50-275/323-OCA-2 I-MFP-155 vab. 1727

Pacific Gas and Electric Company

170G20

MFP Exhibit 155
8/21/93 DOLLIE FEISEL Reptr.

'93 OCT 28 P5:51

May 17, 1991

PG&E Letter No. DCL-91-134

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 1-91-002-01
Reactor Trip on Steam Generator Low Level with Steam Flow/Feedwater Flow Mismatch due to Personnel Error

Gentlemen:

Pursuant to 10 CFR 50.73(a)(2)(iv), PG&E is submitting the enclosed Licensee Event Report (LER) concerning a reactor trip on steam generator low level coincident with steam flow/feedwater flow mismatch due to personnel error. This revision is being submitted to present the results of investigations and corrective actions regarding four nonsafety-related equipment failures following the trip.

This event has in no way affected the health and safety of the public.

Sincerely,

J. D. Shiffer

cc: Ann P. Hodgdon
John B. Martin
Phillip J. Morrill
Paul P. Narbut
Harry Rood
CPUC
Diablo Distribution
INPO

DC1-91-WP-N012

Enclosure

5366S/0085K/ALN/2246

NUCLEAR REGULATORY COMMISSION
Docket No. 50-275-OLA Official Exh. No. MFP-155
In the matter of PACIFIC GAS and ELECTRIC CO
Staff ___ IDENTIFIED ✓
Applicant ___ RECEIVED ✓
Intervenor ✓ REJECTED ___
Cont'g Off'r ___
Contractor Ann Riley & Assoc's DATE 8-21-93
Other ___ Witness ___
Reporter Dollie Feigel

RECEIVED
MAY 20 1991
P-S
Corresp. Control Center

9401070042 930821
PDR ADOCK 05000275
G PDR

LICENSEE EVENT REPORT (LER) 170G20

FACILITY NAME (1) DIABLO CANYON UNIT 1	DOCKET NUMBER (2) 0 5 0 0 0 2 7 5	PAGE (3) 1 OF 8

TITLE (4) REACTOR TRIP ON STEAM GENERATOR LOW LEVEL WITH STEAM FLOW/FEEDWATER FLOW MISMATCH DUE TO PERSONNEL ERROR

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
02	01	91	91	- 0 0 2 -	0 1	05	17	91		0 5 0 0 0

OPERATING MODE (9) 1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (11)
POWER LEVEL (10) 0 9 0	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, SENIOR REGULATORY COMPLIANCE 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 February 1, 1991, at 0909 PST, with Unit 1 operating at approximately 90 percent power, a Unit 1 reactor trip occurred due to Steam Generator (SG) 1-4 low level coincident with a steam flow/feedwater flow mismatch. A four-hour, non-emergency report was made in accordance with 10 CFR 50.72 (b)(2)(ii) on February 1, 1991, at 1149 PST.

The root cause of the event was determined to be personnel error (cognitive). Personnel erecting scaffolding in the area of feedwater regulating valve FCV-530 inadvertently closed valve AIR-I-1-1041, which isolated instrument air supply to feedwater regulating and bypass valves FW-1-FCV-530, 540, 1530, and 1540. The regulating and bypass valves failed closed on loss of instrument air, isolating feedwater flow to two SGs. SG levels decreased and the unit tripped.

The corrective actions include: (1) issuance of a procedure to secure sensitive air valves against accidental closure; (2) issuance of a maintenance policy requiring that all work to be performed prior to the outage must be on the approved daily schedule; (3) issuance of a Work Planning policy on marking work packages; and (4) revision of Administrative Procedure (AP) C-59, "Elevated Work Structures," to clarify responsibilities and restrictions for erection and removal of scaffolding.

I. Plant Conditions

Unit 1 was in Mode 1 (Power Operation) at approximately 90 percent power.

II. Description of Event

A. Event:

On December 5, 1990, Maintenance Planning requested that scaffolding be provided to support the inspection and repair of Main Feedwater Regulating Valve (SJ)(FCV) FW-1-FCV-530, which was scheduled for inspection during the upcoming Unit 1 refueling outage.

In December 1990 and January 1991, the work package for erection of the scaffolding was prepared. The work package contained the work order, elevated work structure request form (scaffolding request), component information, and an area map showing the location of equipment. However, the scaffolding request was not fully completed. The dates by which the scaffolding was required were not filled in and the request was not marked as "outage," "pre-outage," or "non-outage" work.

On January 28, 1991, Operations signed off on their review of the scaffolding request. In accordance with Administrative Procedure (AP) C-59, "Elevated Work Structures," Operations reviews scaffolding requests for their potential effect on equipment. Although AP C-59 requires a sketch on the scaffolding request form, it does not specify the level of detail required. The sketch on this scaffolding request form did not provide sufficient detail to show the exact location where the scaffolding was to be erected. Operations did not recognize that the area designated for scaffolding erection included FW-1-FCV-530.

Outage management had scheduled the scaffolding for erection on February 4, during the outage and after plant shutdown. Since erection was scheduled to be performed after plant shutdown, the work package did not appear on the daily (pre-outage) schedule. Work packages on the daily schedule receive additional plant management review and approval before being issued for work.

On January 30, 1991, prior to the scheduled time and during plant operation, erection of scaffolding began near valve AIR-I-1-1041, which is the instrument air isolation valve (LD)(ISV) for valves FW-1-FCV-530 and 540 and Bypass Feedwater Regulating Valves FW-1-FCV-1530 and 1540.

On February 1, 1991, three contractor personnel were setting planks on the scaffolding structure around FW-1-FCV-530 when they heard a hissing noise followed by a loud bang. At approximately 0908 PST, the control room received steam generator (AB)(SG) level deviation alarms for Steam Generators (SGs) 1-3 and 1-4. Operators responded by attempting to manually open valves FW-1-FCV-530 and 540 from the control room.

On February 1, 1991, at 0909 PST, a Unit 1 reactor trip occurred due to SG 1-4 low level coincident with a steam flow/feedwater flow mismatch. A four-hour, non-emergency report was made in accordance with 10 CFR 50.72 (b)(2)(ii) on February 1, 1991, at 1149 PST.

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

None.

C. Dates and Approximate Times for Major Occurrences:

1. January 30, 1991: Erection of scaffolding began.

2. February 1, 1991, at 0909 PST: Event/Discovery date. Reactor trip due to SG 1-4 low level with steam flow/feed flow mismatch while contractor personnel were setting planks on scaffolding structure.

3. February 1, 1991, at 1149 PST: The 4-hour non-emergency report to the NRC required by 10 CFR 50.72 (b)(2)(ii) was made.

D. Other Systems or Secondary Functions Affected:

1. Circulating Water Pump (KE)(P) 1-1

Circulating Water Pump (CWP) 1-1 failed to automatically restart following the 12kV bus power transfer to startup power. All CWP 1-1 automatic transfer logic circuits were tested, but no defective components were identified.

2. 25kV Motor-Operated Disconnect (EL)(MOD)

The 25kV Motor-Operated Disconnect (MOD) switch did not open fully. In preparation for outage work on the MOD, plastic sheeting had been securely taped to the MOD manual drive shaft and switch actuating shaft. When the MOD was signaled to open, the turning drive shaft began to wrap the attached plastic around the shaft. The shaft continued to turn until enough plastic had accumulated to contact the nearby switch actuating shaft. Continued wrapping forced the switch actuating shaft outward (horizontally), simulating "stop logic," which de-energized the MOD power and control circuit.

3. Control Rod Drive Mechanism Cooling Fan (CD)(FAN) E-13

Control Rod Drive Mechanism (CRDM) Cooling Fan E-13 failed to automatically restart following the power transfer. Electrical Maintenance believes that a combination of reduced component part clearances due to age-related shrinkage and dimensional changes resulting from elevated operating temperatures caused the magnetic starter to fail.

4. Main Turbine Stop Valve (SB)(ISV) FCV-145

Main Turbine Stop Valve FCV-145 did not fully close. Stroke time was determined to be very slow (in excess of two minutes). A brass bushing on the actuator spring can assembly was found to be very dry and scored. After replacement of the bushing, stroke times were acceptable.

E. Method of Discovery:

The event was immediately apparent to plant operators due to alarms and indications received in the control room.

F. Operator Actions:

Operators noted that valves FW-1-FCV-530 and 540 were closed and attempted to manually open them from the control room. No response was observed. Operators were about to manually attempt to open bypass valves FW-1-FCV-1530 and 1540 from the control room when the reactor tripped. (However, since closure of AIR-I-1-1041 isolated instrument air to the bypass valves as well as the regulating valves, the bypass valves would not have opened either.)

Following the trip, operators stabilized the plant in Mode 3 (Hot Standby) using Emergency Operating Procedures (EPs) E-0, "Reactor Trip or Safety Injection," and E-0.1, "Reactor Trip Response." Operators restarted CWP 1-1, manually opened the MOD switch, restarted CRDM fan E-13, and locally verified the position of FCV-145. Immediately downstream of FCV-145, Turbine Governor Control Valve (SB)(FCV) MS-1-FCV-141 had closed on the turbine trip, so no other immediate operator action was necessary regarding FCV-145.

G. Safety System Responses:

1. The reactor trip breakers (JC)(BKR) opened.

2. The main turbine (TA)(TRB) tripped.

3. The motor-driven and turbine-driven auxiliary feedwater pumps (BA)(P) started.

III. Cause of the Event

A. Immediate Cause:

The immediate cause of the unit trip was valves FW-1-FCV-530 and 540 failing closed upon loss of instrument air.

B. Root Cause:

The root cause of the unit trip was determined to be personnel error (cognitive). Following the trip, an immediate investigation determined that instrument air isolation valve AIR-I-1-1041 was in a closed position. The investigation concluded that the valve had been inadvertently closed by the personnel who were erecting scaffolding.

C. Contributory Causes:

1. Prior to the commencement of outage activities, Maintenance management decided that outage-related work should not be performed before the outage unless it was on the daily schedule. Management expectations regarding this decision were not met, since the decision was implemented only within the outage scheduling group, and not emphasized to all other applicable plant personnel. Pre-outage work is screened by plant management, and work that may affect safety-related components is not allowed to be put on the daily schedule. However, because the personnel building the scaffolding were unaware of the management decision, they started the work early even though it was not on the daily schedule.

2. AP C-59 did not provide adequate guidance to preclude the erection of scaffolding around sensitive plant equipment during operation. Specifically, AP C-59 did not define sensitive equipment that would require additional review, did not adequately identify components to be worked, did not specify responsibilities for completing the scaffolding request form, and did not identify the allowed start date or plant modes for the work.

3. The AIR-I-1-1041 valve handle requires a 90 degree rotation to move to the closed position and rotates easily. Therefore, though the handle is near a wall and not easily accessible, it can be closed accidentally.

IV. Analysis of the Event

A. Safety Analysis:

Inadvertent loss of all main feedwater flow is a Condition II event, which has been analyzed in Final Safety Analysis Report Update Section 15.2.8, "Loss of Normal Feedwater." The reactor trip and automatic start of the auxiliary feedwater pumps ensured that an adequate supply of water was provided to the SGs to provide for the cooldown of the reactor per design. All safety-related equipment functioned as designed.

Failure of a CWP to automatically restart upon transfer to offsite power could render the condenser steam dump valves unavailable to remove excess heat following a turbine trip. A previous Westinghouse analysis, in support of raising the setpoint of the reactor trip on turbine trip permissive from 10 percent to 50 percent thermal power, took credit for condenser steam dump valve operation. However, unavailability of the condenser steam dump system during a turbine trip has been analyzed in Final Safety Analysis Report Update Section 15.2.7, "Loss of External Electrical Load and/or Turbine Trip." The reactor would trip on the turbine trip if above 50 percent power, or on reactor coolant system conditions if below 50 percent. Pressurizer spray and pressurizer power-operated relief valves are available, though the pressurizer safety valves and steam generator safety valves alone are adequate to protect the reactor coolant system and steam generators against overpressure.

Based on the above analysis, the conditions of this event were bounded by previous safety analyses. Consequently, the inadvertent isolation of main feedwater and subsequent reactor trip did not adversely affect the health and safety of the public.

V. Corrective Actions

A. Immediate Corrective Actions:

1. The plant was stabilized in Mode 3 using EPs E-0 and E-0.1.

2. CWP 1-1 and CRDM fan E-13 were restarted and the MOD was manually opened.

B. Corrective Actions to Prevent Recurrence:

1. Reactor Trip

a. Operating Procedure (OP) K-1:IV, "Compressed Air System - Preventing Inadvertent Operation of Instrument Air Valves," was issued to require that the handles of sensitive air valves be tie-wrapped to secure the valves against accidental closure.

b. A maintenance policy was issued requiring that all authorized pre-outage work must be shown as a line item on the daily schedule to ensure that proper reviews are performed.

c. Work Planning Department Policy No. 11 was issued requiring work package folders to be labeled as follows: (1) Work package folders for work to be performed on the daily schedule shall have no schedule designated on the cover; and (2) Work package folders for work to be performed during a planned refueling outage shall have the outage designated on the cover (e.g. 1R4, 2R4).

d. AP C-59 will be revised to clarify responsibilities and restrictions for erection and removal of scaffolding. Training on the revision will be conducted in accordance with AP E-4 Supplement 6, "Procedure Review and Revision Control."

2. CWP 1-1

With no CWP available, the 40 percent condenser steam dump valves are not available to remove heat following a turbine trip. A previous Westinghouse analysis, in support of raising the setpoint of the reactor trip on turbine trip permissive from 10 percent to 50 percent thermal power, took credit for CWP and steam dump valve operation. The analysis showed that there was no increase in the probability of challenges to the pressurizer power-operated relief valves (PORVs) during a turbine trip at 50 percent power if the steam dump valves are available. Challenges to the PORVs increase the probability of a small break LOCA resulting from a stuck open PORV.

However, potential unavailability of the CWPs and condenser steam dump valves does not affect the ability of the PORVs and other safety-related equipment to perform their safety functions, if needed.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
DIABLO CANYON UNIT 1	0 5 0 0 0 2 7 5	91	- 0 0 2	- 0 0	8 OF 8

TEXT (17)

Unit 1 CWPs have failed to automatically restart following three recent Unit 1 reactor trips. As a compensatory measure to prevent challenges to the PORVs following a turbine trip, the trip permissive setpoint has been set to nominal 15 percent of rated thermal power. This compensatory measure is documented in Justification for Continued Operation (JCO) 91-02, which will remain in effect until the reliability of a Unit 1 circulating water pump to restart after a transfer from auxiliary power to start-up power is shown. Action has been initiated to investigate and resolve this problem.

3. 25kV MOD

Signs were installed near the MOD drive shafts and chain linkages to warn against placing equipment near the moving parts.

4. CRDM Fan E-13

Starters of the type in CRDM Fan E-13 have been replaced.

5. Main Turbine Stop Valve FCV-145

A preventive maintenance activity will be initiated to periodically lubricate the brass bushing in hydraulic actuators of the type used in FCV-145.

VI. Additional Information

A. Failed Components:

None.

B. Previous LERs:

None.

C. Additional Information:

Historically, the most common reactor trip initiators have been associated with SG level control, turbine trips, main feedwater pumps, maintenance, and procedure or design deficiencies. PG&E has an ongoing Trip Reduction Program to evaluate further plant improvements to reduce reactor trips.