NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

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 0 4

TITLE (4)
MANUAL REACTOR TRIP

EVENT DATE (5)			LER NUMBER (6)					REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 2	1 7	8 5	8 5	-	0 1 1	-	0 1	0 5	0 2	8 5		0 5 0 0 0
												0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)					
POWER LEVEL (10)	0 4 8	20.402(b)		20.405(c)	X	50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)	
		20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)	
		20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
DAVID P. SISK, REGULATORY COMPLIANCE ENGINEER	AREA CODE 8 0 5 — 5 9 5 - 7 3 5 1

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	

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

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

At 0415 PST, February 17, 1985, with Unit 1 in Mode 1 (48% power), plant operators initiated a manual reactor trip in response to the automatic shutdown of main feedwater pumps 1-1 and 1-2 from excessive thrust bearing wear indication. In accordance with procedures, the plant was stabilized in Mode 3 (hot standby).

The thrust bearing wear indicators were determined to be conservatively set (I&C verified) to allow for readjustment as operational and transient data become available. No damage was observed.

To prevent recurrence, the thrust bearing wear detection system has been reset to new values recommended by the vendor (Westinghouse).

Subsequent to this reactor trip, several events took place in the balance of plant (BOP). The most significant was a water hammer event in one of the main feedwater bypass lines. The feedwater bypass lines are not within the ASME Section XI Code boundary. This supplemental report provides a detailed account of the events that occurred in the BOP.

IE22

NRC Form 366
(9-83)

1/1

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
DIABLO CANYON UNIT 1	0 5 0 0 0 2 7 5	8 5	— 0 1 1	— 0 1	0 2 OF 0 4

TEXT (If more space is required, use additional NRC Form 366A's) (17)

At 0415 PST, February 17, 1985, with Unit 1 in Mode 1 (at 48 percent power), plant operators initiated a manual reactor trip in response to the automatic shutdown of main feedwater pumps 1-1 and 1-2 (SJ)(P) from excessive thrust bearing wear indication.

The calibration of the thrust bearing wear detection system was verified by Instrumentation and Controls technicians. The setpoints of the feedwater pumps thrust bearing wear indicator had been set to recommended specifications, however, due to a lack of transient thrust data, these specifications were too conservative. Both thrust bearings were inspected with no damage observed.

To prevent recurrence, the setpoints of the main feedwater pumps thrust bearing wear detection system were reset to new values recommended by the manufacturer (Westinghouse). In addition, a 0.5 second time delay will be added to accommodate transient conditions.

Subsequent Events (Low pressure turbine rupture disc perforation, hydrogen cooler leak, water hammer, and snubber damage.)

Following the reactor trip, the Operations staff was in the process of stabilizing the balance of plant (BOP), establishing a minimum flow path for the condensate and condensate booster pumps (SD) (P), and reestablishing secondary chemistry. To achieve these results, it was necessary to jumper to open the high temperature (320°F) interlocks provided in FCV-420 (SD) (V) (feedwater to main condenser recirculation valve). Opening FCV-420 allowed condensate at 370°F to enter the atmospheric condenser (SG) (COND) and flash to steam. The low pressure turbine casing overpressurized, resulting in two rupture disks (TA) (RPD) perforating and five more exhibiting signs of stress.

Water was discovered in the drain piping of the hydrogen system. It was believed to come from the hydrogen coolers. The hydrogen gas pressure was observed to increase. To facilitate repair of the hydrogen cooler (TK) (CLR) leakage and lower the pressure, the Operations staff commenced venting the hydrogen gas. During the hydrogen gas venting, the hydrogen gas dryer was reactivating the desiccant. The higher pressure venting hydrogen entered a common vent header in the gas dryer (TK) (DRY). The energized gas dryer ignited a small volume of hydrogen gas, which resulted in minor damage to the power supply cabinet door (TK) (CDR).

At the completion of the hydrogen gas purge from the generator, the running condensate and condensate booster pumps were stopped. This action, with feedwater valve FCV-420 still open, resulted in a rapid depressurization of the feedwater piping and the stagnant high temperature water downstream of where the recirculating line through FCV-420 comes off, flashed to steam. The resulting water hammer damaged 8 of the 10 snubbers (SJ) (SNB) installed on the bypass line associated with feedwater line 1-2.

Subsequent to the above events, the two perforated and five damaged rupture disks in the low pressure turbine casing were replaced. Examination and hydrotesting of the hydrogen coolers indicated no structural problems and the hydrogen cooler leak was repaired by replacing the faulty gaskets. To prevent future hydrogen combustion, warning signs have been posted cautioning operators not to vent hydrogen in parallel with drying the hydrogen desiccant.

Snubber Inspection

On February 17, 1985, a visual inspection of snubbers outside containment did not reveal any inoperable snubbers. On March 7 and 8, 1985, mechanical inspection of snubbers on the 1-2 feedwater bypass line revealed eight inoperable snubbers. By March 9, 1985, the eight inoperable snubbers on the 1-2 feedwater bypass had been replaced and declared operable. All of the feedwater bypass line snubbers on the other three loops and four snubbers on the loop 2 main feedwater line were mechanically inspected and found to be operable. The balance of the main feedwater mechanical snubbers outside containment have been inspected, and no damage was found. All snubber supporting structures were visually inspected and determined to be functional.

The Action Statement for Technical Specification Section 3.7.7.1 requires restoration of operability and an engineering evaluation within 72 hours, or systems attached to inoperable snubbers be declared inoperable. The loop 2 feedwater regulating valve bypass line and associated valve, FCV-1520, were used during the unit startup on February 25 and 26, 1985, although this line was technically inoperable as of February 20, 1985 in accordance with the Action Statement of Technical Specification 3.7.7.1. Appropriate action and engineering evaluations were performed within 72 hours of discovery, and the surveillance requirement of Technical Specification 4.7.7.1d (transient event inspections) was performed well within the allowed 6-month period following the transient event.

Engineering Safety Evaluation

According to the engineering evaluation's, the ramifications of operating the plant with these eight snubbers inoperable are as follows:

1. System Interaction Program (SIP)

   These lines were evaluated for SIP considerations prior to any permanent supports being installed. Inoperability of the snubbers is not a SIP concern.

2. Jet Impingement/Pipe Break

   Breaks at the nozzle to the main feedwater lines have been considered. This review was extended to consider any potential break in the bypass line's entire length. No impact on plant safety was indicated.

3. Safety-related Piping Integrity

Because of the relative size difference between the bypass line and main feedwater line, and distance between the ASME Section XI Code break and the bypass line, failure of the bypass line supports would not have a detrimental effect on the integrity of the ASME Class II feedwater line.

4. Operability of the Feedwater Regulating Bypass Valve

If the bypass regulating valve were to fail to close during a seismic event because of the inoperable snubbers, no reduction in plant safety would occur. Failure of a main regulating valve to close during a main steam line break (MSLB) is already considered in the plant safety analyses. Failure of a bypass regulating valve to close would be a less severe event due to its smaller size. On an MSLB, one of the scenarios is that a regulating valve fails open. The main steam isolation valves (MSIVs) will close 10 seconds after the break with the feedwater containment isolation valves closing within 1 minute after initiation of a break. The case of an MSLB occurring during startup and shutdown with an inoperable bypass regulating valve and a MSIV failure (a single failure) on the broken loop can be easily recovered due to a lower decay heat, lower flow rate to the steam generator, and the feedwater containment isolation valve can still be closed within 1 minute to isolate feedwater. The fact that the bypass regulating valves are used only for a brief time during startup and shutdown means that, when they are open, their operability is verified shortly thereafter when the valve is closed at greater than 20% power.

Based on the engineering evaluation, it was concluded that no unreviewed safety conditions were created by plant operation with inoperable snubbers in feedwater bypass line 1-2. Instructions have been issued to all operators restricting the use of FCV-420.

PACIFIC GAS AND ELECTRIC COMPANY

PG&E — 77 BEALE STREET • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-4211 • TWX 910-372-6587

JAMES D. SHIFFER
VICE PRESIDENT
NUCLEAR POWER GENERATION

May 2, 1985

PGandE Letter No.: DCL-85-178

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

Re: Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
Licensee Event Report 85-011-01
Manual Reactor Trip

Gentlemen:

Pursuant to 10 CFR 50.73(a)(2)(i) and (iv), PGandE is submitting the enclosed revision to a Licensee Event Report concerning a manual reactor trip in response to the automatic shutdown of both main feedwater pumps. This revision provides greater details of balance-of-plant events that occurred after the reactor trip.

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

Kindly acknowledge receipt of this material on the enclosed copy of this letter and return it in the enclosed addressed envelope.

Sincerely,

for J. D. Shiffer

Enclosure

cc: J. B. Martin
Service List

IE22
1/1