ENCLOSURE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY Diablo Canyon Power Plant Unit 2	Docket No. 50-323 Facility Operating License No. DPR-82 License Amendment Request No. 90-02

Pursuant to 10 CFR 50.90 and 50.91(a)(5), Pacific Gas and Electric Company (PG&E) hereby applies to amend its Diablo Canyon Power Plant (DCPP) Facility Operating License No. DPR-82 (License).

The proposed change amends the Unit 2 Technical Specifications (Appendix A of the License) regarding Technical Specification 3.4.2.2.

Information on the proposed change is provided in Attachments A and B.

This change has been reviewed and is considered not to involve a significant hazards consideration as defined in 10 CFR 50.92 or an unreviewed environmental question. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

Subscribed to in San Francisco, California this 20th day of February 1990.

OFFICIAL SEAL
BIANCA E. ZELNIK
NOTARY PUBLIC - CALIFORNIA
San Francisco County
My Commission Expires July 30, 1991

Respectfully submitted,

Pacific Gas and Electric Company

By _____
J. D. Shiffer
Senior Vice President
and General Manager
Nuclear Power Generation

Howard V. Golub
Richard F. Locke
Attorneys for Pacific
Gas and Electric Company

By _____
Richard F. Locke

Subscribed and sworn to before me
this 20th day of February 1990

_____
Bianca E. Zelnik, Notary Public
for the City and County of San Francisco
State of California

My commission expires July 30, 1991.

3082S/0078K

Attachment A

EMERGENCY TECHNICAL SPECIFICATION CHANGE TO TECHNICAL SPECIFICATION 3.4.2.2
OPERATION WITH ONE PRESSURIZER SAFETY VALVE INOPERABLE AND DISABLED

A. INTRODUCTION

On February 20, 1990, the acoustic monitor alarm on the Unit 2 pressurizer safety valves actuated. Investigation by plant operators identified elevated pressurizer tail pipe temperature, increases in the pressurizer relief tank pressure, and over a longer time trend, increases in pressurer relief tank level. On the basis of these observations, it was concluded that pressurizer safety valve 8010B had begun to leak.

If a pressurizer safety valve leaks to the extent that the loop seal is lost, a downward shift in the valve lift pressure is expected. This shift increases the likelihood of the valve opening below the specified lift pressure. A valve lift, followed by a failure to close, creates the potential for a small break loss of coolant accident (LOCA). Even if the valve reseats at a lower pressure and terminates the discharge, the RCS blowdown will result in an RCS system depressurization transient.

The benefits of disabling a leaking safety valve are, therefore, to prevent the consequences of inadvertent valve actuation and lessen the likelihood of a possible small break LOCA. There are no adverse consequences of disabling one valve, because the remaining valves have sufficient pressure relief capacity. Additionally, operability of a power operated relief valve (PORV) in the automatic mode with its associated block valve open will be required for additional conservatism.

B. DESCRIPTION OF AMENDMENT REQUEST

This license amendment request (LAR) proposes to revise Technical Specification (TS) 3.4.2.2, "Safety Valves - Operating," to allow continued operation with one pressurizer Code safety valve inoperable and disabled, until no later than March 11, 1990.

The proposed TS change allows operation with two pressurizer safety code valves operable provided the third pressurizer Code safety is disabled to be incapable of opening and that one PORV is operable in the automatic opening mode with its associated block valve open.

Changes to the TS are noted in the marked-up copy of the applicable TS (Attachment B).

C. BACKGROUND

Overpressure protection of the Reactor Coolant System (RCS) in Modes 1, 2, and 3 is provided by the Code safety valves located on the RCS pressurizer. There are a total of three pressurizer Code safety valves with a nominal lift setting of 2485 psig. There are also three PORVs located on the RCS pressurizer. The flow capacity of one PORV is approximately 50 percent of the flow capacity of one safety valve.

Recent safety valve testing results show that the absence of a water loop seal can significantly reduce the lift point (up to eight percent) when the valve lift point setting was established in the presence of a loop seal. For example, a valve set with a loop seal present will lift at a lower pressure with the loop seal absent. PG&E considers that a safe and prudent approach to the effects of the loss of loop seal is to disable the leaking safety valve, as proposed in this LAR.

Review of industry operating reports indicates several cases of safety valve leakage. In one case, a leaking safety valve has been reported to have inadvertently lifted and resulted in a manual reactor trip and RCS depressurization to the point of near actuation of safety injection. There has also been a case where excessive leakage from a safety valve resulted in rupture of the pressurizer relief tank (PRT) rupture disc and release of the tank contents to the containment building. Similar to the leakage experienced on safety valve 8010B, DCPP has previously experienced a problem with leakage through a pressurizer safety valve into the PRT as described in Licensee Event Report (LER) 2-89-006, dated September 26, 1989. The leakage experienced at DCPP did not exceed the limits in TS 3.4.6.2, "Reactor Coolant System - Operational Leakage," and did not rupture the PRT rupture disc; however, PG&E voluntarily shut down the plant to replace the valve. Replacement of a safety valve requires a plant cooldown to Mode 5 (less than 200 degrees) because the valve is not isolable from the RCS.

Since the RCS overpressure protection function can be satisfied by operation of two of the three installed safety valves, disabling the leaking safety valve eliminates the potential of a small break LOCA due to inadvertent opening of the valve. The valve can be disabled with the plant at power. The personnel radiation exposure which would be encountered during this activity is considered to be acceptable.

Disabling is a feature of the valve design and is a relatively simple operation. The valve cap has a plug located over the top of the valve stem. To disable the valve, the plug is removed from the cap and a screw is installed in the threaded hole and turned down against the top of the valve stem. Consequently, the valve stem cannot lift and the valve disc is held down against its seat.

D. JUSTIFICATION

Disabling the leaking safety valve 8010B avoids any potential for unexpected transients and possible challenges to ESF associated with

inadvertent opening of the valve. Valve opening, with subsequent closure at a lower pressure, could lead to RCS depressurization. Operation with excessive leakage could cause rupture of the PRT disc and release of its contents to the containment. In addition, failure of the valve to reclose creates the potential for a small break LOCA. The proposed revision provides for a safe and prudent action that reduces the likelihood of such events.

An increase in valve leakage would result in the loss of loop seal and shift in the valve setpoint by 4 to 8%, in excess of the value stated in the LCO. Without this relief, it will be necessary to shut the unit down and initiate the outage prior to its scheduled time. In addition, the unit's generation is required at the present time to provide continuity of service within the service territory.

The Emergency TS change to allow Unit 2 continued operation with two safety valves operable meets the requirements of the ASME Code and of Section 5.2.2 of the Standard Review Plan (SRP), "Overpressure Protection." Continued operation of the plant with two of the three safety valves operable will meet the FSAR Update acceptance criteria for overpressure protection, and result in no operational concerns, while alleviating potential safety concerns involved with operation or plant shutdown with a leaking safety valve.

As an additional conservatism, not required by the analysis, the proposed change requires one pressurizer PORV to be operable and its associated block valve to be open. The operability requirement for the PORV to meet this revised LCO is different from the requirements for the PORVs given in TS 3.4.4, "Reactor Coolant System Relief Valves." The PORV is to be in the automatic opening mode to meet the revised requirements of TS 3.4.2 since the PORV is functioning as a safety valve. Whereas, to meet the requirements of TS 3.4.4, the PORVs may be in manual with the block valves closed, since this specification addresses a different function for the PORVs.

E. SAFETY EVALUATION

The function of the pressurizer safety valves is to ensure that the primary system pressure does not exceed design limits. These design limits are separated into several categories depending upon the classification of the event being evaluated. As noted in Section 15.2.8 of the SRP, the pressure in the RCS should be maintained below 110 percent of design pressure (2750 psia) for low probability events (ANS Conditions I, II, and III) and below 120 percent of design pressure (3000 psia) for very low probability events (ANS Condition IV). The PORVs provide additional relief capacity and actuate at a lower pressure than the safety valves. This reduces the actual duty on the safety valves, because the PORVs may provide sufficient relieving flow to avoid lifting the safety valves for most pressure excursions.

1. Design Basis Accidents

An evaluation of this proposed change was performed by Westinghouse to support a reduction in the number of operable pressurizer safety valves from three to two. This evaluation determined the impact on the DCPP FSAR Update Chapter 15 accident analyses. The LOCA analyses are not limiting for this evaluation, since they are primary-side depressurization events. Two non-LOCA events were identified as being potentially limiting. The evaluation of these two events demonstrates that two pressurizer safety valves provide sufficient overpressure protection to satisfy the design limits and do not violate the acceptance criteria for the accident analysis.

DCPP FSAR Update Chapter 15 transients that rely on opening of the pressurizer safety valves for accident mitigation are listed below.

FSAR Section	
15.2.7	Loss of External Electrical Load and/or Turbine Trip
15.4.2.2	Major Rupture of a Main Feedwater Pipe
15.4.4	Single Reactor Coolant Pump Locked Rotor
15.4.6	Rupture of a Control Rod Mechanism Housing

Of these events, the Condition II Loss of External Electrical Load and/or Turbine Trip (Section 15.2.7) and the Condition IV Single Reactor Coolant Pump Locked Rotor (Section 15.4.4) events are the most limiting accidents. For the remainder of the FSAR Update events, there is sufficient margin available to the RCS pressure safety limit such that no additional analysis is required. The following addresses the Loss of External Load and/or Turbine Trip and Single Reactor Coolant Pump Locked Rotor events.

The evaluation is described as follows:

a. Loss of External Electrical Load and/or Turbine Trip

For this ANS Condition II event, cases are analyzed at beginning and end of life conditions both with and without pressurizer control. Of these four cases, only the cases without pressurizer control are limiting in terms of peak RCS pressure. For the cases without pressurizer control, the event is terminated by a high pressurizer pressure reactor trip. This event was analyzed assuming that only two of the three pressurizer safety valves were operable. The analyses results show that the RCS pressure safety limit (110 percent) is satisfied and the conclusions of the FSAR Update remain valid.

b. Single Reactor Coolant Pump Locked Rotor

Although this is an ANS Condition IV event, meeting the ANS Condition I, II, and III criterion of 2750 psia has been applied. Based upon the conservative analysis performed for the

VANTAGE 5 fuel upgrade, only 40 percent of the full steam relief of the pressurizer safety relief capacity was assumed. This corresponds to less than two of the three safety valves. The results of this analysis show that the RCS pressure safety limit (110 percent) is satisfied and the conclusions of the FSAR Update remain valid.

The effect of clearing the loop seal is not explicitly modeled in the existing safety valve methodology currently used by Westinghouse. WCAP-10105, "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Valve Test Program," explicitly analyzed the effect of clearing the loop seal and concluded that there was enough margin in the Westinghouse overpressure analyses to envelop that effect, and therefore, the existing methodology was acceptable. The results of this analysis were evaluated in light of WCAP-10105 and it was shown that the RCS pressure safety limit (110 percent) is satisfied and the conclusions of the FSAR Update remain valid.

2. Anticipated Transients Without Scram

Anticipated Transients Without Scram (ATWS) are not part of the DCPP FSAR Update analysis and license basis. However, in response to NRC questions in a NRC/PG&E meeting on July 20, 1989, the effect of having two of three safety valves operable was evaluated for the ATWS condition.

Two analyses have addressed this question. In performing the analyses, the methodology described in WCAP-11993, "Joint WOG/Westinghouse Program: Assessment of Compliance with the ATWS Rule Basis for Westinghouse PWRs," was used. This methodology employs the use of a probabilistic model for ATWS that includes the important factors involved in assessing ATWS core damage frequency. The model used in the reference study was shown to be conservative with respect to Diablo Canyon.

The results of the Westinghouse PRA ATWS analysis show that the target core melt frequency of 1E-5 is met for the case of one safety valve being disabled for an entire cycle and under the assumption of no PORVs available. In the case of one PORV available, the results are improved and slightly below the target. In both cases, the intent of the ATWS rule is met for Diablo Canyon operation with one safety valve disabled.

In addition, PG&E has performed an independent probabilistic analysis using the DCPP PRA model. The results of this preliminary analysis demonstrate that the Westinghouse ATWS results are conservative in that they show that the target core melt frequency of 1E-5 is met.

3. Potential for Inadvertent Valve Opening

The safety valves are a pressure boundary component and must remain closed when not required to open in order to maintain RCS integrity.

There has been at least one event where a plant experienced a plant trip as a result of the inadvertent opening of a leaking safety valve and subsequent RCS depressurization. Because of the significant effect on lift pressure of the water loop seal, the loss of a loop seal can lower the safety valve lift point to approach the operating pressure, thus increasing the possibility of inadvertent opening. For this reason, DCPP is requesting approval in this LAR to disable a leaking safety valve in order to prevent this possibility and the potential for a small break LOCA.

Conclusions

The results of this evaluation show that only two of the three pressurizer safety valves are needed to function properly for RCS overpressure protection and pressure integrity. Westinghouse performed an evaluation of affected Design Basis Accidents, which demonstrates that the overpressure protection limits of the accident analysis and the ASME Code requirements continue to be met for DCPP operation with a reduction in the number of operable pressurizer safety valves from three to two. In addition, ATWS was evaluated, and operation of only two pressurizer safety valves has been shown to satisfy the ATWS rule target core melt risk of 1E-5.

Disabling of the leaking safety valve prevents the possibility of a spurious opening and associated RCS depressurization. Therefore, RCS overpressure protection is maintained by the TS revision requested by this LAR. Further, the disabling of a leaking safety valve will enhance plant safety by reducing the potential of a small break LOCA in the event that the inadvertent opening and subsequent depressurization is followed by a failure of the safety valve to reseat.

The proposed change to TS 3.4.2.2 provides for continued Unit 2 operation with one pressurizer safety valve inoperable, provided that the leaking valve is disabled and that one PORV is operable in the automatic mode with its block valve open. These provisions address the industry experience with leaking safety valves. The availability of the PORV provides additional assurance of overpressure protection as it provides additional relieving capacity and reduces the duty on the other two safety valves.

Based on the results of the evaluation, PG&E believes there is reasonable assurance that this change will not adversely affect the health and safety of the public.

F. NO SIGNIFICANT HAZARDS EVALUATION

PG&E has evaluated the hazard considerations involved with the proposed amendment focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

The following evaluation is provided for the three categories of the significant hazards consideration standards.

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

   The pressurizer safety valves are designed to mitigate overpressurization transients in the Reactor Coolant System (RCS). A safety evaluation for plant operation with two of three pressurizer safety valves operable (one pressurizer valve inoperable and disabled) has been performed. The results show the RCS overpressure limits of the two limiting accidents previously analyzed, Loss of External Load and/or Turbine Trip and Reactor Coolant Pump Locked Rotor, are not exceeded for the case of operation with two pressurizer safety valves. The change reduces the potential for RCS depressurization resulting from spurious leaking safety valve actuation.

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

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

   The change does not affect the method by which any safety-related system performs its function. The two safety valves will operate in the same manner and provide the same characteristic valve response as prior to the proposed change. The potential consequences of two valve operation have been addressed in the safety evaluation and demonstrated to be acceptable.

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

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

A safety evaluation for operation with one safety valve inoperable and disabled demonstrated that the RCS overpressure limits of the two limiting accidents previously analyzed, Loss of External Load and/or Turbine Trip and Reactor Coolant Pump Locked Rotor, are not exceeded for plant operation with two pressurizer safety valves. In addition, the requirement for one operable PORV in automatic mode with its associated block valve open as a condition for operation with safety valve 8010B disabled provides additional pressure relieving capability. This provides additional conservatism since the PORV relief capacity is not included in the accident analysis evaluation.

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

G. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

In conclusion, based on the above evaluation, PG&E submits that the activities associated with this LAR satisfy the no significant hazards consideration standards of 10 CFR 50.92(C) and, accordingly, a no significant hazards consideration finding is justified.

H. ENVIRONMENTAL EVALUATION

PG&E has evaluated the proposed change and determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released off site, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c) (9). Therefore, pursuant to 10 CFR 51.22(B), an environmental assessment of the proposed change is not required.

Attachment B

MARKED-UP TECHNICAL SPECIFICATIONS

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