



**Pacific Gas and  
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PG&E Letter DCL-00-160

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


Docket No. 50-275, OL-DPR-80  
Diablo Canyon Unit 1  
Licensee Event Report 1- 2000-012  
Unit 1 Automatic Reactor Trip Due to Test Equipment Failure

Dear Commissioners and Staff:

PG&E is submitting the enclosed licensee event report regarding an automatic reactor trip due to test equipment failure.

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

Sincerely,



For DHO

David H. Oatley

cc: Ellis W. Merschoff  
David L. Proulx  
Girija S. Shukla  
Diablo Distribution  
INPO

Enclosure

SHC/2246/N0002120

IE#2

# LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) <b>Diablo Canyon Unit 1</b>										DOCKET NUMBER (2) <b>0 5 0 0 0 2 7 5</b>						PAGE (3) <b>1 OF 5</b>				
TITLE (4) <b>Unit 1 Automatic Reactor Trip Due to Test Equipment Failure</b>																				
EVENT DATE (5)			LER NUMBER (6)						REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)								
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER			REVISION NUMBER			MO	DAY	YEAR	FACILITY NAME				DOCKET NUMBER			
<b>11</b>	<b>20</b>	<b>2000</b>	<b>2000</b>	<b>-</b>	<b>0</b>	<b>1</b>	<b>2</b>	<b>-</b>	<b>0</b>	<b>0</b>	<b>12</b>	<b>20</b>	<b>2000</b>							
OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (11)																		
<b>1</b>		<div style="display: flex; justify-content: space-between;"> <span><b>X</b> 10 CFR</span> <span><b>50.73(a)(2)(iv)</b></span> </div>																		
POWER LEVEL (10)		<div style="display: flex; justify-content: space-between;"> <span>OTHER</span> <span>(SPECIFY IN ABSTRACT BELOW AND IN TEXT, NRC FORM 366A)</span> </div>																		
<b>0</b>	<b>4</b>	<b>6</b>																		
LICENSEE CONTACT FOR THIS LER (12)																				
<b>Roger Russell - Senior Regulatory Services Engineer</b>															TELEPHONE NUMBER					
															AREA CODE		NUMBER			
															<b>805</b>		<b>545-4327</b>			
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)																				
CAUSE	SYSTEM	COMPONENT		MANUFACTURER			REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT		MANUFACTURER			REPORTABLE TO EPIX				
<b>X</b>	<b>I</b>	<b>G</b>	<b>H</b>	<b>S</b>	<b>U</b>			<b>N</b>	<b>K</b>	<b>N</b>	<b>No</b>									
SUPPLEMENTAL REPORT EXPECTED (14)												EXPECTED			MON	DAY	YR			
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)												[X] NO			SUBMISSION DATE (15)					

**ABSTRACT (Limit to 1400 spaces. i.e., approximately 15 single-spaced typewritten lines.) (16)**

On November 20, 2000, at 0352 PST, with Unit 1 in Mode 1 (Power Operation) at approximately 46 percent power, plant operators experienced an automatic reactor trip. Incore flux map testing was in progress when an intermittent electrical short circuit in test equipment attached to Nuclear Instrumentation (NI)-44, concurrent with a preexisting tripped condition associated with NI-41, provided a two out of four coincidence resulting in a reactor trip. This event is an engineered safety feature and reactor protection system actuation.

On November 20, 2000, at 0445 PST, a 4-hour nonemergency report was made in accordance with 10 CFR 50.72(b)(2)(ii).

Corrective actions to preclude recurrence include revising procedures, and providing a case study regarding this event and expected test prerequisites to appropriate plant personnel.

# LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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									YEAR	SEQUENTIAL NUMBER				REVISION NUMBER			
Diablo Canyon Unit 1	0	5	0	0	0	2	7	5	2000	-	0	1	2	-	0	0	2 OF 5

TEXT

## I. Plant Conditions

Unit 1 was in Mode 1 (Power Operation) at approximately 46 percent power.

## II. Description of Problem

### A. Background

Technical Specification (TS) Surveillance Requirement (SR) 3.3.1.6, Table 3.3.1-1, Function 6, requires a power range incore/excore calibration above 75 percent rated thermal power (RTP) at least once per 92 effective full power days. The intent of this specification is to relieve the plant from having to perform the calibration quarterly below 75 percent. However, the TS do not prohibit calibration below this power if desired.

During beginning of cycle power ascension testing, Surveillance Test Procedure (STP) R-40, "Reload Power Ascension Testing," requires heat balance testing be performed at power levels of approximately 30, 50, 73, 90 and 100 percent RTP as necessary. Singlepoint calibration of Nuclear Instrumentation (NI) is typically performed between 48 and 50 percent RTP. After reaching the expected operating power level (normally 100 percent RTP), STP R-40 then calls for a multiplepoint calibration to be performed within 15 percent of the expected operating power.

The acceptance criteria of STP R-13B, "Nuclear Power Range Incore/Excore Single-Point Calibration Data," requires the incore Axial Offset (AO) determined by the full-core flux map to be within the range of  $\pm 10$  percent AO. Otherwise, another full-core flux map is required to be performed with control rods repositioned to give the required axial offset. The incore/excore gain adjustment factor must be less than 1.16 in order for the results of this procedure to be used.

### B. Event Description

On November 17 and 18, 2000, following the Unit 1 tenth refueling outage (1R10), various plant power ascension tests were being performed. At 26 percent RTP, while performing PEP R-3A, "Use of Flux Mapping Equipment" and STP R-3D, "Routine Monthly Flux Map," it was determined that STP R-13B could not be performed because the AO was outside the prerequisite range of  $\pm 10$  percent.

On November 19, 2000, at 1355 PST, while performing STP R-2B1, "Plant Process Computer Operator Heat Balance" NI-41 was declared inoperable because it could not be adjusted to within  $\pm 2$  percent calculated reactor

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TEXT

power. The bistables were placed in the tripped condition as required by TS 3.3.1, "Reactor Trip System Instrumentation," required action D.1.1.

On November 20, 2000, at 0352 PST, PEP R-3A was in progress with test equipment installed on all NI channels. NI-41 was in the tripped condition. While testing equipment, in preparation for data collection, a trip signal was generated due to an intermittent electrical short in a toggle selector switch of the digital volt meter (DVM) attached to NI-44. This resulted in a two out of four coincidence which led to a Reactor Protection System (RPS) and an Engineered Safety Feature (ESF) actuations. Plant operators confirmed the reactor trip, verified proper ESF actuations, and initiated actions to stabilize the unit in Mode 3.

On November 20, 2000, at 0445 PST, a 4-hour nonemergency report was made in accordance with 10 CFR 50.72(b)(2)(ii).

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

NI-41 had been declared inoperable and the associated bistables had been placed in the tripped condition.

D. Other Systems or Secondary Functions Affected

None.

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

None.

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TEXT

## G. Safety System Responses

1. The reactor trip breakers [JC][BKR] opened.
2. The main turbine [TA][TRB] tripped.
3. The control rod drive mechanism [AA][DRIV] allowed the control rods to drop into the core.
4. The three auxiliary feedwater pumps [BA][P] started.

## III. Cause of the Problem

### A. Immediate Cause

The immediate cause of this event was an intermittent electrical short in a toggle selector switch on the DVM connected to NI-44, which caused a reactor trip to occur when the test engineer manipulated the switch.

### B. Root Cause

The root cause of this event was the decision made to proceed with testing on the redundant NIs prior to restoring the NI-41 channel to service.

## IV. Assessment of Safety Consequences

There were no safety consequences involved in this event because all safety equipment functioned as designed, and no design basis limits were exceeded.

An inadvertent reactor trip from 100 percent power is a previously analyzed Final Safety Analysis Report Update (FSAR), Chapter 15, Condition II event and bounds this reactor trip from 46 percent power. All ESF equipment functioned as expected in the FSAR analysis to assure that no design basis limits were exceeded. Therefore, the health and safety of the public were not adversely affected by this event.

The condition is not a safety system functional failure.

The condition was evaluated using the NRC's Significance Determination Process in accordance with NRC Inspection Manual Chapter 0609; but since reactor trip is a performance indicator, this event does not require screening.

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TEXT

## V. Corrective Actions

### A. Immediate Corrective Actions

1. The procedures that are used to perform NI incore/excore cross calibration testing were revised to allow eight DVMs to be connected to NI at one time. This will limit the need to use toggle selector switches when performing NI testing. The practice had been to use a four DVM configuration to test NIs which created the necessity to toggle back and forth from the upper range to the lower range of each NI channel when collecting data.
2. A memo was sent to plant personnel warning of the possibility of an electrical short when using DVMs.

### B. Corrective Actions to Prevent Recurrence

1. A case study will be provided to the appropriate Engineering, Maintenance, and Operations personnel regarding this event and expected test prerequisites.
2. Procedures STP R-2B1, STP R-2B2 and STP R-13B will be revised to clarify adjustments and test prerequisites.

## VI. Additional Information

### A. Failed Components

An electrical short in a toggle selector switch that was part of a DVM connected to NI-44.

### B. Previous Occurrences

None.