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PG&E Letter DCL-02-090

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

Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
Licensee Event Report 1-2002-004-00
Automatic Reactor Trip on Low Steam Generator Level Due to Feedwater Flow Control Valve Closure

Dear Commissioners and Staff:

In accordance with 10 CFR 50.73(a)(2)(iv)(A), PG&E is submitting the enclosed licensee event report regarding an automatic reactor trip when Steam Generator 1-1 Inlet Flow Control Valve FCV-510 failed closed.

This event did not adversely affect the health and safety of the public.

Sincerely,

For DHO

David H. Oatley

mrb/2246/N0002147
Enclosure
cc/enc: Ellis W. Merschoff
David L. Proulx
Girija S. Shukla
Diablo Distribution
INPO

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LICENSEE EVENT REPORT (LER)

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TITLE (4)

Automatic Reactor Trip on Low Steam Generator Level Due to Feedwater Flow Control Valve Closure

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
6	3	2002	2002	- 0 0 4 -	0 0	08	02	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 (SPECIFY IN ABSTRACT BELOW AND IN TEXT, NRC FORM 366A)
0 8 6	

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	P M	B 0 4 5						

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 June 3, 2002, at 1302 PDT, while Unit 1 was in Mode 1 (power operation) at 86 percent power, Unit 1 experienced an automatic reactor trip due to low-low steam generator (SG) water level in SG 1-1. Feedwater flow was terminated to SG 1-1 when the feedwater flow control valve, FCV-510, failed closed. The valve failed closed, as designed, on loss of air when a valve-mounted air test connection broke off after contacting structural steel during thermal expansion of the feedwater piping. The test connection was reinstalled in a configuration to prevent contact with structural steel during thermal expansion. Similar installations were visually inspected during the subsequent power ascension to assure there were no other problems resulting from thermal movement of piping.

The root cause of the event was a deficiency in the design-change and work-control processes to consider configuration control of trip-sensitive instruments that are mounted on plant piping subject to thermal expansion during power ascension.

As corrective action to prevent recurrence, PG&E will identify trip-sensitive components on thermally expanding piping and include provisions in the design change and work-control processes to consider their configuration during modification and maintenance.

TEXT

I. Plant Conditions

Unit 1 was in Mode 1 (Power Operation) at 86 percent power.

II. Description of Problem

A. Background

One main feedwater isolation valve (MFIV) [JM][ISV] and one main feedwater regulating valve (MFRV) [JM][FCV] and MFRV [JM][FCV] bypass valve are located on each main feedwater [SJ] line outside containment. The MFIVs and MFRVs are located upstream of the auxiliary feedwater (AFW) [BA] injection point so that AFW may be supplied to the steam generators (SG) [SB] following MFIV or MFRV closure.

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

FCV-510 [SJ][FCV] is the MFRV that supplies feedwater to SG 1-1. It receives a 3 to 15 psi input signal for full valve travel. The test connection port that failed is positioned 90 degrees from the orientation of the input signal port. The entire valve mechanism is connected to a feedwater pipe that thermally expands as power increases.

B. Event Description

On June 3, 2002, at 1301 PDT, a feed flow less than steam flow alarm was received in the Control Room for SG 1-1. The control operator was unsuccessful in reopening FCV-510, and bypass valve FCV-1510 was opened.

At 1302 PDT, the shift foreman gave the direction to manually trip the reactor. Before the reactor was manually tripped, the reactor automatically tripped on SG 1-1 Low-Low level.

A walkdown was then conducted, which revealed that FCV-510 had failed closed due to a break in the root thread of the input gauge port on the positioner. Technicians concluded that the break was due to a bending

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(moment) force applied by thermal expansion of the piping when the connection began to press against the structural steel.

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

None.

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

The event was immediately apparent to plant operators from alarms and indications received in the control room due to SG 1-1 feedwater flow being less than steam flow and decreasing SG level.

F. Operator Actions

Operators placed FCV-510 in manual, but attempts to open the valve were unsuccessful because the instrument air signal to the manifold was then venting directly to atmosphere. When FCV-510 did not respond, and SG 1-1 water level continued to decrease, operators opened bypass valve FCV-1510.

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 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 of the trip was the loss of input signal air to FCV-510 because of a fractured inlet port on the positioner manifold (PO-510). The fractured inlet port resulted from an overload failure due to a bending force.

Tubing fittings, length, orientation, and position indicate that contact with structural support steel occurred. The test connection broke due to a bending (moment) force as the feedwater pipe expanded. The break was a typical aluminum overload failure at the thread root with no corrosion, manufacturing or over-torqued fitting defects found.

B. Root Cause

The root cause of the event was a deficiency in the design-change and work-control processes to consider configuration control of trip-sensitive instruments that are mounted on plant piping that is subject to thermal expansion during power ascension.

IV. Assessment of Safety Consequences

A reactor trip due to a loss of normal feedwater is a previously analyzed Condition II event described in the Final Safety Analysis Report (FSAR) Update, Section 15.2.8, "Loss of Normal Feedwater." A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur. Significant loss of water from the Reactor Coolant System (RCS) could conceivably lead to core damage. If the plant is tripped before the SG heat transfer capability is reduced, the primary system variables never approach a departure from nucleate boiling condition. The FSAR Update analysis shows that following a loss of normal feedwater, an AFW supply of a total of 410 gpm to two SGs is capable of removing the stored and residual heat, thus preventing either over-pressurization of the RCS or loss of water from the RCS.

There were no safety consequences involved in this event because FCV-510 performed its required function by failing closed. The SG water level dropped to the setpoint for the automatic reactor trip, the reactor automatically tripped, and three of three auxiliary feedwater pumps started as designed.

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.

V. Corrective Actions

A. Immediate Corrective Actions

The damaged positioner manifold was replaced and tubing reoriented.

B. Corrective Actions to Prevent Recurrence

1. PG&E will identify trip-sensitive components on thermally expanding piping.

2. PG&E will include provisions in the design-change and work-control processes to consider the configuration of sensitive components during modification and maintenance.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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VI. Additional Information

A. Failed Components

Control signal/test connection on the PO-510 manifold [SJ][PM] of FCV-510.

B. Previous Similar Events

1. Licensee Event Report (LER) 2-85-022-00, "Reactor Trip Resulting from a MFRV Closure Due to a Loose Wiring Connection," DCL-86-008, dated January 20, 1986, reported a similar failure. 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 corrective actions from that LER would not have prevented the event described in this current LER.

2. LER 2-2002-002, "Unit 2 Manual Reactor Trip Due to Loss of Main Feedwater to a Steam Generator," DCL-02-077 dated July 10, 2002, reported a failure of the solenoid valve for FCV-540. The 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 MFRV FCV-540 to close. Investigations indicated the SV failed due to thermal fatigue. The corrective actions from that LER would not have prevented the event described in this current LER.