent low and was adjusted. In the third test, it lifted 2.5 percent low and was within the TS Table 3.7.1-2 limit, but lower than the as-left TS SR 3.7.1 lift setting tolerance of 1 percent, so it was further adjusted. The spring crack was found prior to performance of the as-left test. The valve is currently gagged closed as a conservative measure.

This amendment will allow an increase in power of DCPP, Unit No. 2 from approximately 80 percent RTP to 100 percent RTP for the remainder of Cycle 15 with MSSV MS-2-RV-224 inoperable. The NRC staff has concluded that exigent circumstances exist because the inoperable MSSV will prevent DCPP, Unit No. 2 from resumption of operation up to the plant's licensed power level for the remainder of Cycle 15, which ends in October 2009. In addition, the NRC staff concluded that the licensee did not abuse the exigent circumstances rule inasmuch as the licensee could not have predicted the failure of the MSSV sooner; the licensee made efforts to restore the MSSV lift settings promptly upon discovering the problem, without success; and the licensee then determined that the MSSV was inoperable. Further, the licensee promptly filed the amendment request upon finding that the MSSV was not repairable, and thus acted in a timely manner in seeking the amendment. These circumstances warrant the issuance of the requested amendment prior to conclusion of the 30-day period for public comment as an exigent circumstances amendment.

The NRC staff published a public notice of the proposed action in *The Tribune* newspaper, located in San Luis Obispo, California, on September 11 and 12, 2009. In that notice, the staff made a proposed determination that the requested action does not involve a significant hazards consideration. The staff's consideration of public comments is discussed in Section 4.0 of this safety evaluation.

6.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in the margin of safety.

As required by 10 CFR 50.91(a), PG&E provided its analysis of the issue of no significant hazards consideration in its letter dated September 8, 2009, as presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This License Amendment Request (LAR) proposes a one-time change to Technical Specification (TS) 3.7.1, "Main Steam Safety Valves (MSSVs)," Table 3.7.1-1, "Maximum Allowable Power Range Neutron Flux High Setpoint With Inoperable MSSVs" to allow an increase in the Power Range Neutron Flux High setpoint from 87 percent rated thermal power (RTP) to 106 percent RTP for Unit 2 Cycle 15 with only Unit 2 main steam relief valve 224 (MS-2-RV-224) inoperable. In addition, the LAR proposed change revises and clarifies the surveillance requirements for the Power Range Neutron Flux High Setpoint during Unit 2 Cycle 15.

The increase in the Power Range Neutron Flux High setpoint TS value does not initiate an accident. Technician adjustments to lower the Power Range Neutron Flux High setpoint could cause a reactor trip; however, this action is already a TS requirement. Thus, increasing the TS setpoint value from the current value will not change the requirement for a technician to adjust the setpoints downward when MSSVs become inoperable, and therefore, will not increase the probability of a reactor trip.

The revision and clarification of the surveillance requirements for the Power Range Neutron Flux High setpoint ensure that this function will actuate as assumed in the safety analyses.

With the increase in the Power Range Neutron Flux High setpoint with only MS-2-RV-224 inoperable during Unit 2 Cycle 15 the remaining MSSVs will continue to prevent overpressure of the main steam [lines] and Steam Generators (SGs), and remove adequate heat from the reactor coolant system (RCS).

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The increase in the Power Range Neutron Flux High setpoint TS value with only MS-2-RV-224 inoperable during Unit 2 Cycle 15 does not initiate an accident and does not change the method by which any safety-related system performs the function.

The revision and clarification of the surveillance requirements for the Power Range Neutron Flux High setpoint will provide assurance that the plant will operate within the limits assumed in the safety analyses.

The proposed change does not result in plant operation outside the limits previously considered, nor allow the progression of transients or accidents in a manner different than previously considered.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The RCS pressure boundary is applicable to the proposed change. With the proposed change all relevant event acceptance criteria were found to be satisfied. Therefore, the proposed change does not involve a reduction in a margin of safety.

With the proposed change, the MSSVs will still prevent SG pressure from exceeding 110 percent of SG design pressure in accordance with the ASME code. The conclusions for the Final Safety Analysis Report accident analyses are unaffected by the change, remain valid, and provide margin.

The instrument surveillance requirement changes for the Power Range Neutron Flux High setpoint ensure that the instrumentation will actuate as assumed in the safety analysis.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on that review, the staff concludes that the amendment meets the three criteria of 10 CFR 50.92. Therefore, the NRC staff has made a final determination that the amendment does not involve a significant hazards consideration.

7.0 REGULATORY COMMITMENT

The licensee, in its letter dated September 8, 2009, made the following regulatory commitment with respect to its licensing amendment request. This commitment was identified in Attachment 4 of this letter.

> PG&E will calculate the required as-found and as-left tolerances in accordance with the guidance of RIS 2006-17 prior to the adjustment of the DCPP Unit 2 Maximum Allowable Power Range Neutron Flux High Setpoint from 87 percent to 106 percent during DCPP Unit 2 Cycle 15.

8.0 STATE CONSULTATION

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

9.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final finding that the amendment involves no significant hazards consideration. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

10.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) involve a significant increase in the probability or consequences of an accident previously evaluated, or (b) create the possibility of a new or different kind of accident from any previously evaluated, or (c) involve a significant reduction in a margin of safety and therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (3) such activities will be conducted in compliance with the Commission's regulations; and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: B. Parks
S. Mazumdar

Date: September 17, 2009

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 No. 50-323

Enclosures:
1. Amendment No. 208 to DPR-82
2. Safety Evaluation

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