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PG&E Letter DCL-02-077

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

Docket No. 50-323, OL-DPR-82
Diablo Canyon Unit 2
Licensee Event Report 2-2002-002-01
Unit 2 Manual Reactor Trip Due To Loss of Main Feedwater to a Steam Generator

Dear Commissioners and Staff:

PG&E is submitting the enclosed revision to the licensee event report regarding a manual reactor trip of Unit 2 due to a February 9, 2002, failure of main feedwater regulating valve FW 2-FCV-540 resulting in isolation of feedwater to steam generator 2-4, and actuation of the auxiliary feedwater system. The changes are noted with revision marks.

This event was considered to be of low risk significance and did not adversely affect the health and safety of the public.

Sincerely,

D. H. Oatley

David H. Oatley

rlr/2246/N0002137
cc/enc: Ellis W. Merschoff
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Diablo Distribution
INPO

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LICENSEE EVENT REPORT (LER)

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TITLE (4)

Unit 2 Manual Reactor Trip Due To Loss of Main Feedwater to a Steam Generator

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
02	09	2002	2002	- 0 0 2 -	0 1	07	10	2002		

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (11)
1	X 10 CFR 50.73(a)(2)(iv)(A)
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	
Roger Russell - Senior Regulatory Services Engineer	805	545-4327

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	S J	S O L	A 6 0 9	Yes					

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

ABSTRACT (Limit to 1400 spaces. i.e., approximately 15 single-spaced typewritten lines.) (16)

On February 9, 2002, at 0337 PST with Unit 2 in Mode 1 (Power Operation) at 100 percent power, plant operators initiated a manual reactor trip due to failure of main feedwater regulating valve (MFRV) FW-2-FCV-540 that isolated feedwater to steam generator 2-4. A 4-hour, non-emergency report was made to the NRC on February 9, 2002, at 0709 PST in accordance with 10 CFR 50.72(b)(2)(iv)(B) and 50.72(b)(3)(iv)(A). Safety systems responded as required, except for an expected automatic reactor trip based on steam generator low level, which is evaluated in LER 1-2002-001, dated April 15, 2002, as referenced in sections II.B. and IV of this LER.

The immediate cause of the failure was excess current in the coil of Asco solenoid valve (SV) SV-540B, which caused failure of a power fuse and forced the FCV-540 to close. The excess current in the SV was caused by failure of the insulation due to thermal aging degradation.

Immediate corrective actions included replacing a failed fuse and the SV. Corrective actions to prevent recurrence include: 1) revising the environmental qualification (EQ) calculation and re-evaluating the replacement period for Asco SVs; 2) replacing Asco SVs or their coils for the MFRVs and bypass valves during the next refueling outage for each unit; 3) reviewing the adequacy of other EQ calculations prepared by the same individuals; and 4) reevaluating preventive maintenance frequencies for all continuously energized EQ solenoid valves. An additional prudent action was completed to prepare a case study to communicate lessons learned from the inadequate EQ calculation.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT

I. Plant Conditions

Unit 2 was in Mode 1 (Power Operation) at 100 percent power.

II. Description of Problem

A. Background

One main feedwater isolation valve (MFIV) and one main feedwater regulating valve (MFRV) and MFRV bypass valve, are located on each of the four main feedwater (MFW) lines outside containment. The MFIVs and MFRVs and bypass valves are located upstream of the auxiliary feedwater (AFW) pump injection point so that AFW may be supplied to the steam generators (SG) following MFIV or MFRV and bypass valve closure.

The MFIVs, MFRVs and bypass valves close on receipt of any safety injection (SI) signal, or SG water level – high high signal. The main feedwater pump (MFWP) turbine is also tripped upon receipt of an SI or SG water level - high high signal. The MFRVs and bypass valves also close on receipt of a T_{avg} - Low coincident with reactor trip (P-4). They may also be manually closed.

B. Event Description

On February 9, 2002, at approximately 0336 PST, the electrical current through solenoid valve (SV) SV-540B circuit exceeded the capacity of fuse FW-2-FCV-540-FU-A[SJ][FU], which opened and isolated power to the SVs supplying control air to open both MFRV FCV-540[SJ][FCV] and MFRV bypass valve FCV-1540. The MFRV bypass valve is normally closed at power and did not move, but it was prevented from opening following the fuse opening.

The MFRV for loop 4 closed, which caused a control room alarm when SG 2-4 feed flow deviated from steam flow. Control of FCV-540 was placed in manual, but attempts to open the valve were unsuccessful because the power to the control air SV-540B[SJ][SOL] was unavailable. Operators noted that one SG 2-4 low-low level bistable actuated – two out of three are needed for automatic reactor trip and actuation of motor-driven AFW pumps.

When FCV-540 did not respond and SG 2-4 water level continued to decrease, licensed operators in the control room, responding in accordance with established procedures, initiated a manual reactor trip at 0337 PST, confirmed the reactor trip, verified appropriate engineered safety features actuations, and initiated actions to stabilize the unit in Mode 3 (Hot

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Standby).

Troubleshooting identified the failed power fuse, and SV-540B, which were both replaced. MFRVs and bypass valves were tested and declared operable.

On February 9, 2002, at 2130 PST, the Plant Staff Review Committee (PSRC) approved the Unit 2 restart with an identified anomaly in that they believed the SG wide range (WR) level indications did not behave as predicted. The WR level indication dropped below 20 percent prior to the first SG narrow range (NR) reactor trip initiate indication activating. The difference between the WR and the NR level channel indicators was identified to be larger than previously observed during operator simulator training.

The PSRC concluded the NR reactor trip channels and WR indication were acceptable for continued operation, based on the following considerations:

(1) There was consistent indication in the three NR level protection channels.

(2) One (of three) SG NR reactor trip initiate bistables activated, which indicated that one of the other two would have tripped the reactor, had it not been manually tripped.

(3) The safety function of the WR level channels is to provide indication after a reactor-trip condition for post accident assessment.

(4) The WR channel is calibrated for steady-state (cold) conditions and may not respond accurately during a significant transient.

(5) The WR and NR channels returned to normal indication following the transient.

The PSRC directed that the level indication anomaly be investigated in accordance with Inter-Departmental Administrative Procedure OM7.ID5, “Issues Needing Validation to Determine Impact on Operability,” to resolve the SG 2-4 level indication differences recorded during the transient.

On February 10, 2002, Unit 2 entered Mode 1, following replacement of the failed power fuse and SV-540B.

Problems identified with the SG NR level channels were reported in LER 1-2002-001, submitted April 15, 2002.

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

None.

D. Other Systems or Secondary Functions Affected

A review of other environmental qualification (EQ) files was conducted to determine if equipment in other systems had been in service beyond its qualified life. Additional EQ program anomalies were identified and included in the corrective action program for resolution through the nonconformance reporting process.

E. Method of Discovery

The event was immediately apparent to plant operators due to alarms and indications received in the control room due to SG 2-4 feedwater flow being less than steam flow, and decreasing SG levels.

F. Operator Actions

Operators placed FCV-540 in manual, but attempts to open the valve were unsuccessful because the power to the control air SV was unavailable. When FCV-540 did not respond and SG 2-4 water level continued to decrease, operators initiated a manual trip at 0337 PST.

G. Safety System Responses

The following systems and equipment were actuated and responded as described:

(1) The reactor trip breakers [JC][BKR] opened.
(2) The control rod drive mechanism [AA][DRIV] allowed the control rods to drop into the core.
(3) The main turbine [TA][TRB] tripped.
(4) The vital buses [EB][BU] transferred to the startup power [EA] source.
(5) The motor driven and turbine driven AFW pumps [BA][P] started.

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III. Cause of the Problem

A. Immediate Cause

The immediate cause was the failure of the electrical fuse providing holding current to the coil of SV-540B, which closed FCV-540. The cause of the fuse failure was a short in the SV-540 coil.

B. Root Cause

SV-540B, which was last replaced in 1993 and is continuously energized to perform its function, failed due to thermal aging degradation.

C. Contributory Causes

The initial preventive maintenance replacement schedule for Asco SVs was 7.5 years for this application, but a subsequent evaluation by PG&E extended the qualified life and replacement schedule to approximately 20 years. Engineers utilized data provided in NUREG/CR 3424, "Equipment Qualification Research, Test Program and Failure Analysis of Class 1E Solenoid Valves," dated November 1, 1983. However, since the plant configuration (stagnant or minimal air flow) was different than the test conditions described in NUREG/CR 3424, that data should not have been used as the basis to extend the qualified life. An industry paper published in 1987 by the Nuclear Users Group for Environmental Qualification provided a caution regarding potential misuse of the test data provided in NUREG/CR 3424.

IV. Assessment of Safety Consequences

The Bases for Technical Specification (TS) 3.7.3, "Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), MFRV Bypass Valves, and Main Feedwater Pump (MFWP) Turbine Stop Valves," states that the safety related function of the MFRV and the MFRV bypass valves is to provide the initial isolation of MFW flow to the secondary side of the SGs following a high energy line break (HELB), which includes a main steam line break (MSLB) or feedwater line break (FWLB). Since the MFRVs and MFRV bypass valves are located in non-safety related piping, the MFIVs also provide safety related isolation of the MFW flow to the secondary side of the SGs a short time later. Closure of the MFRVs and MFRV bypass valves or tripping of the MFWPs and closure of the MFIVs a short time later terminates feedwater flow to the SGs, terminating the event for FWLBs occurring upstream of the MFIVs or MFRVs. The consequences of breaks in the main steam lines or in the MFW lines downstream from the MFIVs

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will be mitigated by their closure. Closure of the MFRVs and MFRV bypass valves, or tripping of the MFWPs and closure of the MFIVs a short time later, effectively terminates the addition of feedwater to an affected SG, limiting the mass and energy release for MSLBs or FWLBs inside containment, and reducing the cooldown effects for MSLBs.

There were no safety consequences involved in this event because the MFRV performed its required function by failing closed. Although the SG water level was actually lower than intended for the automatic reactor trip before the operators manually tripped the reactor, adequate water was available in the affected SG and in the other three SGs to remove the heat from the Reactor Coolant System.

This event was bounded by the loss of normal feedwater accident, which only requires two SGs fed by one AFW pump to remove heat from the reactor following a trip. Based on the above information, PG&E used the NRC's significance determination process and believes the condition had low risk significance. Therefore, the event did not adversely affect the health and safety of the public.

NOTE: LER 1-2002-001, submitted April 15, 2002, addressed SG narrow range low-low level instrumentation inaccuracy. A 4-hour, non-emergency report addressing this issue was made to the NRC on February 14, 2002, at 1730 PST, in accordance with 10 CFR 50.72(b)(2)(i) and 50.72(b)(3)(v)(A).

V. Corrective Actions

A. Immediate Corrective Actions

Troubleshooting identified the failed fuse and SV-540B, which were both replaced. MFRVs and bypass valves were tested and found to be operable.

B. Corrective Actions to Prevent Recurrence (CAPRs)

1. Engineering will revise the EQ calculation and re-evaluate the replacement period for Asco SVs.

2. Asco SVs or their coils for the MFRVs and bypass valves will be replaced during the next refueling outage for each unit.

3. The adequacy of other EQ calculations prepared by the same individuals will be reviewed and corrected as necessary.

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4. Preventive maintenance frequencies for all continuously energized EQ solenoid valves will be reevaluated and revised as needed.

C. Prudent Action

1. A prudent action was completed to prepare a case study to communicate lessons learned from the inadequate EQ calculation.

VI. Additional Information

A. Failed Components

Asco solenoid-operated air valve[SJ][SOL], model L206-381-6F.
Fuse (Buss ABC5, 250V) [SJ][FU] for power supply to solenoid-operated air valves.

B. Previous Similar Events

A similar event was reported in Licensee Event Report 2-85-022-00, “Reactor Trip Resulting From A MFRV Closure Due To A Loose Wiring Connection,” DCL-86-008, dated January 20, 1986. However, the root cause of the 1985 event was due to construction personnel inadvertently bumping a junction box, causing a loose wiring connection to momentarily open, and the CAPRs from that LER would not have prevented the event described in this current LER.