Pacific Gas and Electric Company®

James M. Welsch
Vice President, Nuclear Generation

Diablo Canyon Power Plant
P.O. Box 56
Avila Beach, CA 93424

805.545.3242
E-Mail: JMW1@pge.com

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PG&E Letter DCL-16-021

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

10 CFR 50.73

Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
Licensee Event Report 2015-001-01, Both Trains of Residual Heat Removal Inoperable Due to Circumferential Crack on a Socket Weld

References:
1. PG&E Letter DCL-15-033, "Licensee Event Report 2015-001-00, Both Trains of Residual Heat Removal Inoperable Due to Circumferential Crack on a Socket Weld," dated March 2, 2015 (ADAMS Accession No. ML15061A548)
2. PG&E Letter DCL-15-062, "Expected Submittal Date for Licensee Event Report 2015-001-01," dated May 7, 2015 (ADAMS Accession No. ML15127A634)

Dear Commissioners and Staff:

In Reference 1, Pacific Gas and Electric Company (PG&E) submitted a Licensee Event Report (LER) to the U.S. Nuclear Regulatory Commission (NRC) related to an event or condition that could have prevented fulfillment of a safety function when both trains of the residual heat removal system were declared inoperable due to a circumferential crack on a socket weld. In this LER, PG&E indicated that it would provide a supplemental LER describing the event cause and corrective actions.

In Reference 2, PG&E informed the NRC that the revised completion date for the associated supplemental LER 2015-001-01 would be January 28, 2016 based on a pending root cause evaluation. However, as discussed with the NRC Staff on January 28, 2016, additional time was required to finalize the root cause evaluation in support of the supplemental LER.

PG&E hereby submits the enclosed supplemental LER, which includes the results of the root cause and corrective actions. All corrective actions have been implemented in accordance with the PG&E Corrective Action Program.

PG&E makes no new or revised regulatory commitments (as defined by NEI 99-04) in this letter.

Sincerely,

James M. Welsch
Vice President, Nuclear Generation

bnsm/50680117
Enclosure
cc/enc: Marc L. Dapas, NRC Region IV Administrator
Siva P. Lingam, NRR Project Manager
Binesh K. Tharakan, Acting NRC Senior Resident Inspector
INPO
Diablo Distribution

NRC FORM 366 (11-2015) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE

Both Trains of Residual Heat Removal Inoperable Due to Circumferential Crack on a Socket Weld

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 05000
12	31	2014	2015 -	001 -	01	02	11	2016	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
3	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☑ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
000	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☑ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☑ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Brandy Lopez, Regulatory Services Engineer	(805) 545-4540

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	BP	N/A	N/A	Y					

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 December 31, 2014, while performing a walkdown as part of a surveillance test procedure, plant personnel identified through-wall seepage in a Diablo Canyon Power Plant Unit 1 socket weld inside containment that provides a flow path to a relief valve protecting a common portion of both trains of the residual heat removal system. Subsequent cleanup of the boric acid accumulation revealed active seepage of 30 drops per minute. A visual inspection identified that the source of the seepage was a circumferential crack on the socket weld.

Pacific Gas and Electric Company determined that the root cause of the cracked socket weld was containment fan cooler unit (CFCU) vibration inducing a resonant condition in the residual heat removal piping that generated stresses above the material endurance limit of the socket weld. Corrective actions included replacing two socket welds, modifying pipe supports, and correcting the condition causing the CFCU vibrations.

This condition did not have an adverse effect on the health and safety of the public.

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

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NARRATIVE

I. Reportable Event Classification

This event is reportable pursuant to the following criteria:

• 10 CFR 50.73(a)(2)(v)(B&D), "Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to: Remove Residual Heat and Mitigate the consequences of an accident"

• 10 CFR 50. 73(a)(2)(vii), "Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to: Remove Residual Heat, and Mitigate the consequences of an accident"

II. Plant Conditions

At the time of the event, Diablo Canyon Power Plant (DCPP) Unit 1 was in Mode 3 (Hot Standby) at normal operating reactor coolant temperature and pressure conditions.

III. Problem Description

A. Background

The function of the emergency core cooling system (ECCS) is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a) loss-of-coolant accident, non-isolable coolant leakage greater than the capability of the normal charging system [CB]
b) rod ejection accident
c) loss-of-secondary-coolant accident, including uncontrolled steam release or loss of feedwater and
d) steam generator tube rupture.

The addition of negative reactivity is designed primarily for the loss-of-secondary-coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant reactor power. The ECCS consists of three separate subsystems: centrifugal charging (high head) [BQ], safety injection (intermediate head), and residual heat removal (RHR) (low head) [BP]. Each subsystem consists of 2 redundant 100 percent capacity trains.

The design function of Relief Valve [RV] RHR-1-RV-8708 is to protect the RHR discharge piping from exceeding its design pressure rating. The inlet pipe to the valve is connected to a 12-inch RHR header line, which provides a flow path for injection to reactor coolant system (RCS) [AB] Hot Legs 1 and 2. This line is occasionally used to fill the reactor cavity during refueling outages. The normal flow path for shutdown cooling (Modes 4 and 5) does not use this line.

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B. Event Description

On December 31, 2014, while performing a walkdown as part of a surveillance test procedure, plant personnel identified an accumulation of boric acid on the inlet pipe to Relief Valve RHR-1-RV-8708. The problem was entered into the corrective action program. Subsequent cleanup of the boric acid accumulation revealed active seepage of 30 drops per minute (dpm). A visual inspection identified a circumferential crack on the socket weld.

The active boric acid leak was located on the socket welded connection of the relief valve inlet pipe to the common header from the RHR pump [P] discharge to RCS Hot Legs 1 and 2. The active boric acid seepage could not be isolated. Both trains of the RHR system were declared inoperable and Technical Specification (TS) 3.0.3 was entered on December 31, 2014, at 1105 PST. DCPP made an 8-hour notification to the NRC regarding an event or condition that could have prevented fulfillment of a safety function (NRC Event Notification Number 50711).

C. Status of Inoperable Structure, Systems, or Components That Contributed to the Event

None.

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

On December 31, 2014, while performing a walkdown as part of a surveillance test procedure, plant personnel identified an accumulation of boric acid on the inlet pipe to Relief Valve RHR-1-RV-8708. Subsequent cleanup of the boric acid accumulation revealed active seepage of 30 dpm. A visual inspection identified that the source of the leak was a circumferential crack on the socket weld at the connection of the relief valve inlet pipe to the RHR line.

F. Operator Actions

Operators declared both RHR trains inoperable. TS 3.0.3 was entered on December 31, 2014, at 1105 PST and was exited at 2256 PST, when the plant entered Mode 4. Additionally, in accordance with TS 3.6.3.C, the associated containment penetration flow path was isolated.

G. Safety System Responses

None.

IV. Cause of the Problem

Pacific Gas and Electric Company determined that the root cause of the cracked socket weld was containment fan cooler unit (CFCU) vibration inducing a resonant condition in the RHR piping that generated stresses above the material endurance limit of the socket weld.

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V. Assessment of Safety Consequences

DCPP assessed the Unit 1 risk significance of seepage in the vent line of the RHR supply line to the Hot Leg 1 and 2 supply lines. This assessment concluded that the incremental conditional core damage probability was less than 1.0E-06 per year. Therefore, this event is not considered risk significant and did not adversely affect the health and safety of the public.

VI. Corrective Actions

A. Immediate Actions

1. Performed repair on cracked socket weld.
2. Installed pipe support on RV-8708 piping.

B. Other Corrective Actions

1. Replaced the previously repaired socket weld and one additional socket weld on the relief valve discharge pipe, during the subsequent Unit 1 nineteenth refueling outage (1R19).
2. Relocated the previously installed pipe supports.
3. Corrected the condition that caused the CFCU vibrations, during the subsequent 1R19.

VII. Additional Information

A. Failed Components

DCPP discovered a circumferential crack on the socket weld on the inlet pipe connection of Relief Valve RHR-1-RV-8708.

B. Previous Similar Events

"Unit 1 Licensee Event Report 2013-005-00, Both Trains of Residual Heat Removal Inoperable Due to Circumferential Crack on a Socket Weld," dated August 22, 2013, is similar to this event.

A. Industry Reports

None.