50-275/323-OLA-2 A-28 PG&E Exhibit 28 8/23/93 Dollie Feigel Reptr 026.1429

Pacific Gas and Electric Company
77 Beale Street
San Francisco, CA 94106
415/973-4684

Gregory M. Rueger
Senior Vice President and
General Manager
Nuclear Power Generation

201951

'93 OCT 28 P6:25

January 11, 1993

PG&E Letter No. DCL-93-006

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 1-92-009-01
Dose Limits Potentially Exceeded from Chemical and Volume Control System Valve Diaphragm Leakage Due to Thermally Induced Degradation

Gentlemen:

Pursuant to 10 CFR 50.73 (a)(2)(ii)(B), PG&E is submitting the enclosed revision to Licensee Event Report (LER) 1-92-009-00 regarding identified leakage through a valve diaphragm that could have resulted in the control room and offsite dose limits being exceeded during the design-basis recirculation phase of a loss-of-coolant accident. This revision is being submitted to report the root cause and corrective actions, and supercedes the previous LER in its entirety.

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

Sincerely,

Gregory M. Rueger

cc: Ann P. Hodgdon
John B. Martin
Mary H. Miller
Sheri R. Peterson
CPUC
Diablo Distribution
INPO

DC2-91-TN-N087

Enclosure

1069S/85K/JCN/2246

NUCLEAR REGULATORY COMMISSION
Docket No. 50-275-OLA Official Exh. No. PG&E-28
In the matter of PACIFIC GAS and ELECTRIC CO
Staff _____ IDENTIFIED ✓
Applicant _____ RECEIVED ✓
Intervenor ✓ _____ REJECTED _____
Contractor Ann Riley & Assocs DATE 8-23-93
Other _____ Witness _____
Reporter Dollie Feigel

LICENSEE EVENT REPORT (LER)

201751

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) DOSE LIMITS POTENTIALLY EXCEEDED FROM CHEMICAL AND VOLUME CONTROL SYSTEM VALVE DIAPHRAGM LEAKAGE DUE TO THERMALLY INDUCED DEGRADATION

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)
06	22	92	92	- 0 0 9 -	0 1	01	11	93		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1
POWER LEVEL (10)	1 0 0

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

X 10 CFR 50.73(a)(2)(ii)(B)

____ OTHER -

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

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	TELEPHONE NUMBER
DAVID P. SISK, SENIOR REGULATORY COMPLIANCE ENGINEER	805	545-4420

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	
X	F E	T H	T 1 8 5	N							

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

ABSTRACT (16)

On June 26, 1992, with Unit 1 in Mode 1 (Power Operation) at 100 percent power, PG&E determined that identified leakage from the chemical and volume control system (CVCS) could potentially cause design-basis dose limits to be exceeded during the recirculation phase of a loss-of-coolant accident (LOCA). A one-hour, non-emergency report was made to the NRC in accordance with 10 CFR 50.72 (b)(1)(ii)(B) on June 26, 1992, at 1549 PDT.

On June 22, 1992, diaphragm valve CVCS-1-547, the emergency borate flow to the volume control tank outlet isolation, was found to be leaking approximately 0.5 gallon per minute. This leakage could have caused 10 CFR 100 and GDC 19 dose limits to be exceeded during a design-basis LOCA.

The root cause of the leakage has been determined to be thermally-induced premature degradation of the valve diaphragm caused by a malfunctioning heat trace controller, resulting in distortion of the diaphragm at the body-to-bonnet joint and breaching of the system pressure boundary.

During the Unit 1 fifth refueling outage, the valve bonnet and diaphragm of CVCS-1-547, as well as the heat trace controllers, were replaced to return the valve to an acceptable configuration. All heat-traced diaphragm valves in the post-LOCA recirculation flow path were inspected and reconfigured as necessary.

I. Plant Conditions

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

II. Description of Event

A. Summary:

On June 22, 1992, diaphragm valve CVCS-1-547 (CB)(V), the emergency borate flow to the volume control tank (VCT)(CB)(TK) outlet isolation in the chemical and volume control system (CVCS)(CB), was found to be leaking approximately 0.5 gallons per minute (gpm) to the auxiliary building (NF) atmosphere.

On June 26, 1992, PG&E determined that identified leakage from the CVCS could potentially cause design-basis dose limits to be exceeded during the recirculation phase of a loss-of-coolant accident (LOCA). A one-hour, non-emergency report was made to the NRC in accordance with 10 CFR 50.72(b)(1)(ii)(B) on June 26, 1992, at 1549 PDT.

B. Background:

Leakage from the post-LOCA recirculation flow path must be limited to meet design-basis dose limits. As specified in the Final Safety Analysis Report (FSAR) Update, the maximum permissible leakage outside of containment from the post-LOCA recirculation loop, while pressurized to post-LOCA pressure, is 0.10 gpm in areas where the plant ventilation exhaust is not filtered by charcoal filters (VF)(FLT) and 0.94 gpm when filtered through charcoal filters (in addition to a postulated residual heat removal pump seal (BP)(SEAL) leakage of 50 gpm).

CVCS-1-547 is located in the boric acid blender (CB)(MIX) room on the 100 foot elevation of the auxiliary building. The boric acid blender room ventilation (VF) exhausts to the plant vent (VL) without passing through charcoal filters. Therefore, any radioactive material that may be released as a result of leakage in this area would be released to the plant vent, which is filtered only by high efficiency particulate air (HEPA) filters (VF)(FLT).

CVCS-1-547 is a manually operated diaphragm valve in the CVCS system. During power operation, this valve normally remains in the open position with system pressure at approximately 23 pounds per square inch, gauge (psig). This valve does not have a safety function to close during either normal or accident conditions. However, this valve does become pressurized as part of the reactor coolant (AB) flow path pressure boundary during the recirculation phase of a LOCA.

This valve also forms part of the flow path for emergency boration. Because the dissolved boron present in the water will precipitate out of

solution at low temperatures, electric heat trace circuitry (FE) is installed to maintain the temperature of lines above 145°F as required by Technical Specifications 3.5.4.2 and 4.1.2.2. Heat tracing is not installed on the bonnet of CVCS-1-547 to minimize any valve diaphragm degradation due to excessive heat.

C. Event Description:

On June 22, 1992, the VCT and centrifugal charging pump (CB)(P) header, including CVCS-1-547, were pressurized to approximately 55 psig. This system condition was the result of maintenance unrelated to CVCS-1-547. Such an evolution is unusual but not outside the allowable CVCS operational limits. The VCT is normally pressurized to approximately 23 psig.

On June 22, 1992, during a routine radiation survey, diaphragm valve CVCS-1-547 was found to be leaking to the room drain at the rate of approximately 0.5 gpm. No boric acid crystals were present, which indicated that the valve had not been leaking for an extended period of time.

The valve bonnet retaining nuts were determined to be "finger-tight" and retorquing the nuts stopped the leakage. The as-left torque on the nuts was in accordance with the valve supplier's requirements.

Investigation determined that the bonnet temperature of CVCS-1-547 was approximately 304°F. No estimate of the time the valve had been at this temperature could be made. Information from the valve vendor (ITT) indicated that the qualified operating limits for the valve diaphragm are 100 psig at 300°F, 175 psig at 250°F and 235 psig at 200°F. Therefore, the as-found condition was in excess of the vendor-recommended limits. No other CVCS diaphragm valves had a measured body temperature over 200°F.

On June 26, 1992, an evaluation determined that the leakage from CVCS-1-547 could have resulted in the control room (NA) and exclusion area boundary 10 CFR 100 thyroid dose limits being exceeded during the recirculation phase of recovery from a design-basis LOCA. A one-hour, non-emergency report was made for Unit 1 in accordance with 10 CFR 50.72 (b)(1)(ii)(B) at 1549 PDT.

During the Unit 1 fifth refueling outage (1R5), which started on September 12, 1992, and ended on November 11, 1992, the valve bonnet and diaphragm of valve CVCS-1-547, as well as the heat trace controllers (FE)(TH), were replaced to return the valve to an acceptable configuration. All heat-traced diaphragm valves in the post-LOCA recirculation flow path were inspected and reconfigured as necessary.

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

None.

E. Dates and Approximate Times for Major Occurrences:

1. June 22, 1992: Event date. CVCS-1-547 was found to be leaking approximately 0.5 gpm.

2. June 26, 1992: Discovery date. Investigation identified that the leakage condition could have resulted in exceeding dose limits. A one-hour, non-emergency report was made to the NRC in accordance with 10 CFR 50.72(b)(1)(ii)(B).

3. July 1, 1992: A leak repair enclosure was installed on CVCS-1-547 to provide system pressure boundary integrity until valve repairs could be performed.

4. November 4, 1992: Unit 1 entered Mode 4 (Hot Shutdown) with CVCS-1-547 operational.

F. Other Systems or Secondary Functions Affected:

None.

G. Method of Discovery:

The leakage was discovered by radiation protection personnel during the performance of a routine radiation survey.

H. Operator Actions:

An operator retorqued the body-to-bonnet nuts on CVCS-1-547 and stopped the leak.

I. Safety System Responses:

None.

III. Cause of the Event

A. Immediate Cause:

CVCS-1-547 had a body-to-bonnet leak exceeding the maximum permissible leakage for unfiltered plant effluent.

B. Root Cause:

Although heat tracing is not installed on the bonnet of CVCS-1-547, the valve had insulation installed. The heat trace controller (thermostat) for this segment of system piping is not at CVCS-1-547. The physical arrangement of the piping at CVCS-1-547 resulted in heat accumulation at the valve, as evidenced by measured valve body temperature. Investigation determined that the heat trace controller for CVCS-1-547 was not turning off.

Although the vendor's qualification for the valve diaphragm temperature was only slightly exceeded, the root cause of the leakage was thermally induced degradation of the CVCS-1-547 diaphragm caused by the heat trace controller for the valve not turning off, resulting in valve diaphragm distortion and breaching of the system pressure boundary.

IV. Analysis of the Event

The leakage from CVCS-1-547 was estimated to be approximately 0.5 gpm. However, this leakage was occurring with the system pressure at approximately 55 psig. Under post-LOCA conditions, the system pressure at this valve would be approximately 200 psig. The equivalent leakage under post-LOCA conditions is postulated to be approximately 9.0 gpm.

A leak of 9.0 gpm in the auxiliary building, filtered only by HEPA filters, could potentially have resulted in control room operator dose exceeding the 10 CFR 50 Appendix A General Design Criterion 19 thyroid limit over the 30-day duration of the design-basis LOCA.

However, post-LOCA emergency response procedures provide for use of self-contained breathing apparatus (SCBAs) and potassium iodide prophylaxis, which would mitigate control room operator dose. Control room radiation conditions would be monitored by area radiation monitors (IL)(MON) located in the control room. Although the monitors are design Class II, they are powered from Class 1E power supplies (IL)(JX). The area radiation monitors would provide sufficient indication to allow control room operators to don SCBA equipment or take additional corrective measures.

A leak of 9.0 gpm from the auxiliary building, filtered only by the HEPA filters, could potentially have resulted in exceeding the 10 CFR 100 2-hour site boundary thyroid dose limit.

However, a design-basis LOCA dose analysis contains many conservative assumptions, particularly with regards to the source term (i.e., fuel damage). Therefore, an analysis was performed using "expected case" LOCA assumptions (no fuel damage). The analysis determined that a 9.0 gpm leak would result in 2-hour site boundary and low population zone doses significantly less than the 10 CFR 100 limit of 300 rem.

Therefore, this event did not adversely affect the health and safety of the public.

V. Corrective Actions

A. Immediate Corrective Actions:

1. Personnel tightened the body-to-bonnet nuts on the valve, which stopped the leakage.

2. A leak repair enclosure was installed on CVCS-1-547.

B. Corrective Actions to Prevent Recurrence:

1. PG&E will document the heat trace program implementation (i.e., that thermostats are appropriately located and components and piping are at a temperature between 70 and 170°F).

2. PG&E replaced the bonnet and diaphragm of CVCS-1-547 during 1R5.

3. PG&E has set the temperature on the piping immediately adjacent to CVCS-1-547 to between 70 and 170°F following replacement of the heat trace controllers during 1R5.

4. PG&E has established acceptable body/bonnet surface temperatures on all diaphragm valves that are in heat-traced systems, including the post-LOCA recirculation flow path.

VI. Additional Information

A. Failed Components:

Heat Trace Temperature Controller, Thermon Manufacturing Co., Type FP Thermon Econtrace, 120 vac.

B. Previous LERs on Similar Events:

LER 2-91-009-01, "10 CFR 100 Dose Limits Potentially Exceeded in the Event of a Design Basis Loss of Coolant Accident Recovery as a Result of Valve Leakage"

This previous LER was also caused by leakage from diaphragm valves in the post-LOCA flow path. The root cause was that one of the valves and certain vendor recommendations were not included in the preventive maintenance program. Because the scope of previous corrective actions did not include heat tracing on diaphragm valves in the post-LOCA flow path, the corrective actions could not have prevented the current LER.