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

February 19, 2016

Mr. Edward D. Halpin
Senior Vice President and
Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56, Mail Code 104/6
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING INCORPORATION INTO LICENSING BASIS OF PRESSURIZER FILLING ANALYSIS FOR A MAJOR RUPTURE OF A MAIN FEEDWATER PIPE ACCIDENT (CAC NOS. MF5785 AND MF5786)

Dear Mr. Halpin:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 223 to Facility Operating License No. DPR-80 and Amendment No. 225 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant (DCPP), Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Updated Final Safety Analysis Report (UFSAR) in response to Pacific Gas and Electric Company's (PG&E's) application dated February 25, 2015, as supplemented by letter dated July 8, 2015.

The amendments incorporate into the licensing basis an analysis of pressurizer filling concerns associated with the main feedwater pipe rupture accident summarized in DCPP UFSAR Section 15.4.2.2. The amendments involve the addition of time-critical operator actions and modifications of the PG&E Design Class I backup nitrogen accumulators, which are credited in the new pressurizer filling analysis.

Enclosure 3 to this letter contains Proprietary Information. When separated from Enclosure 3, this letter is DECONTROLLED.

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,

Siva P. Lingam

Siva P. Lingam, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosures:
1. Amendment No. 223 to DPR-80
2. Amendment No. 225 to DPR-82
3. Safety Evaluation (Proprietary)
4. Safety Evaluation (Non-Proprietary)

cc w/encls 1, 2 and 4: Distribution via Listserv

ENCLOSURE 1

AMENDMENT NO. 223 TO FACILITY OPERATING LICENSE NO. DPR-80

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNIT NO. 1

DOCKET NO. 50-275

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

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 223
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 (PG&E, the licensee), dated February 25, 2015, as supplemented by letter dated July 8, 2015, 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. 223, 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 90 days following PG&E implementation of Design Class I backup nitrogen accumulator modifications, planned for the Unit 2 19th refueling outage 2R19. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report as described in the licensee's letter dated February 25, 2015.

FOR THE NUCLEAR REGULATORY COMMISSION

R. Pascarelli

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
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: February 19, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 223

TO FACILITY OPERATING LICENSE NO. DPR-80

DOCKET NO. 50-275

Replace the following page of the Facility Operating License No. DPR-80 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Facility Operating License No. DPR-80

REMOVE	INSERT
-3-	-3-

(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. 223 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;

ENCLOSURE 2

AMENDMENT NO. 225 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNIT NO. 2

DOCKET NO. 50-323

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. 225
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 (PG&E, the licensee), dated February 25, 2015, as supplemented by letter dated July 8, 2015, 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. 225, 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 90 days following PG&E implementation of Design Class I backup nitrogen accumulator modifications, planned for the Unit 2 19th refueling outage 2R19. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report as described in the licensee's letter dated February 25, 2015.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
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: February 19, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 225

TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NO. 50-323

Replace the following page of the Facility Operating License No. DPR-82 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Facility Operating License No. DPR-82

REMOVE	INSERT
-3-	-3-

(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. 225, 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.

ENCLOSURE 4
(NON-PROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 223 AND 225 TO

FACILITY OPERATING LICENSE NOS. DPR-80 AND DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-275 AND 50-323

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 223 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 225 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letter dated February 25, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15056A773), as supplemented by letter dated July 8, 2015 (ADAMS Accession No. ML15190A048), Pacific Gas and Electric Company (PG&E, the licensee) submitted a license amendment request (LAR), which proposes to incorporate into the licensing basis an analysis of pressurizer filling concerns. These filling concerns are associated with the main feedwater pipe rupture accident that was summarized in Diablo Canyon Power Plant (DCPP), Unit Nos. 1 and 2 Updated Final Safety Analysis Report (UFSAR) Section 15.4.2.2. Portions of the letter dated July 8, 2015, contain sensitive unclassified non-safeguards information and have been withheld from public disclosure pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* (10 CFR). The proposed amendment involves the addition of time critical operator actions (TCOAs) and modifications of the PG&E Design Class I backup nitrogen accumulators, which are credited in the new pressurizer filling analysis. Included within the DCPP LAR was a request for the U.S. Nuclear Regulatory Commission (NRC) to approve four changes associated with TCOAs.

The supplemental letter dated July 8, 2015, 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 April 28, 2015 (80 FR 23605).

1.1 Background

Under certain accident conditions, such as those experienced following a feedwater line break (FLB) event, it is predicted that the pressurizer could reach a water-solid (filled) condition. If this occurs, water relief through the pressurizer safety valves (PSVs) may also occur. Water relief through the PSVs could lead to subsequent failure of the PSVs to reseat, resulting in an unisolable breach of the reactor coolant pressure boundary. The current licensing basis provides an analysis of the pressurizer filling condition during a spurious safety injection (SSI) event, which includes credited TCOAs. However, as discussed in the UFSAR, the current licensing basis does not include analysis of the pressurizer filling scenario specific to the FLB event and credits the SSI pressurizer filling analysis as the limiting scenario. Hence there was no verification that the operator response times and backup nitrogen capacities established for the SSI event would be adequate to prevent water relief through the PSVs during an FLB event. The proposed amendment would revise UFSAR Section 15.4.2.2 to incorporate the results of a bounding pressurizer filling analysis and credited TCOAs specific to the FLB pressurizer filling event.

1.2 Reason for Proposed Changes

In 1998, Westinghouse Electric Company LLC (Westinghouse) issued a notification regarding new information on a key assumption made in its previous 1988 analyses of PSV water relief capability. Specifically, Westinghouse determined the water temperature must remain above 613 degrees Fahrenheit (°F) to justify stable PSV operation.

Based on the new information, the licensee determined that during an SSI event the PSVs could relieve water at a low enough temperature that they might not properly reseat. This could potentially create a small break loss-of-coolant accident (SBLOCA). This is contrary to UFSAR Section 15.2, which requires that a Condition II fault (such as an SSI event) does not progress to cause a more serious Condition III or IV fault (such as SBLOCA).

In response to the Westinghouse notification in 1998, the licensee submitted an LAR dated September 24, 2002, "License Amendment Request (LAR) 01-08 Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves [PORVs]; Pressurizer Safety Valve Loop seal Temperature" (ADAMS Accession No. ML031290380), that proposed a change to credit automatic actuation of the PG&E Design Class I PORVs for response to a pressurizer filling scenario during an SSI event instead of crediting the PSVs. This change was supported by the upgrade of the PORV pneumatic operator to Instrument Class IA and the automatic control circuitry to Class IE. A new TCOA was introduced in the LAR for operators to ensure a PORV is available (check/open a PORV block valve) within 11 minutes of event initiation. By letter dated July 2, 2004, the NRC approved this LAR and concluded that automatic actuation of the PG&E Design Class I PORVs may be credited for mitigation of the SSI event, and that the TCOAs defined for mitigation of the SSI event were acceptable (ADAMS Accession No. ML041950260).

In Inspection Report dated July 23, 2010, "Diablo Canyon Power Plant – NRC Component Design Bases Inspection Report 05000275/2010007 and 05000323/2010007" (ADAMS Accession No. ML102040823), the NRC issued a Severity Level IV non-cited violation (NCV) to the licensee for failing to update the UFSAR in accordance with 10 CFR 50.71(e). Specifically, the licensee did not update UFSAR Section 15.4.2.2.2 to reflect the 1998 changes to the original Westinghouse analysis. The need to revise UFSAR Section 15.4.2 for this issue had previously been identified by Westinghouse and PG&E. Although appropriate changes had been made for the SSI analysis in UFSAR Section 15.2, changes had not been made for the FLB analysis as of 2010 when the NRC identified the violation during the Component Design

Basis Inspection. The violation was entered into the DCPP corrective action program. The licensee submitted this LAR to resolve the nonconformance identified by the NCV, by updating the FLB pressurizer filling analysis in the UFSAR to address water relief through the PSVs.

The proposed license amendment would revise DCPP UFSAR Chapter 15, Section 15.4.2 to incorporate an analysis of pressurizer filling concerns associated with the main feedwater pipe rupture accident. The LAR proposes to add Section 15.4.2.4, "Major Rupture of a Main Feedwater Pipe for Pressurizer Filling," with the following subsections: Acceptance Criteria, Identification of Causes and Accident Descriptions, Analysis of Effects and Consequences, Results, and Conclusion. Additionally, the LAR proposes to add new TCOAs credited in the FLB pressurizer filling analysis and to update other sections of the UFSAR as appropriate to reference the new analysis. The proposed amendment also changes Technical Specification (TS) Bases B3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," to add that the PORVs mitigate the consequences of a pressurizer filling condition during the FLB event, in addition to the SSI event.

In addition, the proposed amendment resolves an issue identified by the Licensing Basis Verification Project during preparation of the UFSAR change to address the NCV. The Licensing Basis Verification Project discovered that TCOAs established for an SSI pressurizer filling event are non-conservative for responding to the FLB accident due to the differences in overpressure characteristics between the two events. Ensuring that conservative choices and assumptions are consistently made during an analysis helps provide reasonable assurance by creating greater safety margins within the analysis. The licensee concluded the UFSAR does not provide sufficient requirements to prevent the PSVs from relieving water during an FLB, because the SSI TCOAs are currently credited for responding to an FLB pressurizer filling event (UFSAR Section 15.4.2.2.3) without verification that these actions can be completed in time to prevent PSVs from relieving water during an FLB. This nonconforming condition was entered into the DCPP corrective action program. This amendment proposes to resolve the nonconforming condition by establishing TCOAs specific to an FLB pressurizer filling event that prevent the PSVs from relieving water.

2.0 REGULATORY EVALUATION

The NRC staff considered the following regulatory requirements and guidance in its review of the current LAR:

- DCPP conforms to 10 CFR, Part 50, proposed Appendix A, General Design Criteria (GDC), which were published on July 11, 1967 (32 FR 10213). The GDCs applicable to this LAR include:

  - GDC 9, 1967 - Reactor Coolant Pressure Boundary (Category A)

    The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

- GDC 33, 1967 - Reactor Coolant Pressure Boundary Capability (Category A)

  The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

- GDC 43, 1967 - Accident Aggravation Prevention (Category A)

  Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

• DCPP Unit 1 Operating License Condition 2.C(6)(f) requires compliance with Section II.D.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980 (ADAMS Accession No. ML051400209), which requires nuclear power generation facilities to develop and execute programs that qualify reactor coolant system (RCS) relief and safety valves under expected operating conditions for design-basis transients and accidents. The qualification is required to include associated control circuitry, piping, and supports, as well as the valves themselves.

• Paragraph 10 CFR 50.34(b) specifies content requirements for the UFSAR including evaluations required to show that safety functions will be accomplished. In addition, 10 CFR 50.71(e) requires periodic updates of the FSAR to ensure the information included in the report contains the latest information developed.

• Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," Criterion 19 - Control room, states, in part, that

  A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.... Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

- 10 CFR 50.120, "Training and qualification of nuclear power plant personnel."

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition:"

  - Chapter 13 addresses "Conduct of Operations"; specific sub-chapters considered in this review were Chapters 13.2.1, "Reactor Operator Requalification Program; Reactor Operator Training," Rev. 3 (ADAMS Accession No. ML070100636), and 13.5.2.1, "Operating and Emergency Operating Procedures," Rev. 2 (ADAMS Accession No. ML070100635).

  - Chapter 18, Rev. 2, provides review guidance for "Human Factors Engineering" (ADAMS Accession No. ML070670253).

- NUREG-1764, Revision 1, "Guidance for the Review of Changes to Human Actions;" September 2007 (ADAMS Accession No. ML072640413).

- NRC Generic Letter No. 82-33, "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability," dated December 17, 1982 (ADAMS Accession No. ML031080548).

- NUREG-0700, Revision 2, "Human-System Interface Design Review Guidelines," May 2002 (ADAMS Accession No. ML021700373).

- NUREG-0711, Revision 3, "Human Factors Engineering Program Review Model," November 2012 (ADAMS Accession No. ML12324A013).

- NRC Information Notice No. 97-78, "Crediting Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," dated October 23, 1997 (ADAMS Accession No. ML031050065).

In accordance with the generic risk categories established in Appendix A, "Generic Human Actions that are Risk-Important," to NUREG-1764, Revision 1, the tasks under review are actions involved in the safety injection (SI) sequence, are actions performed during shutdown, and are actions involving risk-important systems, and are, therefore, considered "risk-important." Because of its estimated risk importance, the NRC staff performed a "Level One" review (i.e., the most stringent of the graded reviews possible under the guidance of NUREG-1764).

3.0 TECHNICAL EVALUATION

3.1 Current Analysis

The current UFSAR (Revision 22) Section 15.4.2.2.3 states that the FLB analysis does not address pressurizer filling during an FLB because (a) it is evaluated in Section 15.2.15.3 for the SSI pressurizer filling event, (b) pressurizer filling concerns during FLB were generically dispositioned by Westinghouse [WCAP-11677, "Pressurizer Safety Relief Valve Operation for Water Discharge During Feedwater Line Break," January 1988], and (c) operator action is credited to preclude water relief by the PSVs [based on actions responding to an SSI event in Section 15.2.15.3]. Section 15.4.2.2.4.1 also states that pressurizer overfill does not require specific evaluation for FLB. As discussed above in Section 1.2, the TCOAs based on the SSI analysis were found to be non-bounding for the FLB accident. In response to this discovery, PG&E commissioned Westinghouse to perform a pressurizer filling analysis specific to the FLB event. The new analysis credits the PG&E Design Class I pressurizer PORVs for pressure relief during the FLB event, credits modifications to the dedicated PG&E Design Class I backup nitrogen system, which the licensee is planning to perform, and calculates the limiting time operators have to perform the credited actions in order to prevent water relief through the PSVs.

3.2 New Analysis

In order to demonstrate that TCOAs can ensure that during an FLB event water is not relieved through the PSVs, resulting in an unisolable break, the licensee performed a new bounding analysis. The results of the analysis provide bounding conditions that are used to define the appropriate TCOAs and equipment design/response that mitigate the consequences of the event. The new analysis showed that water relief through the PSVs is prevented assuming the following TCOAs: (1) ensure a PORV is available within 8.6 minutes, (2) isolate the faulted steam generator (SG) within 10 minutes, (3) isolate charging flow within 25 minutes, and (4) stop reactor coolant pump (RCP) seal injection flow within 45 minutes. The new analysis credits the PG&E Design Class I pressurizer PORVs for pressure relief during the accident, credits modifications to the dedicated PG&E Design Class I backup nitrogen system, which the licensee is planning to perform, and calculates the maximum time operators have to ensure a PORV is available to prevent water relief through the PSVs.

In addition, the analysis calculates that the maximum number of PORV open and close cycles with either steam or water that could occur before termination of the FLB event assuming TCOAs are taken is 295. Increasing the nitrogen supply to accommodate cycling the PORVs for at least 300 cycles will ensure sufficient time for operators to terminate an FLB pressurizer filling event before water relief through the PSVs occurs. Modifications to the dedicated PG&E Design Class I backup nitrogen system are planned to provide the pressurizer PORVs the increased capacity to function for the number of cycles calculated. The modifications are scheduled to be completed during the Unit 1 and Unit 2 19th refueling outages (1R19 and 2R19, respectively). The LAR does not request approval of the modifications to the backup nitrogen supply system, which has been performed without prior NRC approval per 10 CFR 50.59.

3.2.1 FLB Transient Description

The main feedwater system is the primary means of removing heat from the RCS through the SGs. The FLB transient, addressed in UFSAR Section 15.4.2.2, is defined as a break in the main feedwater line located between a main feedwater isolation valve and its associated SG, downstream of the last check valve. Following an FLB accident, secondary water level decreases in the SGs. Steam generator inventory is lost through the break until auxiliary feedwater (AFW) flow is initiated, after which level will begin to recover in the SGs being fed with AFW (i.e., the intact SGs). Depending on the AFW flow available, there is the potential for an increase in RCS temperatures in the early part of the post-trip transient, along with an increase in RCS volume due to thermal expansion. Following initiation of the FLB accident, a low steam line pressure setpoint will be reached in the faulted loop, causing actuation of the SI signal and consequently starts the two PG&E Design Class I charging pumps. The RCS inventory addition from the charging pumps and RCS thermal expansion contributes to pressurizer filling. The SI signal also results in isolation of instrument air from containment thus requiring use of the PG&E Design Class I backup nitrogen system in order to make use of the PORVs.

The RCS heatup is eventually mitigated by the combined effects of the AFW flow to the intact SGs and charging flow to the RCS. If pressurizer filling occurs, the PORVs are available to relieve liquid inventory from the RCS, as long as an air supply is available from instrument air to containment or from the PG&E Design Class I backup nitrogen accumulators. The TSs define a PORV as operable even if its block valve is closed (provided the PORV is otherwise functional), so operators may need to take action to open the block valve to enable the PORV to provide water relief. In addition, operator actions are taken to control pressurizer level by isolating charging flow, establishing instrument air to containment, and establishing normal RCS letdown. Operating procedures will be revised to direct operators to cycle the PG&E Design Class I charging pumps off and on as necessary to control pressurizer level (while maintaining RCP seal cooling) if letdown cannot be established. This eliminates reliance on PG&E Design Class II systems (instrument air and RCS letdown). Once the pressurizer level is reduced and controlled or the pressurizer pressure is lower than the PSV setpoint, the potential for water relief through the PSVs is eliminated and the FLB pressurizer filling event is terminated.

3.2.2 FLB Pressurizer Filling Analysis

The purpose of the FLB pressurizer filling analysis is to demonstrate that if pressurizer filling occurs during an FLB event, the reactor coolant pressure boundary will be maintained through operator actions and equipment design/response that mitigate the consequences of the event before water relief through the PSVs occurs. Through assumptions and selection of critical limiting parameters, the new analysis minimizes the calculated time to pressurizer filling and maximizes the number of PORV cycles required to provide RCS pressure relief until the FLB filling event is terminated. The minimum time to pressurizer filling and the subsequent first PSV lift are used to specify the maximum time operators have to perform required operations (i.e., ensure a PORV is available, isolate faulted SG, etc.) and since these operator actions are credited in the analysis, they are designated as TCOAs. The licensee stated that while the new analysis uses the same computer code used for the analysis of record, the input includes different assumptions selected to minimize pressurizer fill time and maximize PG&E Design Class I pressurizer PORV cycles.

There are four TCOAs credited in the analysis of the FLB accident, as described below.

- Ensure a PORV is available - If no pressurizer PORV relief is available at the start of the FLB accident (e.g., due to a failure of one of the two PG&E Design Class I PORVs and with the other isolated by the block valve), the operators will ensure a PORV is available in time to prevent water relief through the PSVs. This analysis determines the time when this TCOA must occur.

- Isolate the faulted SG - Similar to the FLB analysis described in Section 15.4.2.2 of the DCPP UFSAR, this analysis models the existing TCOA for isolating the faulted SG within 10 minutes after reaching the low-low SG water level setpoint to direct all available AFW flow to the intact SGs. The 10-minute time is used as an input assumption in the new analysis.

- Isolate charging flow - Following initiation of the FLB accident, a low steam line pressure setpoint will be reached in the faulted loop, causing an SI signal to be generated and SI flow initiation to occur. The SI flow results in a reactor coolant inventory addition during the transient. To reduce the inventory addition, the operators will take action to isolate charging flow, which stops all flow from the charging pumps except RCP seal injection flow. This analysis determines the time when this TCOA must occur.

- Stop RCP seal injection flow - Following the TCOA to isolate charging flow, the only remaining source of addition to the reactor coolant inventory is RCP seal injection flow. To mitigate this remaining inventory addition, the operators will take action to reset Phase B containment isolation, restore component cooling water flow to the RCPs and stop the RCP seal injection flow. This analysis determines the time when this TCOA must occur.

The licensee stated that the system response following an FLB accident was calculated with a detailed simulation of the plant in accordance with the NRC-approved methodology for a 4-loop plant using the Westinghouse version of the RETRAN-02 computer code (RETRAN-02W). The NRC's generic approval of the Westinghouse methodology is documented in the safety evaluation report (SER) dated February 11, 1999, which is included in WCAP-14882-P-A, RETRAN 02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA [loss-of-coolant accident] Safety Analyses, April 1999 (not publicly available, proprietary). The SER concluded that the use of the methodology as described in

WCAP-14882-P-A is acceptable provided three conditions are met. These conditions and how they are met are as follows:

- The transients and accidents RETRAN-02W approved for use are listed in Table 1-1 of WCAP-14882-P-A.

  This condition is met because feedwater line rupture is included in the list.

- The WCAP applies to Westinghouse-designed 2, 3, and 4-loop plants.

  This condition is met because DCPP Units 1 and 2 are Westinghouse-designed 4-loop plants.

- Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis.

  The licensee provided an acceptable source and justification for the input data used in the analysis.

The new analysis models a simultaneous loss of main feedwater to all SGs and subsequent reverse blowdown of the faulted SG. The RETRAN-02W code simulates core neutron kinetics, the RCS (with a multi-node vessel model and individual reactor coolant loops), the pressurizer and PSVs, the individual SGs, and the main steam safety valves. The model also simulates various features of the reactor trip system, engineered safety features actuation system, and plant control systems. The code computes pertinent variables, including power level, pressurizer pressure, pressurizer water volume, reactor coolant temperatures, SG water level, and core decay heat.

The relief valve flow discussion in the February 11, 1999, NRC SER to WCAP-14882-P-A states, in part, that "The calculation of critical flow for subcooled or saturated conditions is not a significant consideration for the non-LOCA transients and accidents to be analyzed by Westinghouse using RETRAN." However, in the case of the current pressurizer filling analysis, the flow through the PORV is a significant consideration, because it affects the number of PORV cycles. Given that the number of PORV cycles was not considered in the generic approval of RETRAN-02W, the staff questioned, in a request for adeditional information (RAI) dated June 4, 2015 (ADAMS Accession No. ML15155B365), SRXB-RAI-1, the applicability of Westinghouse's existing NRC-approved non-LOCA safety analysis methodology in computing the PORV mass flow rate and number of PORV cycles.

The licensee responded, in a letter dated July 8, 2015, that WCAP-14882-P-A includes a provision for using RETRAN-02W to calculate flow rate through the pressurizer PORVs under liquid relief conditions. WCAP-14882-P-A, Section 3.10.2, "Pressurizer Pressure Control," states the following regarding the PORV modeling for water relief:

> **[[**
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> **]]**

Calculation of the **[[**

]] is used to account for the change in mass flow rate that occurs when water (instead of steam) is relieved through the valves. The licensee also states that because WCAP-14882-P-A includes a provision for calculating flow rate through the PORVs under liquid relief conditions, the existing NRC-approved Westinghouse non-LOCA safety analysis methodology is applicable for computing the PORV mass flow rate and number of PORV cycles for the FLB pressurizer filling analysis. In addition, by letter dated February 28, 2005 (ADAMS Accession No. ML050140453), the NRC previously approved the use of RETRAN-02W for the Seabrook Station SSI pressurizer-filling analysis as described in Section 4.2 of the subject LAR. Based on an evaluation of the licensee's reasoning and a review of the previous approval for Seabrrok Station, the NRC staff found that the use of RETRAN-02W is acceptable for the licensee's current calculation of critical flow for subcooled or saturated conditions through the PORVs for the FLB transient. In addition, additional conservatisms are used in computation of the PORV flow as discussed below.

The licensee stated that the pressurizer PORV relief flow is controlled by the choking velocity. As discussed in Section 3.5.2 of WCAP-14882-P-A, the valve flow area is based on the **[[**

]] In SRXB-RAI-2 dated June 4, 2015, the NRC staff questioned if the flow rate and associated critical flow model are conservative for both operator action time(s) and number of pressurizer PORV cycles.

In its RAI response July 8, 2015, the licensee stated that the PORV flow rate and associated critical flow model used in the FLB pressurizer filling analysis are conservative for both operator action times and number of pressurizer PORV cycles, and are consistent with WCAP-14882-P-A. Appendix B of the WCAP includes, "Westinghouse Responses to NRC Requests for Additional Information." In WCAP-14882-P-A, Appendix B, letter NSD-NRC-98-5765 dated August 26, 1998, the response to Question 7 provided background information that is relevant. The response noted that the Westinghouse RETRAN computer code allows the user to select between different options for controlling the choked flow limitations for each junction, and that Westinghouse elected to use the **[[**

]]. The response also stated:

> In reality, the model selected makes little difference because the choked flow rate used for the pressurizer and steam line relief and safety valves in the Westinghouse RETRAN model is [[
>
>]]

As a result, regardless of the choked flow model selected, the Westinghouse RETRAN model [[

]] This minimizes the water relieved with each PORV cycle and thus minimizes the volume of water required to refill the pressurizer and to increase the pressure to the opening setpoint for the next PORV opening. This minimizes the time until the next opening, increasing the frequency of the PORV cycling and minimizing the time operators have to stop the event before 300 PORV cycles are reached. This model and the other assumptions noted in Section 3.4.3 of the LAR ensure that the results of the analysis are conservative to determine the minimum required operator action times and the maximum number of pressurizer PORV cycles.

In SRXB-RAI-2 dated June 4, 2015, the NRC staff also questioned if other critical flow models were considered and how the modelled pressurizer PORV flow rate compares to the design flow rate. In its RAI response dated July 8, 2015, the licensee stated that other RETRAN critical flow models were not considered in the analysis. As discussed above, the PORV flow rate is modeled in the analysis to be consistent with the minimum design flow rate by calculation of the [[ ]] Using the minimum design flow rate in the analysis means that more PORV cycles are cnecessary to relieve the required amount of water. The NRC staff concludes that this is acceptable as it results in a conservative flow rate leading to the maximum number of PORV cycles.

The licensee stated that a [[ ]] is included in the PORV model to account for the change in mass flow rate that occurs when water (instead of steam) is relieved through the valves. In SRXB-RAI-3 dated June 4, 2015, the NRC staff requested the licensee to describe [[ ]] in more detail and state what value was used in the analysis. In its RAI response dated July 8, 2015, the licensee stated that the analysis model was conservatively developed to calculate water relief through the PORVs based on a minimum relief capacity through the PORV. [[

]]

The resulting calculated [[ ]] is used in the PORV model to account for the change in mass flow rate that occurs when water (instead of steam) is relieved through the valves. The NRC staff concludes that this method is acceptable as it results in a conservative flow rate leading to the maximum number of PORV cycles.

The assumptions made in the new FLB pressurizer filling analysis are similar to those used in the current FLB analysis. However, there are several assumptions used in the current FLB analysis for calculation of the minimum margin to hot-leg saturation that may not be conservative for pressurizer filling concerns. The assumptions for the new pressurizer filling analysis are conservatively chosen to minimize the time to reach a water-solid condition in the pressurizer (i.e., pressurizer filled) and maximize the number of pressurizer PORV cycles predicted. The licensee performed sensitivity studies for numerous parameters to determine the appropriate conservative assumptions applicable to the pressurizer filling condition.

Separate cases to accommodate different limiting assumptions were analyzed by the licensee to determine the minimum time by which the operators would need to ensure a PG&E Design Class I PORV is available and the times by which the operators would need to isolate charging flow and subsequently stop RCP seal injection flow. Cases were also analyzed both with and without offsite power available to determine the more limiting condition. Given the new analyses use different cases for the different time critical operator actions, the NRC staff questioned, in SRXB-RAI-9 dated June 4, 2015, if an analysis was performed using all four of the proposed time critical operator actions in a single run to assure that all the acceptance criteria are met at the same time.

In its RAI response dated July 8, 2015, the licensee stated that an analysis was not performed using all four of the proposed TCOAs in a single run. The purpose of the new analysis is to determine the limiting time operators have to perform TCOAs that prevent water relief through the PSVs. Separate analysis cases were run to calculate the limiting time to ensure a PORV is available (TCOA-1) because the conditions that result in limiting values for ensuring a PORV is available are different than the conditions that result in limiting times to isolate charging flow (TCOA-3) and stop RCP seal injection flow (TCOA-4). Time to isolate the faulted generator (TCOA-2) remains constant as an input to the analysis. For example, the limiting condition for times to isolate charging flow (TCOA-3) and stop RCP seal injection flow (TCOA-4) occurs when a PORV is available from the beginning of the event to relieve steam (since this maximizes the number of PORV cycles). The time required to ensure a PORV is available in this scenario is not relevant, since a PORV is already available from the beginning of the event (the PORV begins to relieve steam at 20.9 seconds from the time the feed line rupture occurs). Another example of the differences in modeling is that the limiting condition for calculating the time to ensure a PORV is available occurs with a loss of offsite power, whereas the limiting condition for the times to isolate charging flow and stop RCP seal injection flow occurs when offsite power is available. After reviewing the licensee's explanation, the NRC staff agrees that, because the limiting values are different for the different TCOAs, it is not necessary to perform an analysis using all four TCOAs in a single run. The NRC staff concludes that this an acceptable method for determining the limiting values for the TCOAs.

The licensee determined the conservative assumptions based on a full-power analysis. The licensee then determined that the full-power condition was bounding. The assumptions used were judged by NRC staff to be appropriate for the pressurizer filling scenario because the assumptions minimized operator action time to omplement TCOAs and maximized the number of PORV cycles before the FLB event was terminated.

In the LAR, one of the assumptions used by the licensee stated that:

> [R]elief through the PORVs that are actuated on the indicated (measured) pressurizer pressure signal (i.e., the PG&E Design Class I PORVs) has been modeled with the assumptions that maximize the number of PORV opening cycles experienced.

In SRXB-RAI-4 dated June 4, 2015, the NRC staff asked the licensee to describe the specific assumptions used to model the PORV to maximize the number of PORV cycles. In its RAI response dated July 8, 2015, the licensee stated that in addition to the conservatism discussed in the calculation of the [[ ]] (as discussed earlier), the assumptions used to model the PORV to maximize the number of PORV cycles include:

1. The minimum PORV opening and closing setpoints, accounting for uncertainty, are used in the model. Use of the minimum setpoints causes the PORVs to cycle more often to control pressure to the lower pressure value, reaching the maximum number of PORV cycles slightly sooner in the analysis.

2. The analysis models the minimum delay times between when the PORV opening or closing setpoint is sensed by the pressurizer pressure transmitter and when the PORV starts to open or close. The calculated opening and closing delay times each include an electronic delay followed by a pneumatic air supply delay.

Using the minimum delay time shortens the time between PORV cycles, which results in the maximum number of PORV cycles slightly sooner in the analysis.

3. The analysis models the minimum opening and closing stroke times from the time the PORV starts to open or close until the time the PORV actuator is pressurized to the full open value or depressurized to the full closed value. Using the minimum stroke time shortens the time between PORV cycles, resulting in reaching the maximum number of PORV cycles being reached slightly sooner in the analysis.

4. The analysis assumes only one PG&E Design Class I PORV is available for the duration of the event. For cases where the single failure assumed was something other than a PG&E Design Class I PORV, both Class I PORVs could be assumed to be available for pressure relief on demand. However, when modeling the two PG&E Design Class I PORVs, the licensee assumed the opening and closing setpoints are identical for each valve. If the opening setpoints are assumed to be offset by even a fairly small difference in pressure, it would be expected that only the PORV with the lower opening setpoint would initially cycle open and close (i.e., it would be expected that this PORV would open quickly enough and have sufficient capacity to maintain pressure below the higher opening setpoint of the second PORV), until the maximum number of cycles has been reached, after which the PORV with the higher opening setpoint would be available. Assuming the opening and closing setpoints for the two valves are identical is equivalent to assuming only one PORV is available for the duration of the event, thereby reaching the maximum number of PORV cycles sooner in the analysis.

The NRC staff evaluated the other assumptions used to model the PORV to the maximum number of PORV cycles, but determined them to not be significant to the outcome. Based on the preceding discussion, the NRC staff concludes that the specific assumptions described above, used by the licensee to maximize the number of PORV cycles, are acceptable.

Given the capacity of the modified PG&E Design Class I backup nitrogen accumulators, the licensee's acceptance criterion is that "transient mitigation must be demonstrated to occur before 300 PORV cycles is reached." For this criterion to be met, the number of pressurizer PORV cycles would have to be 299 or less. Section 3.4.4 of the LAR states, in part, that "The system response showing that the maximum number of PORV cycles is not reached is presented in new UFSAR Figures 15.4.2-24 through 15.4.2-27." In Figure 15.4.2-26 of the UFSAR markup, the red curve, indicating calculated PORV cycles, appears to slightly exceed the dotted grid line at 300 cycles. The NRC staff asked the licensee, in SRXB-RAI-5 dated June 4, 2015, to confirm that the actual number of pressurizer PORV cycles in the analysis is below 300.

In its RAI response dated July 8, 2015, the licensee stated that the total number of PORV cycles shown on the graph in Figure 15.4.2-26 of the UFSAR markup is 300 cycles. Table 15.4-8, Sheet 5 of 5 of the UFSAR markup indicates that the steam bubble forms again in the pressurizer at 6,723 seconds (112.05 minutes), before the maximum number of 300 PORV cycles is reached at 7,137.6 seconds (119 minutes). Once the steam bubble forms, the transient is ended, since the PSV is no longer in jeopardy of relieving water. The number of PORV cycles experienced before the steam bubble forms at 6,723 seconds is 295. To ensure this is clear in the UFSAR markup, the second paragraph on page 15.4-44 will be revised to the following:

> The system response is presented in Figures 15.4.2-24 through 15.4.2-27. Table 15.4-8, "Sequence of Events," indicates that a steam bubble forms again in the pressurizer at 6723 seconds. This occurs at cycle 295 before the maximum number of 300 PORV cycles is reached at 7137.6 seconds.

The NRC staff concludes that this is acceptable, because the number of PORV cycles calculated to use the most conservative assumptions and limiting values to ensure the highest number of PORV cycles possible is calculated, is less than 300 when the steam bubble forms in the pressurizer thus preventing water relief through the PSVs.

For a given opening setpoint pressure, an increase in the closing setpoint pressure will result in additional pressurizer PORV cycles, which will deplete the backup nitrogen supply more quickly. The NRC staff asked the licensee what pressurizer PORV opening/closing setpoint pressures were used in the analysis and how these compare to the current plant setpoints. The staff questioned, in SRXB-RAI-6 dated June 4, 2015, the uncertainty in the physical PORVs opening/closing pressures, including any setpoint drift effects and if these uncertainties were taken into consideration in the analysis.

In its RAI response dated July 8, 2015, the licensee stated that pressurizer pressure signals are fed to the PORVs, with PORVs opening on high pressure and closing on low pressure. The current plant setpoint pressures are 2,335 pounds per square inch gage (psig) for opening and 2,322.5 psig for closing. The closing setpoint is based on the opening setpoint minus 12.5 psig (1 percent of span) dead band. The calculated overall channel uncertainty for the PORV actuation is plus or minus 22.63 psig, which includes sensor drift (plus or minus 1.2 percent of span), rack drift (plus or minus 0.2 percent of span) and display device drift (plus or minus 0.25 percent of span), in addition to other uncertainties such as calibration accuracies and sensor temperature effects.

The FLB pressurizer filling analysis uses minimum setpoint pressures of 2,305 psig for opening and 2,292.5 psig for closing. Use of the minimum setpoints causes the PORVs to cycle more often to control pressure to the lower value, resulting in reaching the maximum number of PORV cycles slightly sooner in the analysis. The minimum values are based on current plant setpoints minus 30 psig to bound the overall channel uncertainty of 22.63 psig. The licensee also provided a table showing the PORV setpoint pressures and uncertainties. Because the selected opening/closing setpoints maximizes the number of PORV cycles that will occur before termination of the FLB event, the NRC staff concludes that this is acceptable.

In the current pressurizer filling analysis, some AFW flow is assumed to begin 1 minute after trip, however, in the UFSAR Section 15.4.2.2 FLB analysis, AFW does not begin for 10 minutes. In SRXB-RAI-7 dated June 4, 2015, the NRC asked the licensee to explain the difference in AFW modelling assumptions between the UFSAR FLB and the new analysis and to describe how the new analysis is conservative.

In its RAI response dated July 8, 2015, the licensee stated that as described in the LAR, Assumption 3.4.3 (13), the AFW system is designed with one turbine-driven AFW pump (TDAFWP) that supplies AFW to all four SGs, and two motor-driven AFW pumps (MDAFWPs) that each independently provide flow to two of the four SGs. All three AFW pumps are designed to begin providing flow within 1 minute after the generation of an SG low-low level trip signal occurs. With the limiting single failure of the TDAFWP, and the assumption that the MDAFWP flow feeding the faulted SG spills out the break, the second MDAFWP is available to provide AFW flow to two of the intact SGs within 1 minute after the SG low-low trip signal occurs. The FLB pressurizer filling analysis also included cases with the single failure as the MDAFWP aligned to the two intact SGs such that AFW flow to the intact loop from the TDAFWP does not start until 10 minutes after the transient begins; however, these cases were not limiting for the pressurizer filling analysis.

For the acceptance criteria in the UFSAR 15.4.2.2 FLB analysis (demonstrate that no bulk boiling occurs in the RCS), there is analysis margin available to assume that no AFW flow is credited until after the operators isolate the faulted SG at 10 minutes. The FLB pressurizer filling analysis presented in the LAR has a more restrictive acceptance criteria (prevent water relief through the PSVs), which is sensitive to the minimum AFW system flow. Therefore, for the cases not assuming single failure of the MDAFWP aligned to the intact loop, the FLB pressurizer filling analysis credits the MDAFWP connected to two intact SGs to begin providing AFW flow at 1 minute after the SG low-low level trip signal occurs, as designed.

The previous analysis made a more conservative assumption relative to the design of the AFW system, assuming a delay of 10 minutes before AFW flow. A longer delay of AFW flow resulted in a more conservative calculation for the SSI analysis. The new analysis assumption of a 1-minute delay before AFW flow is also conservative relative to the design of the AFW system in the context of the pressurizer filling scenario, as a faster AFW flow results in the pressurizer filling faster.

The NRC staff concludes that the AFW assumptions are acceptable as the faster AFW flow leads to a more conservative result in the new analysis relative to the acceptance criteria of preventing water relief through the PSVs.

The loss of both PG&E Design Class I pressurizer PORVs would result in water relief through the PSVs. The NRC staff questioned, in SRXB-RAI-10 dated June 4, 2015, if there is any single assumed failure (common cause) that would cause loss of backup nitrogen to both safety grade pressurizer PORVs.

In its RAI response dated July 8, 2015, the licensee stated that there is no single failure that would cause loss of backup nitrogen to both safety grade pressurizer PORVs. As described in the LAR, Section 3.1, each unit at DCPP has two PG&E Design Class I pressurizer PORVs, each with an independent PG&E Design Class I backup nitrogen supply system to provide motive force when instrument air is lost to containment. The only physical plant change being made in support of the LAR is to increase the size of the nitrogen backup accumulator to allow for sufficient PORV cycles to mitigate the FLB pressurizer filling event. A PG&E design change is being implemented to install the larger nitrogen accumulators along with some minor changes in the mounting configuration, in order to ensure that the two nitrogen backup supply systems remain independent per PG&E Design Class I requirements.

3.2.3 FLB Pressurizer Filling Analysis Results

The results of the analysis demonstrate that if the pressurizer fills following an FLB, the following TCOAs preclude water relief through the PSVs:

- Ensure a Pressurizer PORV is Available within 8.6 minutes
- Isolate the Faulted SG within 10 minutes (input assumption)
- Isolate Charging Flow within 25 minutes
- Stop RCP Seal Injection Flow within 45 minutes

If no pressurizer PORV relief is available at the start of the transient because of a failure of one of the PG&E Design Class I PORVs and the other is isolated by its respective block valve, operator action is required to ensure a PG&E Design Class I PORV is available in time to prevent water relief through the PSVs. The analysis determined that the minimum time to pressurizer filling is 8.3 minutes and the minimum time to subsequently lift the PSVs is 8.6 minutes. Therefore, the operators must ensure a PG&E Design Class I PORV is available within 8.6 minutes of event initiation.

In UFSAR Section 15.2.15.2, Spurious Safety Injection (SSI) Pressurizer Overfill Analysis, the operators are credited with making a pressurizer PORV available within 11 minutes of the initiation of the event. This event is considered a Condition II fault of moderate frequency. For this case, the 11-minute time frame is consistent with times in the American National Standards Institute/American Nuclear Society (ANSI/ANS)-58.8-1994 standard which assumes 5 minutes for diagnosis and one additional minute for each action. However, in the current FLB pressurizer overfill analysis, the event is a Condition IV limiting fault (much lower frequency of occurrence) and credits operator action within 8.6 minutes. This value is significantly lower than the ANSI/ANS-58.8-1994 standard which assumes 20 minutes for diagnosis and five additional minutes for each operator action for this type of event. In SRXB-RAI-8 dated June 4, 2015, the NRC staff asked the licensee to address the discrepancy between the proposed operator action time and the ANSI/ANS standard.

In its RAI response dated July 8, 2015, the licensee stated that ANSI/ANS-58.8 is referenced in Information Notice (IN) 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," and states, in part, that "ANSI-58.8 provides estimates of reasonable response times for operator actions; however, licensees may use time intervals derived from independent sources provided they are based on analyses with consideration given to human performance." IN 97-78 also provides nine criteria used by the NRC to review analyses crediting time critical operator actions, which address factors that affect human performance. Section 3.5 of the LAR provides a discussion of the proposed TCOAs with respect to the nine criteria, and the NRC acceptance is addressed in Section 3.3 of this safety evaluation.

Accounting for 20 minutes of diagnosis time per the standard is not appropriate, since DCPP emergency operating procedures (EOPs) are symptom-based. This is supported by the existing TCOA for the FLB accident to isolate a faulted SG within 10 minutes, which was included in the original DCPP Final Safety Analysis Report (FSAR) approved by the NRC.

The action to ensure a PORV is available within 11 minutes is an existing TCOA for the SSI event. The action to make a PORV available is also an existing early action in EOPs used to respond to transients involving a reactor trip, SI, loss of reactor coolant, or loss of secondary coolant. Since operators typically perform this action in less than 5 minutes, the proposed TCOA of 8.6 minutes allows ample recovery time from any credible error. PG&E conducted three simulator runs of the FLB event that demonstrated operators can perform the proposed credited actions within the analyzed timeframes. Results of the three demonstrations were provided. The NRC staff concludes that this is acceptable as PG&E conducted simulator demonstrations of the FLB event that demonstrated operators can perform the credited actions within the analyzed timeframes.

The licensee found that the full-power cases are limiting when compared to the lower power cases. The results for Unit 2 are more limiting than those calculated for Unit 1. The primary difference is that the model for Unit 2 considers the plant changes associated with the upflow conversion/upper head temperature reduction. In addition, the upper end of Tavg window is slightly higher for Unit 2, resulting in an initial pressurizer water level that is also slightly higher.

With respect to the single failure scenarios, the licensee found the failure of the TDAFWP is limiting for the calculation of minimum time to pressurizer filling, unless one of the two PG&E Design Class I PORVs is already blocked at the start of the transient. If a PORV is blocked, the failure of the other PG&E Design Class I PORV is limiting for pressurizer filling. The failure of a Class I PORV is also limiting for the calculation of the operator actions times required to ensure that transient mitigation is complete before the maximum number of PORV cycles is reached.

For cases with offsite power available, pressurizer pressure is maintained after the RCP seal injection flow is stopped, since the pressurizer heaters (specifically, the backup heaters, actuated on high pressurizer level deviation) continue to operate. However, as a steam bubble forms again in the pressurizer and the pressurizer water volume begins decreasing, relief flow switches from water to steam. At this point, there is no longer a concern relative to water relief through the PSVs and transient mitigation is complete for these cases. For cases with a loss of offsite power, pressurizer pressure decreases after the RCP seal injection flow is stopped and the pressurizer PORV setpoint and PSV setpoint are no longer challenged, signifying transient mitigation is complete.

3.3 Human Factors

3.3.1 Description of Operator Action(s) Added/Changed/Deleted

In an RAI response letter dated July 8, 2015, the licensee stated that the following four time critical operator actions (TCOA) will be changed:

1. Ensure a PG&E Design Class I pressurizer PORV is available within 8.6 minutes (previously 11 minutes),

2. Isolate the faulted SG within 10 minutes,

3. Isolate charging flow within 25 minutes (previously 14-15 minutes), and

4. Stop RCP seal injection flow within 45 minutes.

The operators have had previous training on and experience with all of the actions being credited as TCOAs, and have demonstrated capability of performing these actions within the allotted timeframes during simulator training. Therefore, the NRC staff concludes that these changes are acceptable.

3.3.2 Operating Experience Review

A precedent review was performed to identify applicable industry experience regarding mitigating the effects of pressurizer filling during a FLB accident. This review identified the previous NRC approval of the Westinghouse version of RETRAN-02 for the Seabrook SSI pressurizer-filling analysis, which was discussed in section 4.2 of the LAR.

As discussed above, the actions being credited in the new FLB pressurizer filling analysis are existing actions in the EOPs. NUREG-1764, Revision 1, Section 3.2 provides that an operating experience review is not necessary if existing human-interface components are to be used without modifications and those components are currently used for safety-related functions. In this case, no changes are being made to controls, alarms, or displays used for the added or modified TCOAs, and those controls, alarms, and displays are currently used for safety-related functions. Therefore, in accordance with NUREG-1764, Revision, Section 3.2, an operating experience review was not performed.

3.3.3 Functional Requirements Analysis and Function Allocation

Because the existing operator actions associated with the proposed change are simple, are part of existing plant procedures, and do not add significant workload, a re-analysis of the functional requirements analysis and function allocation is not necessary. No further analysis is needed beyond that provided by the licensee. The NRC staff concludes that the changes are acceptable based on the fact that there are no additional complex operator actions and, therefore, no significant change to operator workload.

3.3.4 Staffing

Based on the simplicity of operation, no new or additional staff are required, nor are there any new or additional qualifications required to perform the actions within the time constraints established. The NRC staff determines that no additional staffing or qualifications, or changes thereto, are required, and concludes that this human performance aspect of the LAR is acceptable.

3.3.5 Probabilistic Risk and Human Reliability Analyses

The licensee chose not to submit a risk-informed application using probabilistic risk assessment/human reliability analyses and, therefore, did not identify any additional human reliability insights that might be applicable to operator performance. The E-1.3 procedure will provide clear guidance on the conditions required to implement necessary actions, and as an approved plant procedure, provide appropriate plant status control. However, because a probabilistic basis for plant changes is not required, this approach is acceptable to the NRC staff.

3.3.6 Human-System Interface Design

The actions being credited in the new FLB pressurizer filling analysis are existing actions in the EOPs. No changes are being made to controls, alarms, or displays.

3.3.7 Procedure Design

The following procedures will be revised as a result of the proposed LAR:

Emergency Operating Procedure (EOP) E-1.1, "Safety Injection Termination" (current revisions Unit 1-29, Unit 2-22) will require revision to implement the license amendment. A step will be added to EOP E-1.1 to cycle the remaining operating charging pump off-and-on to control pressurizer level if letdown cannot be established. This action is currently included in EOP FR-I.1, "Response to High Pressurizer Level," but has not been previously credited as a TCOA.

Based on the procedure changes described by the licensee, the NRC staff concludes that appropriate revisions to plant procedures have been made or will be made in support of the proposed LAR.

3.3.8 Human Factors Verification and Validation

3.3.8.1 Discuss the sample of operators used for validation and how it was representative of the population of operators to perform the four actions.

The licensee conducted three simulator demonstrations of the FLB event that demonstrated operators can perform the proposed credited actions within the analyzed timeframes. Crews did not have prior review of the event or training on the new step to be added to EOP E-1.1. The participants in the three demonstrations described below were representative of the minimum crew required to on duty in the control room during plant operation. Use of the minimum crew for one of the demonstrations allows for a more conservative evaluation of the ability of the crew to implement the TCOAs in time.

Demonstration 1: A three-person crew consisting of one senior reactor operator (SRO) and two reactor operator (RO) qualified personnel.

Demonstrations 2 and 3: Two typical control room crews in operations continuing training. Each crew consists of a mix of SRO and RO qualified personnel.

To implement the license amendment, the TCOA to stop RCP seal injection flow will be added to EOP E-1.1. The TCOAs will be formally documented and validated in accordance with the TCOA program described in Section 3.3 of the LAR, and all operations crews will be trained on the change in accordance with the operating procedure change process.

3.3.8.2 Results of the validation including operators' response times for each action, and the total time for each operator to perform all four actions.

PG&E conducted three simulator runs of the FLB event that demonstrated operators can perform the proposed credited actions within the analyzed timeframes. Results of the three demonstrations are provided below. The time to perform TCOA-4 represents the total time to perform all four actions. The times were recorded by crew and no operator errors were observed.

Proposed FLB TCOA Demonstrations [1]

Proposed TCOA No.[2]	Proposed TCOA Description	Credited Time	Crew Response Times			
			Demo 1	Demo 2	Demo 3	AVG
1	Ensure a PORV is available	8.60	2.90	3.57	4.57	3.68
2	Isolate the faulted SG[3]	10.00	6.75	9.45	9.05	8.42
3	Isolate charging flow	25.00	12.75	15.85	14.73	14.44
4	Stop RCP seal injection flow	45.00	34.90	27.75	29.00	30.55

[1] All times are in minutes.
[2] TCOA numbers correspond to numbers assigned in the LAR on page 15.
[3] Existing TCOA for FLB accident.

The validation demonstrates that all of the TCOAs can be performed within the allotted timeframes requested by the license amendment. The NRC staff has concluded that the results of the validation are acceptable.

3.4 Summary

The new FLB pressurizer filling analysis demonstrated that the acceptance criterion, no liquid relief through the pressurizer PSV, can be met assuming operators successfully perform TCOAs. The TCOAs prevent water relief through the PSVs by ensuring the PORVs are available for pressure relief following an FLB, and by stopping reactor coolant addition via charging flow. By preventing liquid relief through the PSVs, leakage through the pressure boundary due to the PSVs failing to reseat is precluded. Thus, the new analysis meets GDCs 9 and 33 by demonstrating the TCOAs prevent liquid relief through the PSVs. The licensee demonstrated TCOAs preclude adverse effects of a loss of normal cooling due to an FLB pressurizer filling event in accordance with GDC 43.

The operator actions credited in the FLB pressurizer filling event analysis will be included in the DCPP TCOA program, which ensures the operators are trained on the actions and the response times for the actions are periodically validated. The NRC staff has reviewed the proposed changes to TCOAs associated with the incorporation into licensing basis of pressurizer filling analysis for major rupture of a main feedwater pipe accident. The NRC staff review basis included the nine criteria provided in IN 97-78. The NRC staff concludes that this request is acceptable with respect to the proposed changes in operator manual actions.

The new FLB pressurizer filling analysis demonstrates that timely performance of TCOAs ensures PORVs are available during an FLB pressurizer filling event and prevent the PSVs from relieving water and subsequently failing to reseat. Thus, the licensee has demonstrated that implementation of the TCOAs will ensure that the reactor coolant pressure boundary integrity will be maintained as required by NUREG-0737. The licensee also performed analyses that confirmed the PG&E Design Class I PORVs and the pressurizer discharge piping are qualified for the hydrodynamic loads resulting from the number of PORV water cycles expected. The PORV pneumatic operators are classified as Design Class I, and the controls and solenoid valves are Instrument Class IA with Class 1E power. The pneumatic operators and control circuitry meet the environmental qualification requirements, ensuring the PORVs are capable of operating under the harsh conditions that could occur in the event of an FLB.

The addition of the new analysis to the DCPP UFSAR satisfies the requirements of 10 CFR 50.34(b) which specifies the content requirements for the UFSAR and 10 CFR 50.71(e) which requires periodic updates of the FSAR to ensure the information included in the report contains the latest information developed.

The NRC staff evaluated the licensee's proposed changes and the licensee's assessment of the impact of the proposed changes to the DCPP UFSAR. The NRC staff found that the new analysis of an FLB using an NRC previously approved code and methodology meets the requirements of GDCs 9, 33, and 43, NUREG-0737 Item II.D.1, and 10 CFR 50.34 and 50.71. Based on the considerations discussed in the preceding sections, the NRC staff concludes that the proposed LAR incorporates into the licensing basis an analysis of pressurizer filling concerns associated with the main feedwater pipe rupture accident summarized in DCPP UFSAR Section 15.4.2.2 and is, therefore, acceptable. Based on the above, the NRC staff concludes that the proposed LAR to add the FLB pressurizer filling analysis to the UFSAR is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding published in the *Federal Register* on April 28, 2015 (80 FR 23605). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: R. Beaton, NRR
M. Keefe, NRR

Date: February 19, 2016

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/

Siva P. Lingam, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosures:
1. Amendment No. 223 to DPR-80
2. Amendment No. 225 to DPR-82
3. Safety Evaluation (Proprietary)
4. Safety Evaluation (Non-Proprietary)

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