Pacific Gas and Electric Company®

James R. Becker
Site Vice President

Diablo Canyon Power Plant
Mail Code 104/5/601
P. O. Box 56
Avila Beach, CA 93424

805.545.3462
Internal: 691.3462
Fax: 805.545.6445

January 22, ~~2009~~ 2010

PG&E Letter DCL-10-004

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

Docket No. 50-323, OL-DPR-82
Diablo Canyon Unit 2
Licensee Event Report 2-2009-002-01
Technical Specification 3.7.1 Violation Due to Cracked Valve Spring

Pacific Gas and Electric Company is submitting the enclosed revision to a licensee event report (LER) regarding a Technical Specification 3.7.1, "Main Steam Safety Valves," violation due to a cracked valve spring resulting in a presumed past inoperability. The initial LER was submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) via DCL-09-071 on October 22, 2009.

There are no new or revised regulatory commitments in this report.

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

Sincerely,

James R. Becker

ddm/2246/50264259
Enclosure
cc/enc: Elmo E. Collins, NRC Region IV
Michael S. Peck, NRC Senior Resident Inspector
Alan B. Wang, NRR Project Manager
INPO
Diablo Distribution

IE22
NRR

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Diablo Canyon Unit 2	05000323	1 OF 7

4. TITLE
Technical Specification 3.7.1 Violation Due to Cracked Valve Spring

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	26	2009	2009	002	01	01	19	2010	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Steven W. Hamilton – Senior Regulatory Services Engineer	(805) 545-3449

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	SB	RV	D243	Yes					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On August 26, 2009, at 12:45 PDT, with Unit 2 in Mode 1 (Power Operation) plant operators declared the main steam safety valve (MSSV) RV-224 inoperable in accordance with Technical Specification (TS) 3.7.1 Limiting Condition for Operation, and reduced power.

On August 26, 2009, at 16:06 PDT, Technical Maintenance personnel completed resetting the power range high flux reactor trip setpoints from 109 percent to 87 percent reactor power completing TS Action 3.7.1.A.1.

This event was the result of a cracked MSSV spring. Based upon the final assessment Pacific Gas and Electric Company presumes the valve was outside the TS allowable setpoint prior to discovery. Immediate corrective actions included gagging the MSSV to preclude inappropriate opening during power operation. The MSSV RV-224 spring was removed during the Unit 2 fifteenth refueling outage. Additional failure analysis supports that the cause of the failure was environmentally induced corrosion initiated cracking with subsequent spring fracture.

TEXT

I. Plant Conditions

Unit 2 was in Mode 1 (Power Operation) at approximately 100 percent reactor power with normal operating reactor coolant temperature and pressure.

II. Description of Problem

A. Background

The Diablo Canyon Power Plants (DCPP) Units 1 and 2 are Pressurized Water Reactors (PWR) with four Reactor Coolant Loops (RCL)[AB] to circulate reactor coolant to each of the four steam generators (SG)[SG]. Each SG is a vertical U-tube design provided by the Nuclear Steam Supply System (NSSS) vendor, Westinghouse. Each SG has five main steam safety valves (MSSVs) for a total of twenty MSSVs, each sized in accordance with the DCPP design and ASME Code requirements.

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the main turbine condenser and circulating water system, is not available.

Technical Specification (TS) 3.7.1, "Main Steam Safety Valves (MSSVs)," requires five MSSVs per SG to be operable in Modes 1, 2, and 3. If one or more MSSVs are inoperable, action to reduce the power range high neutron flux trip setpoint per Table 3.7.1-1 must be taken within 4-hours.

The MSSVs are located on each main steam header, outside the primary containment structure, upstream of the main steam isolation valves (MSIVs). The MSSVs have sufficient capacity to limit the secondary system pressure to less than 110 percent of the SG design pressure. The MSSV design includes staggered setpoints, according to TS Table 3.7.1-2, so that only the needed MSSVs will actuate. MS-2-RV-224 has the highest required "as found" lift setting at 1115 psig (plus or minus 3 percent), and RV-11 has the lowest at 1065 psig (plus 3 percent, minus 2 percent) for SG 2-3. Staggered setpoints reduce the potential for valve chattering due to steam pressure that is insufficient to fully open all valves during an overpressure event.

TEXT

The operability of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve SG overpressure, and reseat when pressure has been reduced. The operability of the MSSVs is verified by periodic surveillance testing in accordance with the Inservice Testing Program (IST) that verifies the "as-left" lift setting is within plus or minus 1 percent for two successive lifts in accordance with Code requirements.

Maintenance Procedure (MP) M-4.18A, "Verification of Main Steam Safety Valves Lift Point Using Fermanite's Trevitest Equipment," performs a semi-automated assisted valve lift methodology. The Trevitest System places an external pulling load on the valve stem in addition to the steam seat forces. When the total lift force overcomes the spring force, the valve lifts off its seat. Once the valve lifts off its seat, the added lift force is quickly released, closing the valve, and the lift setpoint is recorded.

MS-2-RV-224 was last replaced with a refurbished relief valve in March 1998. Testing is performed every other fuel cycle, and is normally performed as preoutage work. MS-2-RV-224 was last tested per TS Surveillance Requirement (SR) 3.7.1.1 in 2006 and the as-found and as-left setpoints were within specification with no adjustments required.

B. Event Description

On August 26, 2009, at 11:25 PDT, SG 2-3 surveillance testing performed in accordance MP M-4.18A identified that the RV-224 lift point was out of tolerance, with the setpoint "as found" 7 percent low.

On August 26, 2009, at 11:36 PDT, plant operators declared RV-224 inoperable and entered TS 3.7.1, Action A.1, due to the low as found lift.

On August 26, 2009, at 12:05 PDT, Engineering personnel performing Trevitesting informed the control operator (CO)/Shift Foreman (SFM) that RV-224 was adjusted within tolerance. (TS Sheet 2-TS-09-0706)

On August 26, 2009, at 12:45 PDT, Engineering performing Trevitesting informed the CO/SFM that RV-224 had a crack in it's spring. Plant operators declared RV-224 inoperable due to the observed degradation. (TS Sheet 2-TS-09-0711 for SG 2-3 safety valve RV-224)

On August 26, 2009, at 13:29 PDT, plant operators commenced ramping the Unit from approximately 100 percent power to 80 percent power to

TEXT

reset the Nuclear Instrument (NI) system power range high neutron flux trip setpoint in accordance with TS 3.7.1, Table 3.7.1-1.

On August 26, 2009, at 14:36 PDT, technical maintenance (TM) personnel commenced resetting the power range high flux reactor trip setpoint to 87 percent power

On August 26, 2009, at 14:37 PDT, plant operators stabilized reactor power ramp at approximately 80 percent.

On August 26, 2009, at 16:06 PDT, TM personnel completed resetting the power range high flux reactor trip setpoints from 109 percent power to 87 percent reactor power completing TS Action 3.7.1.A.1.

On August 26, 2009, at 16:27 PDT, maintenance personnel placed a gag on the MSSV to preclude inappropriate opening of the valve.

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

None.

D. Other Systems or Secondary Functions Affected

No additional safety systems were adversely affected by this event.

E. Method of Discovery

During Trevitesting on MS-2-RV-224, an area of degradation was noted on the valve spring while making adjustments to spring load in accordance with MP M-4.18A.

F. Operator Actions

Plant operators declared the MSSV inoperable, reduced reactor power and initiated actions to reset the NI system setpoints in accordance with TS Action 3.7.1.A.1.

G. Safety System Responses

None.

TEXT

III. Cause of the Problem

A. Immediate Cause

Preliminary ISI evaluation of the cracked spring concluded that the most likely failure mechanism is corrosion pitting of the spring. This is based upon high strength spring steel's low tolerance for surface flaws such as pitting, such that once a corrosion pit achieves a certain size, a spring will crack rather quickly through the thickness (cross section) of the spring.

It appears that corrosion between the spring and the bottom spring holder provided the environment for the formation of the pitting of the spring surface. Once the pit reached a critical size, the spring cracked through the cross section. Examination of the opening of the crack showed a fractured surface that is corroded, indicating that the spring has been cracked for some time.

B. Cause

A cause evaluation found the spring material to be in accordance with the original design specification. The evaluation of the fracture was hampered by the excessive amount of corrosion product on the fracture surface. This amount of corrosion was a strong indicator that the spring had been cracked for some time and the crack had propagated over a large percentage of the spring cross section before it failed due to overload. It was determined that the crack initiated at the edge of the spring that interfaces with the bottom spring washer; however, the exact initiation site could not be determined.

The cause evaluation concluded that the original spring coating was inadequately maintained, which contributed to the environmentally induced corrosion initiated cracking and subsequent fracture of the valve spring.

IV. Assessment of Safety Consequences

There were no safety consequences as a result of this event.

The Unit 2 reactor was maintained in Mode 1, with TS-required equipment operable, while reactor power was decreased to allow reset of the NI system setpoints in accordance with TS Action 3.7.1.A.1.

TEXT

PG&E presumes RV-224 would have lifted low, outside the TS 3.7.1 allowable lift setting limits prior to discovery, for a period longer than allowed by the TS Action. Although the valve was capable of performing its safety function of steam pressure relief, it was inoperable, and could have resulted in an out-of-sequence MSSV valve lift. The remaining 19 MSSVs were available and capable of relieving steam pressure as necessary to protect the unit from overpressurization. An out-of-sequence MSSV lift could result in additional cooldown following a reactor trip; however, this condition is bound by a failed open MSSV, a previously analyzed condition in Final Safety Analysis Report (FSAR) Update, Section 15.2.14, "Accidental Depressurization of the Main Steam System." Therefore, the MSSV pressure relief system was capable of performing its safety function and this condition was not a safety system functional failure.

This event is not considered risk significant and it did not adversely affect the health and safety of the public.

V. Corrective Actions

A. Immediate Corrective Actions

Plant operators entered TS 3.7.1, maintained Unit 2 in Mode 1, and decreased reactor power to allow reset of the NI system setpoints in accordance with TS Action 3.7.1.A.1

Visual inspections were performed on all Unit 1 and Unit 2 MSSV springs. No linear indications were observed other than RV-224.

B. Corrective Actions to Prevent Recurrence (CAPR)

The RV-224 valve spring was removed and sent offsite for detailed material analysis of the failure.

The RV-224 valve body will be shipped to an independent valve facility for replacement of the spring, valve refurbishment, and steam testing. Upon satisfactory completion of the offsite refurbishment and testing, the valve will be returned to warehouse spares for future use.

The MSSV Springs will be replaced on a six refueling outage cycle (6R) frequency. The new springs will have a corrosion protection coating.

TEXT

VI. Additional Information

A. Failed Components

MSSV manufactured by Dresser Industries, a 6 inch ASME Code valve. Model: 3707RAX621.

B. Previous Similar Events

None.

C. Industry Reports

None.