September 26, 2003

MEMORANDUM TO: File

FROM: Girija S. Shukla, Project Manager, Section 2 **/RA/**
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: DIABLO CANYON NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 – LICENSEE'S RESPONSE TO NRC'S REQUEST FOR ADDITIONAL INFORMATION – LICENSE AMENDMENT REQUEST 01-08, "CREDIT FOR AUTOMATIC ACTUATION OF PRESSURIZER POWER OPERATED RELIEF VALVES; PRESSURIZER SAFETY VALVE LOOP SEAL TEMPERATURE" (TAC NOS. MB6758 AND MB6759)

On September 12, 2003, Pacific Gas and Electric Company (PG&E or licensee) submitted the attached draft response to the NRC's request for additional information (RAI) dated May 28 and June 5, 2003.

Docket Nos. 50-275
and 50-323

Attachment: Draft RAI Response

September 26, 2003

MEMORANDUM TO: File

FROM: Girija S. Shukla, Project Manager, Section 2 **/RA/**
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: DIABLO CANYON NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 – LICENSEE'S RESPONSE TO NRC'S REQUEST FOR ADDITIONAL INFORMATION – LICENSE AMENDMENT REQUEST 01-08, "CREDIT FOR AUTOMATIC ACTUATION OF PRESSURIZER POWER OPERATED RELIEF VALVES; PRESSURIZER SAFETY VALVE LOOP SEAL TEMPERATURE" (TAC NOS. MB6758 AND MB6759)

On September 12, 2003, Pacific Gas and Electric Company (PG&E or licensee) submitted the attached draft response to the NRC's request for additional information (RAI) dated May 28 and June 5, 2003.

Docket Nos. 50-275
and 50-323

Attachment: Draft RAI Response

DISTRIBUTION:
PUBLIC
PDIV-2 Reading
RidsNrrPMGShukla
RidsNrrLAEPeyton

**ADAMS ACCESSION NO.: ML032790217** **NRR-088**

| OFFICE | PDIV-2/PM | PDIV-2/LA | PDIV-2/SC |
|---|---|---|---|
| NAME | GShukla | EPeyton | SDembek |
| DATE | 9/25/03 | 9/25/03 | 9/25/03 |

DOCUMENT NAME: C:\ORPCheckout\FileNET\ML032790217.wpd
OFFICIAL RECORD COPY

September xx, 2003

PG&E Letter DCL-03-0xx

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Request for Additional Information Regarding License Amendment Request 01-08, "Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature"

Dear Commissioners and Staff:

Pacific Gas & Electric (PG&E) letter DCL-02-115, dated September 24, 2002, submitted License Amendment Request (LAR) 01-08, "Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature." LAR 01-08 would modify Technical Specification (TS) 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," and the licensing basis to credit -automatic actuation of the Class 1 PORVs, instead of the pressurizer safety valves (PSVs), to limit reactor coolant system (RCS) pressure changes for the spurious operation of the safety injection system at power event, and other design basis accidents. Also, TS 3.4.10, "Pressurizer Safety Valves," would be revised to allow PSV loop seal temperatures to be less than the lower design temperature during plant heatup and cooldown in Mode 3, and in Mode 4 when any RCS cold leg temperature is greater than the low temperature overpressure protection arming temperature specified in the pressure temperature limits report, provided at least one Class I PORV is available and capable of providing automatic pressure relief. This would allow gradual stabilization of the loop seal temperatures, and avoid having to partially drain the loop seals to establish the proper PSV inlet temperature.

On May 28 and June 5, 2003, the NRC staff transmitted several requests for additional information concerning LAR 01-08. PG&E's responses to the staff's questions are provided in Enclosure 1.

This additional information does not affect the results of the technical evaluation and no significant hazards consideration determination previously transmitted in PG&E Letter DCL-02-115.

If you have any questions or require additional information, please contact Stan Ketelsen at (805) 545-4720 or Tom Grozan at (805) 545-4231.

Sincerely

David H. Oatley

*Vice President and General Manager - Diablo Canyon*

jer/3664
Enclosures
cc: Edgar Bailey, DHS
Thomas P. Gwynn
David L. Proulx
Diablo Distribution
cc/enc: Girija S. Shukla

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

| | | |
|---|---|---|
| | ) | Docket No. 50-275 |
| In the Matter of | ) | Facility Operating License |
| PACIFIC GAS AND ELECTRIC COMPANY | ) | No. DPR-80 |
| | ) | |
| Diablo Canyon Power Plant | ) | Docket No. 50-323 |
| Units 1 and 2 | ) | Facility Operating License |
| | ) | No. DPR-82 |

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath states that he is Vice President and General Manager - Diablo Canyon of Pacific Gas and Electric Company; that he has executed this response to the NRC request for additional information on License Amendment Request 01-08 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

David H. Oatley
*Vice President and General Manager - Diablo Canyon*

Subscribed and sworn to before me this xxth day of September, 2003.

Notary Public
County of San Luis Obispo
State of California

**PG&E Response to NRC Requests for Additional Information Regarding License Amendment Request 01-08, "Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature."**

Questions 1 through 11 were received on May 28, 2003.

NRC Question 1

*The licensee proposed change to TS 3.4.10, "Pressurizer Safety Valves", would require that at least one Class I PORV available for automatic pressure relief while the PSVs are considered as inoperable due to low PSV loop seal temperature. Since the PORVs are active components which are subject to single active failure following an event, please explain how this proposed change will satisfy the single failure criteria for system required to mitigate a design basis transient or accident while the plant is operating in Modes 3 & 4 and the RCS cold leg temperature is greater than LTOP arming temperature.*

PG&E Response

Technical Specification (TS) 3.4.11 requires that all power-operated relief valves (PORV) be operable in Modes 1, 2, and 3. Therefore more than one PORV will be available for pressure relief in Mode 3. The PORVs are not required to be operable in Mode 4 when the reactor coolant system (RCS) cold leg temperature is above the low temperature overpressure (LTOP) arming temperature in the Pressure Temperature Limits Report (PTLR). This is because both pressure and core energy are decreased and the pressure surges are much less significant than in the Modes 1, 2, and 3. The Bases for TS 3.4.10 state the pressurizer safety valves (PSV) are conservatively required to be operable in Mode 3 and portions of Mode 4 although the listed accidents may not require the PSVs for protection. The accidents listed are not likely to occur in Mode 4.

The proposed change modifies the TS 3.4.10 Actions, and would allow gradual stabilization of the loop seal temperatures, thus avoiding partial draining of the loop seals to establish the proper loop seal inlet temperature. If the proposed change had been written as a TS Action (e.g., if the PSV loop seal temperature is less than the lower design temperature, immediately verify that at least one PORV is available, and capable of providing automatic pressure relief, and take action to restore the loop seal temperature to within its design limit), the single failure criterion would be temporarily relaxed (a single failure need not be postulated such that a loss of function occurs) during the limited time the loop seal temperature is less than the design temperature.

The note is intended to provide protection while not forcing the plant to be placed in a lower mode without both PORVs being operable or available. In Mode 3, 1 of the 2 Class I PORVs can be inoperable for 72 hours. The note allows for the PORV(s) to be determined operable to meet Limiting Condition for Operation (LCO) 3.4.11 prior to Mode 3 entry during plant startup, or to meet the LCO 3.4.12 prior to the RCS temperature decreasing below the LTOP arming temperature during a plant shutdown, where the PORVs are required to provide overpressure protection. Based on the limited time the plant would be in Mode 4 during startup or shutdown, the proposed change is justified.

NRC Question 2

*The licensee indicated in its submittal that the safety grade PORVs are also credited in other design basis accidents. Please provide the following:*

*(a) A list of other design basis events using PORVs for accident mitigation. Explain why they are qualified for your evaluation in accordance with 10 CFR 50.59 and do not require the staff review.*
*(b) Address the staff concern of a single failure of the mitigating system during those design basis events assuming only one operable PORV required by the proposed TS 3.4.10.*

PG&E Response

(a) The following summary discusses the design basis accidents that credit the pressurizer PORV for mitigation and the license amendment or safety evaluation that incorporated mitigation into the Diablo Canyon Power Plant (DCPP) licensing basis.

Steam Generator Tube Rupture (SGTR)

PG&E credits the manual actuation of the PORVs to depressurize the RCS as one of the operator actions required to mitigate the SGTR accident. In PG&E Letter DCL-88-114, “Steam Generator Tube Rupture (SGTR) Analysis,” dated April 29,1988, PG&E submitted the revised SGTR analysis using the NRC accepted Westinghouse Owners Group methodology established in WCAP-10698. The plant specific SGTR analysis for DCPP was documented in WCAP-11723 and provided as an enclosure to DCL-88-114. In Section C.4, WCAP-11723 states that the operator action to depressurize the RCS is accomplished using the pressurizer PORVS, when the reactor coolant pumps (RCP) are not running and pressurizer sprays are not available.

In DCL-88-114, PG&E identified the pressurizer PORVs and their backup air supply as being available for SGTR mitigation.

In NRC Letter dated April 3, 1991, “Closeout of Steam Generator Tube Rupture Analysis Issue for Diablo Canyon Power Plant, and Finding of Compliance with Condition 2.C.(9) of Unit 2 Operating License DPR-82,” the enclosed safety evaluation (SE) confirmed that the DCPP PORVs primary motive power was non safety-related instrument air, but acknowledged the availability of safety grade back air supply. The SE confirmed the list of equipment credited for SGTR mitigation to be acceptable.

In PG&E Letter DCL-01-115, “License Amendment Request 01-05, Revision to Technical Specification 1.1, ‘Definitions, Dose Equivalent I-131,’ and Revised Steam Generator Tube Rupture and Main Steam Line Break Analyses,” PG&E submitted LAR 01-05 to revise the dose consequences for the SGTR accident and identified in Section 4.1.2 that the pressurizer PORVs are credited to depressurize the RCS since the limiting case assumes the RCPs are tripped and pressurizer spray is not available. The NRC accepted this dose analysis in License Amendments (LA) 156 (Unit 1) and 156 (Unit 2) dated February 20, 2003.

<u>Low Temperature Overpressure Protection</u>

TS 3.4.12, “Low Temperature Overpressure Protection (LTOP) System,” requires two of the three PORVs to be Class 1 rated for LTOP pressure protection. All of the PORVs are air operated, and the two safety-related PORVs have a nitrogen gas backup to the non safety-related air supply. This credited function of the PORVs was accepted by NRC as documented in LAs 133 (Unit 1) and 131(Unit 2) and the associated SE dated May 3, 1999.

<u>Steam Generator Alternate Repair Criteria</u>

In PG&E Letter DCL-97-034, “License Amendment Request 97-03, Voltage-Based Alternate Steam Generator Tube Repair Limit for Outside Diameter Stress Corrosion Cracking at Tube Support Plate Intersections,” dated February 26, 1997, PG&E confirmed that DCPP was in compliance with the recommendations identified in Generic Letter (GL) 95-05, “Voltage-Based Repair Criteria For Westinghouse Steam Generator Tubes Affected By Outside Diameter Stress Corrosion Cracking.” These recommendations are listed in Attachment 1 of GL 95-05 which is entitled; “Guidance for a Proposed License Amendment to Implement an Alternate Steam Generator Tube Repair Limit for Outside Diameter Stress Corrosion Cracking at the Tube Support Plate Intersections.” Attachment 1, Section 2, “Tube Integrity Evaluation,” discusses crediting the pressurizer PORVs for limiting the maximum primary to secondary pressure during a main steam line break (MSLB) event. GL 95-05, Attachment 1, Section 2 states:

> “For plants in which the TS do require the PORVs to be operable, the assumed differential pressure for the conditional burst probability calculation may be based on the PORV setpoint in lieu of the safety valve setpoint with similar adjustments. The TS requirements for operation with PORV block valves closed due to leaking PORVs should be in accordance with Enclosure A of Generic Letter 90-06, "Resolution of Generic Issue 70, ‘Power-Operated Relief Valve and Block Valve Reliability,’ and Generic Issue 94, ‘Additional Low-Temperature Overpressure Protection for Light-Water Reactors,’ Pursuant to 10 CFR 50.54(f)." That is, electrical power to the block valves must be maintained to allow continued operation with the block valves closed, as required in the sample technical specification Section 3.4.4 of GL 90-06.”

This credit for PORV mitigation was incorporated into the DCPP design and licensing basis upon NRC acceptance documented in LAs 124 (Unit 1) and 122 (Unit 2) dated March 12, 1998.

Appendix R Safe Shutdown

In response to GL 81-12, “Fire Protection Rule” and GL 86-10 “Implementation of Fire Protection Requirements,” PG&E established that the pressurizer PORVs would be available to perform a controlled RCS depressurization in order to achieve safe shutdown conditions as defined in 10 CFR 50 Appendix R. RCS overpressure protection for the Appendix R safe shutdown scenario is provided by the pressurizer safety valves. Since NRC acceptance of these generic letter responses was not required, crediting the pressurizer PORVs for the Appendix R safe shutdown scenario was incorporated into the DCPP design basis per 10 CFR 50.59.

Anticipated Transients Without Scram (ATWS) Mitigating System Actuation Circuitry (AMSAC)

Automatic actuation of the PORVs is credited in the generic ATWS analysis for AMSAC provided in topical report WCAP-10858-P-A, Revision 1, which was referenced in PG&E Letter DCL-87-258, “Plant-Specific AMSAC Design,” dated October 30, 1987. NRC acceptance of the DCPP AMSAC design was provided in NRC Letter dated August 15, 1988, “Safety Evaluation of the AMSAC System, PG&E’s Proposed Method of Implementing the Requirements of 10 CFR 50.62 (ATWS) for Diablo Canyon.”

(b) The NOTE being added to LCO 3.4.10 is acceptable since the only credible event in Mode 3 that credits the automatic actuation of the pressurizer PORV for RCS overpressure protection (in lieu of the PSV) is the spurious safety injection (SSI) event. (In Mode 3 with only core decay heat, ATWS and MSLB events are not considered RCS overpressure events.) The SSI analysis results and the operator simulator demonstration times submitted as part of LAR 01-08 have established that one PORV is capable of mitigating the SSI event, and that there is adequate time available for the operators to make a pressurizer PORV available without challenging a PSV. TS 3.4.11 requires that both Class I PORVs and their associated block valves be operable. The operator action credited for the mitigation of the SSI event is to make sure both PORVs are unblocked and available for automatic actuation. Therefore, in the event of the single failure of one PORV, or the associated block valve, the other class I PORV is still available to mitigate the event.

NRC Question 3

*The current TS allows PORVs to be isolated by their associated block valves to stop excessive seat leakage during power operation. Discuss the plant Emergency Operating Procedures (EOPs) that provide guidance to reactor operators for using those PORVs in accident mitigation using plant indications.*

PG&E Response

As discussed in the response to Question 2, only two events during power operation, the SGTR and SSI events, credit operator action to unblock the PORVs and make them available for automatic actuation. The following applicable EOP procedural steps that the operators would use are summarized below:

EOP E-0, “Reactor Trip or Safety Injection,” Step 9.d requires checking power available to at least one block valve. Step 9.e requires checking at least one block valve open.

EOP E-3, “Steam Generator Tube Rupture,” Step 12 requires operators to make power available to all three PORV block valves. Step 21.a requires making at least one PORV available. Step 21.b requires opening one PORV.

EOP ECA-3.3, “SGTR Without Pressurizer Pressure Control,” provides additional guidance to operators for an event in which the operators are unable to open a PORV. Step 4 of this procedure provides operator guidance for the restoration of the pressurizer PORVs.

NRC Question 4

*TS 3.4.11 requires that all PORVs shall be operable during Modes 1, 2 and 3. The proposed change to TS 3.4.10 seems contradict to this requirement during Mode 3 operation.*

PG&E Response

The apparent contradiction referenced in this question can be resolved based on the technical difference between the applicable terms “operable” and “available for automatic actuation.” Per TS 3.4.11 all three PORVs must be operable, however only one needs to be unblocked and available for automatic actuation per the proposed change to TS 3.4.10. The SSI analysis demonstrates that there is more than adequate time for the operators to unblock any other PORVs to ensure that even with a single failure of one PORV, another Class I PORV will be available to mitigate the event. Therefore, the proposed text in TS 3.4.10 and TS 3.4.11 are not contradictory and are consistent with the assumptions in the SSI analysis.

NRC Question 5

*Figure 3 of the licensee's submittal indicates that the pressurizer will become water solid in all three cases of the analyses. Please discuss the design adequacy of the PORVs, the associated tail piping and relief tank during this event.*

<u>PG&E Response</u>

NRC Letter dated January 27, 1986, "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," issued the staff safety evaluation report (SER) for safety and relief valve testing for DCPP Units 1 and 2, in accordance with NUREG-0737, Item II.D.1. The SER provided acceptance of the adequacy of PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water transition, and subcooled water fluid) and acceptance of qualification to function properly during and following licensing basis seismic events. The SER also provided acceptance of the safety and relief valve piping between the pressurizer nozzles and the pressurizer relief tank (PRT). The adequacy of the PRT is discussed in Final Safety Analysis Report Update (FSARU) Section 5.5.10, "Pressurizer Relief Tank."

Note that LAR 01-08 proposes no changes to the design of the PORVs or related piping systems. The current approved mechanical design will remain as is. PG&E has upgraded the Instrument Class II portion of the PORV automatic actuation circuitry for Unit 2 and will complete the upgrade for Unit 1 at the next refueling outage. These upgrades were evaluated and determined not to require prior NRC approval in accordance with 10 CFR 50.59.

<u>NRC Question 6</u>

*Verify the capacity certification for the plant PORVs in accordance with the requirements of Section III of the ASME Code.*

<u>PG&E Response</u>

The pressurizer PORVs were procured and manufactured in accordance with USAS (now known as ANSI) B16.5. The requirements for these valves were established in a vendor equipment specification. There were no requirements for capacity certification in the specification. The design requirements for these valves are discussed in FSAR Update section 5.2 and the applicable codes for these valves are provided in Table 5.2-2.

The DCPP design as summarized in FSARU Table 5.5-16 contains three PORVs, two Class I, and one non-class I, each with a rated flow capacity equal to 50 percent of a safety valve or 210,000 lb/hr at 2350 psig. NUREG-0737, Item II.D.1 required all PWR plant licensees and applicants to demonstrate that their pressurizer safety valves, PORVs, PORV block valves, and all associated discharge piping will function adequately under conditions predicted for design basis transients and accidents. This requirement was met as documented in the NRC SER issued on January 27, 1986 (see response to Question 5 above).

NRC Question 7

*In accordance with Section III of the ASME Code, verify that only ½ of the total certified capacity of the available number of PORVs has been credited for applicable overpressure events in the plant safety analysis.*

PG&E Response

PG&E credits one PORV for SSI mitigation, equal to one third of the total PORV capacity, and equal to one half of the Class I PORV capacity, consistent with Section III of the ASME Code.

NRC Question 8

*Verify that the plant PORVs have been qualified to function properly during and following licensing basis seismic events.*

PG&E Response

As stated in the response to Question 5 above, NRC Letter dated January 27, 1986, "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," issued the SER for safety and relief valve testing for DCPP Units 1 and 2, in accordance with NUREG-0737, Item II.D.1. The SER provided acceptance of the adequacy of PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water transition, and subcooled water fluid) and acceptance of qualification to function properly during and following licensing basis seismic events. The SER also provided acceptance of the safety and relief valve piping between the pressurizer nozzles and the PRT.

NRC Question 9

*It is not clear how the crediting of PORV capacity for certain Mode 3 or 4 events reduces the loading conditions on the pressurizer safety valve (PSV) discharge piping. An analyzed event is the inadvertent opening of a PSV; therefore, the PSVs should be postulated to open spuriously, whether or not the PORV capacity is credited. Provide justification for not considering the loads from spuriously opening PSVs during the applicable Modes 3 or 4 conditions.*

PG&E Response

FSARU Section 15.2.13 “Accidental Depressurization Of The Reactor Coolant System,” states that the bounding accidental RCS depressurization event is based on the spurious opening of a PSV. This is an event which is analyzed at power to assure that the event does not result in the departure from nucleate boiling ratio decreasing below the minimum safety analysis limit. Since this event is limiting based on analyzing a maximum decrease in RCS pressure, the PSV safety function of RCS overpressure protection is not required for mitigation. The analysis is performed assuming bounding core kinetics and limiting Mode 1 reactor power conditions. This event is not specifically evaluated for conditions other than Mode 1.

The design basis of the pressurizer loop seal and discharge piping is to ensure that the PSV can successfully open and close as needed to mitigate a limiting RCS overpressure condition. The hydraulic loads on the discharge piping associated with a spurious opening of a PSV in Mode 3 or 4 conditions do not represent an adverse impact on any safety function required for mitigation such that the accidental RCS depressurization event currently analyzed in the FSAR remains bounding.

The adequacy of the design of the PSV and PORV discharge piping was included in the Westinghouse report “Pressurizer Safety and Relief Line Evaluation Summary Report - AM-SSA-2534, S. O. PGE/145” submitted by PG&E Letter from Philip A. Crane, Jr., to Harold R. Denton (NRC) dated December 13, 1982, and referenced in the NRC SER issued January 27, 1986, discussed above.

<u>NRC Question 10</u>

*It appears that a more direct and advantageous method of reducing the loads on the PSV discharge piping would be to eliminate the need for a loop seal upstream of the valves by refitting the PSVs with steam trim internals, which have a reduced tendency for seat leakage. Provide a discussion which addresses the need to maintain the loop seals, instead of eliminating them to significantly reduce discharge loads.*

<u>PG&E Response</u>

PG&E has considered several options to address the low loop seal temperature issue, including elimination of the loop seal under the pressurizer safety valves, but does not consider this a viable alternative. PG&E’s understanding is that a steam seat is much more sensitive to safety valve nozzle loads than a water seat. PG&E has measured the nozzle loads in the past and has implemented changes to reduce them to the point the valves have a good history of no seat leakage. Due to DCPP’s seismic requirements, PG&E does not believe that the piping support arrangement can be modified to reduce nozzle loads to the values necessary to assure no valve leakage with a steam seat.

<u>NRC Question 11</u>

*By letter dated January 27, 1982, the staff provided a safety evaluation of NUREG-0737, Item II.D.1 for Diablo Canyon 1 and 2. Therein, the staff found that according to EPRI report NP-2296 dated January 1981, the pressurizer would not become water solid until after 20 minutes following a spurious actuation of high pressure injection. The staff also found that this provided ample time for operator action to terminate the injection event and prevent water discharge through the PORVs and PSVs. The September 24, 2002, submittal states that 603$^0$F water could discharge through the PSVs in only 16 minutes. Provide a discussion to address this apparent inconsistency.*

<u>PG&E Response</u>

As discussed in the background section of LAR 01-08, several additional issues have impacted the SSI analysis and resulted in reduced margin with respect to the time available for pressurizer overfill mitigation during a spurious safety injection event, as compared to the original analyses from EPRI Report NP-2296. Consequently, PG&E has had to (1) perform a much more detailed modeling of operator actions, and (2) upgrade the PORVs to Class I status in order to credit them for mitigation. A more detailed discussion of the historical evolution of the DCPP SSI analysis is provided below:

There have been several significant industry issues that have been identified since EPRI Report NP-2296 was published in January 1981. In Nuclear Safety Advisory Letter (NSAL) -93-013, “Inadvertent ECCS Analysis at Power,” dated June 30, 1993, Westinghouse identified that the original analysis methodology might not be conservative for evaluating pressurizer overfill since it was established for verifying minimum departure from nucleate boiling ratio (DNBR) limits. Soon after this, Westinghouse issued Supplement 1 to NSAL-93-013 dated October 28, 1994, which identified that the analysis methodology did not include modeling the operation of the non safety-related positive displacement pump (PDP) which could lead to earlier pressurizer overfill times than previously analyzed. Based on Westinghouse indications that pressurizer overfill could occur as early as 10 minutes into the event, PG&E initiated a nonconformance report (N0001973) to address the issue.

The issues were finally resolved when Westinghouse issued a new SSI analysis (Westinghouse Letter PGE-96-565 dated May 31, 1996) that credited operator action to terminate the event within 16 minutes. The analysis also demonstrated that the minimum liquid temperature of 603.2°F, which was relieved through the pressurizer safety valves for this event was acceptable. This analysis still represents the design basis analysis for pressurizer overfill that is referenced in the DCPP FSARU Section 15.2.13. In December 1997, Westinghouse informed PG&E that a review of the FSARU analysis for a spurious operation of the SI system at power event identified that a temperature coefficient used for PSV modeling was treated incorrectly. The Westinghouse review determined the temperature coefficient defined in WCAP-11677, "Pressurizer Safety Relief Valve Operation for Water Discharge During a Feedwater Line Break," January 1988, Appendix A is not a constant as previously treated, but rather varies with temperature. Previously, it was concluded that PSV operability would be maintained for a water relief temperature above 600°F. With the temperature coefficient correctly treated as a variable, the water temperature must remain above 613°F in order to justify stable PSV operation.

Westinghouse Letter PGE-98-502 dated January 15, 1998, identified that the pressurizer spray function could provide more effective pressure reduction and an earlier pressurizer overfill condition than previously analyzed. In order to resolve these outstanding issues, PG&E submitted DCL-99-071, “License Amendment Request (LAR) 99-01, Unreviewed Safety Question - Spurious Operation of the Safety Injection System at Power,” dated May 21, 1999, which was based on revised Westinghouse analysis results that credited the pressurizer PORVs for mitigation of the SSI event and prevented opening of the pressurizer safety valves. After discussions with the NRC staff regarding the proposed LAR, PG&E withdrew the request pending review of other options.

In summary, numerous analysis and operational issues have been identified since the original EPRI report was published, which have significantly impacted the SSI analysis results. Therefore, for LAR 01-08, PG&E has performed a more detailed modeling of operator actions, and also upgraded the pressurizer PORVs to Class I in order to credit them for mitigation of the SSI event.

<u>NRC Question 12</u>

*The following question was provided in a series of communications from June 5 to June 13, 2003, and is summarized as follows: How will actual operator action times for an SSI event be verified as conservatively bounded by the action times assumed in the SSI analysis?*

<u>PG&E Response</u>

The operator action times assumed in the SSI analysis are time critical operator actions (TCOA) established and controlled by Administrative Procedure OP1.ID2, “Time Critical Operator Action.” Operators will receive specific training (both initial and requalification) on the functional importance of performing these actions with respect to acceptable mitigation of the SSI event and ensuring that pressurizer overfill does not occur. Operating crew performance will be evaluated during an appropriate SSI scenario on the simulator and the recorded operator action times will be compared to the TCOAs assumed in the SSI analysis. If the recorded operator action times are not conservatively bounded by the TCOA times assumed in the analysis, then an evaluation must be performed to demonstrate acceptable results with respect to preventing pressurizer overfill condition, or the operator crews will be retrained on performing the appropriate actions within acceptable time frames.