LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Diablo Canyon Unit 1	0 5 0 0 0 2 7 5	1 OF 9

TITLE (4)

Reactor Coolant System Outside Design Basis for Inadvertent Emergency Core Cooling System Actuation at Power Due to Non-Conservative Assumptions for Pressurizer Safety Valve Operation

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MON	DAY	YR	YR	SEQUENTIAL NUMBER	REVISION NUMBER	MON	DAY	YR	FACILITY NAME	DOCKET NUMBER
1	22	98	98	- 0 0 1 -	0 0	2	23	98	Diablo Canyon Unit 2	0 5 0 0 0 3 2 3

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (11)
1	X 10 CFR 50.73(a)(2)(ii)(B)
POWER LEVEL (10)	OTHER
1 0 0	(SPECIFY IN ABSTRACT BELOW AND IN TEXT, NRC FORM 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
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COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRPDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRPDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MON	DAY	YR
[X] YES (If yes, complete EXPECTED SUBMISSION DATE)	[ ] NO		11	15	98

ABSTRACT (16)

On January 22, 1998, at 0912 PST, PG&E determined that the reactor coolant system was outside design bases for an inadvertent emergency core cooling system actuation at power. A revised analysis indicated that water may pass through the pressurizer safety valves (PSVs) at temperatures lower than required to ensure valve operability. A 1-hour non-emergency report was made to the NRC at 0925 PST, in accordance with 10 CFR 50.72(b)(1)(ii)(B).

The root cause was determined to be personnel error (cognitive) by non-utility personnel who failed to account for the temperature dependence of an empirically derived coefficient (alpha). This coefficient is used in an equation for calculating the minimum water relief temperature to ensure stable PSV operation.

PG&E performed an assessment demonstrating that the operability of the PSVs would be maintained. This assessment credits a revised emergency operating procedure that terminates a safety injection prior to the PSVs lifting. It also places restrictions on positive displacement charging pump operation.

The corrective actions for this condition have not yet been determined. PG&E will supplement this LER to report corrective actions.

TEXT

I. Plant Conditions

Units 1 and 2 were in Mode 1 (Power Operation) at 100 percent power.

II. Description of Problem

A. Summary

On January 22, 1998, at 0912 PST, PG&E determined that the reactor coolant system (RCS)(AB) was outside design bases for an inadvertent emergency core cooling system (ECCS)(BQ) actuation at power. A revised analysis indicated that water may pass through the pressurizer safety valves (PSVs)(RV) at temperatures lower than required to ensure valve operability. A 1-hour non-emergency report was made to the NRC at 0925 PST, in accordance with 10 CFR 50.72(b)(1)(ii)(B).

B. Background

The RCS transports heat from the reactor to the steam generators. Each unit has three safety-related PSVs that are part of the RCS pressure boundary and prevent RCS components from exceeding their design pressure ratings.

Each unit also has three power operated relief valves (PORVs). The PORVs' safety-related functions in Mode 1 provide RCS pressure boundary and manual pressure control for accident mitigation. Additionally, nonsafety-related automatic RCS pressure relief is provided to minimize challenges to the safety valves.

Technical Specification (TS) 3/4.4.4, "Reactor Coolant System- Relief Valves," requires the PORVs and their associated block valves to be operable. The block valve must be closed if a PORV has excessive seat leakage. TS 4.4.4.1 requires a channel calibration of the actuation instrumentation on each PORV and that each PORV be operated through one cycle of travel at least once every 18 months.

NUREG-0737, item II.D.1, required qualification of the PSVs under expected operating conditions for design basis transients and accidents. The testing must demonstrate that the valves will open and shut under the expected flow conditions.

C. Event Description

On June 30, 1993, Westinghouse issued Nuclear Safety Advisory Letter (NSAL)-93-013 regarding potential non-conservative assumptions that were used in the licensing analysis of the inadvertent operation of the ECCS at power accident. This condition could result in the PSVs actuating and relieving water. Since the PSVs were not originally qualified for water relief, there was the possibility the PSVs might not shut after actuating. This would result in loss of coolant since the PSVs cannot be isolated. PG&E initiated its problem resolution process and contracted Westinghouse to perform the analysis recommended in NSAL-93-013.

On February 11, 1994, Westinghouse completed the analysis and confirmed that overfill of the pressurizer did not occur. PG&E closed the problem tracking resolution in response to this analysis.

On October 28, 1994, Westinghouse issued NSAL-93-013, Supplement 1. The supplement indicated that operating the positive displacement pump (PDP) during normal operation would tend to aggravate the event by reducing the time to reach a pressurizer water solid condition.

On December 14, 1994, Westinghouse sent PG&E the supporting final safety analysis report (FSAR) mark-ups for the new analysis. During the FSAR Update review, operations personnel recognized they could not support the analysis assumptions for a 10 minute operator action time to isolate safety injection. PG&E requested Westinghouse to perform a reanalysis.

On May 31, 1996, Westinghouse provided a new evaluation that indicated overfill and limited water relief through the PSVs might occur if PORV actuation is not credited. The analysis used the methodology in WCAP-11677, "Pressurizer Safety Relief Operation for Water Discharge During a Feedwater Line Break," as the basis for the justification. The WCAP was developed from data extracted from the EPRI-CE PWR Safety Valve Test Report (EPRI NP-2770-LD, Volume 6). Analysis results indicated that water relief would only occur at temperatures of 603°F or more, and use of Westinghouse WCAP-11677 formulas indicated that this was acceptable. Simulator runs verified that operators could terminate the event prior to water temperature decreasing to less than 603°F.

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On August 13, 1996, PG&E Letter DCL-96-177, “Reanalysis of Inadvertent Emergency Core Cooling System Actuation Accident,” dated August 13, 1996, provided a copy of a reanalysis of the inadvertent ECCS actuation accident to the NRC. This transmittal included a 10 CFR 50.59 safety evaluation for the FSAR Update change. The safety evaluation concluded that the PSVs were demonstrated to be operable for the limited amount of water relief based on testing performed by EPRI.

On December 18, 1997, Westinghouse informed PG&E that using the EPRI test data to support the 1996 inadvertent ECCS actuation analysis could be non-conservative. The coefficient alpha in WCAP-11677 had been assumed to be a constant over all conditions. Westinghouse determined a better and more conservative approach was to make alpha temperature dependent.

On December 19, 1997, PG&E completed a preliminary engineering evaluation to confirm that there was not an immediate operability concern. This evaluation was based on the availability of the PORVs automatic actuation circuitry that had performed as designed during past ECCS actuation events. The evaluation also identified numerous conservatisms in the FSAR analysis that result in pressurizer overfill well before simulator model or engineering's expectations.

On December 23, 1997, PG&E prepared a prompt operability assessment (POA) and placed administrative controls to maintain the PDPs shut down but available during normal operation. Maintaining the PDPs shut down decreases the potential ECCS injection flow rate and prevents pressurizer overfill from occurring.

On December 26, 1997, as part of an unrelated effort to streamline the emergency operating procedures (EOP), PG&E revised E-0, “Reactor Trip or Safety Injection,” to streamline operator actions in response to an ECCS actuation. The revision resulted in a decrease in the time required to terminate an ECCS actuation.

On January 13, 1998, Westinghouse completed their evaluation and formally notified PG&E that the 1996 inadvertent ECCS actuation at power analysis was non-conservative. The analysis also provided information regarding appropriate restrictions on operation of the PDP.

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On January 14, 1998, the PG&E POA was revised to permit limited operation of the PDPs. The revised POA determined that using the revised EOPs, the inadvertent ECCS actuation would be terminated prior to lifting the PSVs. Compensatory measures were established to control PDP operation to support the POA.

On January 22, 1998, at 0912 PST, PG&E determined that the RCS may have been outside design bases for an inadvertent ECCS actuation at power in the past, under certain equipment configurations, and depending on the EOP E-0 revisions in effect at the time.

On January 22, 1998, at 0925 PST, PG&E made a 1-hour non-emergency report to the NRC in accordance with 10 CFR 50.72(b)(1)(ii)(B).

D. Inoperable Structures, Components, or Systems that Contributed to the Event

None.

E. Dates and Approximate Times for Major Occurrences

1. January 22, 1998, at 0912 PST: PG&E determined that the RCS was outside design bases for an inadvertent ECCS actuation.

2. January 22, 1998, at 0925 PST: A 1-hour non-emergency report was made to the NRC in accordance with 10 CFR 50.72(b)(1)(ii)(B).

F. Other Systems or Secondary Functions Affected

None.

G. Method of Discovery

PG&E was notified by Westinghouse.

H. Operator Actions

None.

I. Safety System Responses

None.

III. Cause of the Problem

A. Immediate Cause

Water predicted to pass through the PSVs during an inadvertent ECCS actuation analysis could have a lower temperature than the valves are qualified to pass.

B. Root Cause

The root cause was determined to be personnel error (cognitive) by non-utility personnel who failed to account for the temperature dependence of an empirically derived coefficient (alpha). This coefficient is used in an equation for calculating the minimum water relief temperature to ensure stable PSV operation.

IV. Analysis of the Event

Inadvertent actuation of the ECCS is a Condition 2 event in the FSAR. The current FSAR analysis states that the PSVs will successfully operate without adverse consequences during the analyzed over fill condition. Offsite power is maintained during this event. PSVs installed at Diablo Canyon Power Plant (DCPP) are only qualified for temperatures of 613°F or higher. Based on the assumptions used in the analysis, the operators cannot terminate the ECCS actuation before the water temperature decreased below 613°F. However, several assumptions in the analysis are more conservative than actual plant conditions. These include PORV unavailability, pressurizer level, centrifugal charging pump (CCP) flow rates, and RCS temperature.

The current operability assessment is based on an analysis that used the current CCP pump curves, rather than the maximum curves used for loss-of-coolant accident sensitivity analyses. The operability assessment also assumes a pressurizer level based on current RCS temperature (T_{avg} = 574°F), rather than the pressurizer level associated with the design RCS temperature (T_{avg} = 577.6°F for Unit 2). The analysis also credits operator actions provided in the revised EOPs to reduce the ECCS injection flow rate. This analysis shows that the

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inadvertent ECCS actuation is terminated prior to overfill of the pressurizer, without crediting operation of the PORVs. This evaluation bounds the current operation of the plant.

For past operability, operational occurrences that could potentially challenge the PSVs include normal operation of the PDP, use of EOP revisions that resulted in slower termination of the inadvertent ECCS event, and instances where the PSVs were found with lift setpoints lower than allowed by TS. Crediting the PORVs for automatic actuation bounds these past operational occurrences.

Although the PORVs are not credited in the accident analysis since they are PG&E design Class II, they are available to mitigate pressure transients. Automatic actuation of any one of the three PORVs will prevent challenging the PSVs during an inadvertent ECCS actuation event. The majority of the automatic circuit components are either Class I or meet Class I requirements. Some sections of the wiring inside the control panels do not meet Class I separation requirements, although all field wiring meets all separation requirements. A failure mode effects analysis (FMEA) was performed to evaluate the PORV automatic control circuits. The FMEA concluded that there were no credible failures that could disable more than one PORV.

The PORV automatic circuitry is maintained such that it is expected to be highly reliable. TS 4.4.4.1 requires remote manual valve stroke and PORV automatic actuation circuitry channel calibration on a refueling basis. PG&E reviewed the surveillance and maintenance history since 1990. The valves have successfully completed all routine stroke tests. No problems were identified with the PORV controllers and actuation circuitry that would have prevented them from actuating automatically. All of the components are part of the Maintenance Rule Program, and no PORV components are in goal setting.

A review of TS equipment tracking sheets from 1992 to present showed that at least two PORVs have been available, with their block valves open, at all times during power operations on each unit. Additionally, the PORVs have performed as designed during ECCS actuation events at DCPP. Consequently, should an inadvertent ECCS actuation occur, the PORVs would have been available and the automatic actuation circuitry would have actuated to relieve RCS pressure and prevent challenging the PSVs.

At certain times in the past, the TS for maximum PSV lift setting tolerance of +/-1 percent drift has been exceeded. A PSV lift setting higher than 1 percent above the setpoint is not a concern for this condition. A PSV lift setting lower

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than the 1 percent tolerance would promote a faster overfill. A review of the PSV lift setting as-found data back to 1989 shows that the most limiting case was a low lift at 2391 psig (-3.8 percent). Since the PORVs are set to lift at 2335 psig, the PSV would not have been challenged at any time.

DCPP has three PORVs. No more than one PORV has been blocked during power operations. If a single failure in the automatic actuating circuitry disabled a second PORV, then one PORV would have been available to relieve pressure and prevent a challenge to the PSVs.

Thus, this event did not adversely affect the health and safety of the public.

V. Corrective Actions

A. Immediate Corrective Actions

1. An operability assessment was performed that evaluated an inadvertent ECCS actuation. This evaluation credited a revised EOP that terminates an ECCS actuation prior to lifting the PSVs. Initially, compensatory measures were taken to control PDP operation to support the operability assessment. Further analysis demonstrated that present plant operating parameters (specifically, current pressurizer level and CCP performance curves) are such that an inadvertent ECCS actuation can be mitigated even if the PDP is operating as long as operating parameters are controlled.

2. PG&E sent a letter to Westinghouse that requested they determine if this issue is reportable to the NRC in accordance with 10 CFR 21. To date, a response has not been received.

B. Corrective Actions to Prevent Recurrence

The corrective actions for this condition have not yet been determined. PG&E will supplement this LER to report corrective actions.

VI. Additional Information

A. Failed Components

None.

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B. Previous LERs on Similar Problems

None.