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AUTH.NAME AUTHOR AFFILIATION
HUG,M.T. Pacific Gas & Electric Co.
SHIFFER,J.D. Pacific Gas & Electric Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 90-015-00:on 901208,Unit 1 in mode 2 (startup) at 2% reactor power,steam generator exceeded 67% narrow range level setpoint.This ESF actuation occurred during transfer of feedwater to main feedwater level.W/910107 ltr.

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Pacific Gas and Electric Company
77 Beale Street
San Francisco, CA 94106
415/973-4684
TWX 910-372-6587
James D. Shiffer
Senior Vice President and
General Manager
Nuclear Power Generation

January 7, 1991

PG&E Letter No. DCL-91-003

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Re: Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
Licensee Event Report 1-90-015-00
ESF Actuation, P-14 (High-High Steam Generator Water Level), Due to Feedwater Regulating and Bypass Valves Leakage

Gentlemen:

Pursuant to 10 CFR 50.73(a)(2)(iv), PG&E is submitting the enclosed Licensee Event Report regarding a high-high steam generator water level signal which initiated a P-14 trip, resulting in an engineered safety feature (ESF) actuation.

Sincerely,

J. D. Shiffer

cc: A. P. Hodgdon
J. B. Martin
P. J. Morrill
P. P. Narbut
H. Rood
CPUC
Diablo Distribution
INPO

DC1-90-OP-N083

Enclosure

5151S/0085K/JHA/2246

IE22
1/1

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 6

TITLE (4) UNIT 1 ESF ACTUATION, P-14 (HIGH-HIGH STEAM GENERATOR LEVEL), DUE TO FEEDWATER REGULATING AND BYPASS VALVES LEAKAGE

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 NAMES	DOCKET NUMBER (S)
										0 5 0 0 0
12	08	90	90	- 0 1 5 -	0 0	01	07	91		0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 2

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (11)

X 10 CFR 50.73(a)(2)(iv)

OTHER - (Specify in Abstract below and in text, NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
MARTIN T. HUG, REGULATORY COMPLIANCE SENIOR ENGINEER	805	545-4005

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES (if yes, complete EXPECTED SUBMISSION DATE)	[X] NO				

ABSTRACT (16)

On December 8, 1990, at 0031 hours PST, with Unit 1 in Mode 2 (Startup) at approximately 2 percent reactor power, steam generator (SG) 1-3 level exceeded the 67 percent narrow range level setpoint initiating a P-14 signal and engineered safety feature (ESF) actuation.

The root cause of the event is leakage through feedwater regulating valve FW-1-FCV-530 and feedwater regulating bypass valve FW-1-FCV-1530. FW-1-FCV-530 was stroke tested during a Unit 1 forced outage in late December 1990, and it was determined that leakage through this valve was due to valve position controller drift. FW-1-FCV-1530 was disassembled and inspected during the Unit 1 forced outage, and it was determined that leakage through this valve was due to a combination of minor valve seat damage and valve position controller drift. Backleakage through check valve FW-1-531 contributed to this event and a slight misalignment in the check valve disc was corrected.

The position controllers for FW-1-FCV-530 and FW-1-FCV-1530 were adjusted to correct for drift during the forced outage. FW-1-FCV-530 exhibited minor leakage during restart, and its position controller will be calibrated during the next refueling outage. The valve seat for FW-1-FCV-1530 was lapped to remove minor irregularities and the valve plug was replaced. The appropriate operating procedure was revised to include steps to check for main feedwater regulating and bypass valve leakage during plant startup.

TEXT (17)

I. Plant Conditions

Unit 1 was in Mode 2 (Startup) at approximately 2 percent reactor power. Plant startup was in progress. The control room operators had just completed the procedural steps to place main feedwater pump (MFP) 1-2 (SJ)(P) in service with pump speed adjusted for a feedwater/main steam (SJ)/(SB) differential pressure of approximately zero psid.

II. Description of Event

A. Event:

On December 8, 1990, at 0031 hours PST, with Unit 1 in Mode 2 (Startup) at 2 percent reactor power, steam generator (SB)(SG) 1-3 exceeded the 67 percent narrow range level setpoint, which initiated a P-14 signal. This engineered safety feature (ESF) actuation occurred during the transfer of the feedwater flow controls (SJ)(FC) from auxiliary feedwater to automatic main feedwater level control and resulted in main feedwater isolation, MFP 1-2 (SJ) trip, and main turbine (TA)(TRB) trip.

The control room operators were in the process of placing the main feedwater system in operation when operators observed that SG 1-3 level was decreasing, with auxiliary feedwater (AFW)(BA) flow to SG 1-3 greater than flow to the other 3 SGs. This problem was initially diagnosed as backleakage through the SG 1-3 inlet check valve, FW-1-531 (SJ)(V), and the feedwater regulating and/or bypass valves, FW-1-FCV-530 (SJ)(FCV) and FW-1-FCV-1530 (SJ)(FCV), respectively. Backleakage through check valve FW-1-531 had been previously experienced during plant conditions when low differential pressures existed across the check valve.

To increase SG 1-3 level, operators increased speed on MFP 1-2 to prevent the backleakage and feed SG 1-3 from the main feedwater system.

The level in SG 1-3 rapidly increased beyond its normal no-load level setpoint. The control room operators attempted to close valve FW-1-FCV-1530 by manually decreasing valve demand, even though the demand was at zero. This action appeared to terminate the initial steam generator level increase, so the decision was made to proceed with the Unit 1 startup.

The main feedwater controls (SJ)(LIK) were placed in automatic and the level in SG 1-3, as indicated on the narrow and wide range level instrumentation (SJ)(LI), immediately began to increase. Wide range level appeared to stabilize shortly into the transient. The control room operators concluded that this level increase was caused by SG swell and since wide range level (SJ)(LI) did not show an increasing

trend, the reactor power was slowly increased. Reactor coolant system (RCS)(AB) temperature was increasing and the steam dumps (SB)(V) opened. The narrow range level increased to 67 percent, causing the P-14 signal to be initiated. This was caused by steam generator overfeeding due to feedwater regulating and bypass valve leakage and subsequent swell during steam dump actuation.

The AFW system remained in service throughout this event, assuring adequate feedwater flow to the SGs.

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

None.

C. Dates and Approximate Times for Major Occurrences:

1. Dec. 8, 1990, 0025 hrs: MFP 1-2 is placed in service.

2. Dec. 8, 1990, 0031 hrs: Event\Discovery date. P-14 ESF signal occurred.

3. Dec. 8, 1990, 0146 hrs: Four hour non-emergency report was made to the NRC in accordance with 10 CFR 50.72.

D. Other Systems or Secondary Functions Affected:

None.

E. Method of Discovery:

The event was apparent to control room personnel due to control room alarms and indications.

F. Operator Actions:

The operators stabilized the plant in Mode 2, verified that the SG levels were being maintained by the AFW system, and reestablished plant conditions to those existing prior to the P-14 signal.

G. Safety System Responses:

1. MFP 1-2 tripped.

2. The main turbine tripped.

3. The main feedwater isolation valves (SJ)(ISV) closed.

4. The main feedwater regulating valves and regulating bypass valves closed.

III. Cause of the Problem

A. Immediate Cause:

The immediate cause for the SG high-high level signal was that SG 1-3 level exceeded the P-14 setpoint due to overfill and subsequent swell during startup.

B. Root Cause:

The root cause of the event is leakage through feedwater regulating valve FW-1-FCV-530 and feedwater regulating bypass valve FW-1-FCV-1530. FW-1-FCV-530 was stroke tested during a Unit 1 forced outage in late December 1990, and it was determined that leakage through this valve was due to valve position controller drift. FW-1-FCV-1530 was disassembled and inspected during the Unit 1 forced outage, and it was determined that leakage through this valve was due to a combination of minor valve seat damage and valve position controller drift.

C. Contributing Cause:

Backleakage through SG 1-3 inlet check valve FW-1-531 due to a slight misalignment of the check valve disc contributed to this event since the leakage initiated the level transient in SG 1-3.

IV. Analysis of the Event

The high-high SG water level P-14 protective interlock is designed to protect the main turbine from water induction and subsequent erosion caused by SG overfill. This feature is designed for equipment protection only, and functioned as designed. Leakage past the feedwater regulating valves is an analyzed event and is bounded by the analysis for a failed full-open feedwater regulating valve. Therefore, the health and safety of the public were not adversely affected by this event.

V. Corrective Actions

A. Immediate Corrective Actions:

None.

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B. Corrective Actions to Prevent Recurrence:

1. Operating procedure L-3, "Secondary Plant Startup," was revised to include steps to check for main feedwater regulating and bypass valve leakage during startup, and to provide guidance for taking action to deal with any leakage identified.

2. The following corrective actions were taken during a Unit 1 forced outage in late December 1990:

   a. The position controller for the feedwater regulating valve FW-1-FCV-530 was adjusted to correct for drift. However, the valve exhibited minor leakage during restart and, in accordance with revised operating procedure L-3, the valve was isolated until the Unit reached 12 percent power. During the next refueling outage, the position controller will be calibrated.

   b. Feedwater regulating bypass valve FW-1-FCV-1530 was disassembled and the valve seat was lapped to remove minor irregularities and the valve plug was replaced. The valve was reassembled and given a successful leak check. The valve position controller was adjusted to correct for drift.

   c. SG 1-3 inlet check valve FW-1-531 was disassembled and inspected, and a slight misalignment of the check valve disc was corrected.

VI. Additional Information

A. Failed Components:

None.

B. Previous LERs on Similar Events:

1. LER 2-90-007, "Unit 2 ESF Actuation, P-14 (High-High Steam Generator Level) Due To Unanticipated Steam Generator Swell At Low Power Levels." These P-14 trips occurred during main turbine testing with the SG level controls in manual. Operators raised the SG levels high in the band in anticipation of shrink with main turbine intentional trip, but the SGs were fed at too high a rate and the warming water swelled to the P-14 setpoint. An Operations Incident Summary was issued to sensitize operators during future similar evolutions.

2. LER 1-87-025, "High Steam Generator Water Level Main Turbine Trip and Feedwater Isolation During Startup Due To Lack Of

TEXT (17)

Guidance For Operators On Proportional Integral Controllers." This P-14 trip occurred due to a lack of guidance for operators when transferring from manual to automatic SG level control. An Operations Memorandum was issued at that time to provide operator guidance.

C. Check valve FW-1-531 backleakage is being evaluated in a separate problem report.