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ACCESSION NBR:8808020167 DOC.DATE: 88/07/25 NOTARIZED: NO DOCKET #
FACIL:50-275 Diablo Canyon Nuclear Power Plant, Unit 1, Pacific Ga 05000275
AUTH.NAME AUTHOR AFFILIATION
SISK,D.P. Pacific Gas & Electric Co.
SHIFFER,J.D. Pacific Gas & Electric Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 88-013-00:on 880625,main steam line isolation due to inadequate communications between I&C technicians.
W/8 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 6
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD5 LA	1	1	PD5 PD	1	1
	ROOD,H	1	1			
INTERNAL:	ACRS MICHELSON	1	1	ACRS MOELLER	2	2
	AEOD/DOA	1	1	AEOD/DSP/NAS	1	1
	AEOD/DSP/ROAB	2	2	AEOD/DSP/TPAB	1	1
	ARM/DCTS/DAB	1	1	DEDRO	1	1
	NRR/DEST/ADS 7E	1	0	NRR/DEST/CEB 8H	1	1
	NRR/DEST/ESB 8D	1	1	NRR/DEST/ICSB 7	1	1
	NRR/DEST/MEB 9H	1	1	NRR/DEST/MTB 9H	1	1
	NRR/DEST/PSB 8D	1	1	NRR/DEST/RSB 8E	1	1
	NRR/DEST/SGB 8D	1	1	NRR/DLPQ/HFB 10	1	1
	NRR/DLPQ/QAB 10	1	1	NRR/DOEA/EAB 11	1	1
	NRR/DREP/RAB 10	1	1	NRR/DREP/RPB 10	2	2
	NRR/DRIS/SIB 9A	1	1	NUDOCS-ABSTRACT	1	1
	REG FILE 02	1	1	RES TELFORD,J	1	1
	RES/DSIR DEPY	1	1	RES/DSIR/EIB	1	1
	RGN5 FILE 01	1	1			
EXTERNAL:	EG&G WILLIAMS,S	4	4	FORD BLDG HOY,A	1	1
	H ST LOBBY WARD	1	1	LPDR	2	2
	NRC PDR	1	1	NSIC HARRIS,J	1	1
	NSIC MAYS,G	1	1			

TOTAL NUMBER OF COPIES REQUIRED: LTTR 46 ENCL 45

RIDS/ADDS RIDS/ADDS

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 5

TITLE (4) MAIN STEAM LINE ISOLATION DUE TO INADEQUATE COMMUNICATIONS BETWEEN I&C TECHNICIANS AND THE CONTROL ROOM OPERATORS

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 6	2 5	8 8	8 8	0 1 3	0 0	0 7	2 5	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 0 0 0

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

[X] 10 CFR 50.73(a)(2)(iv)

[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)

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)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)

[X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (16)

On June 25, 1988, at 2101 PDT, with the unit in Mode 4 (Hot Shutdown), a main steam line isolation actuation occurred when Instrument and Controls (I&C) personnel tripped the high steam flow bistables while performing a surveillance test procedure. The main steam isolation valves were reopened at 2123 PDT. The 4-hour nonemergency report required by 10 CFR 50.72(b)(2)(ii) was completed at 2231 PDT.

All four reactor coolant loop Tavg channels were less than 543 degrees Fahrenheit; therefore, tripping of two high steam flow bistables completed the logic required for a main steam line isolation signal. All 4 main steam isolation valves and inside containment blowdown isolation valves closed as designed.

This event was determined to have been caused by inadequate communications between the I&C technicians and the control room operators. To prevent recurrence of this event, a new procedure to provide guidelines for specific instructions, reviews, and approvals for troubleshooting activities will be developed. Operations and I&C personnel will receive additional training on this event. Until the new procedure is issued, I&C will issue a policy reminding I&C personnel that the new procedure will contain a requirement that the tripping of any protective or safeguards bistable must be specifically approved in writing by the unit shift foreman.

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I. Initial Conditions

Unit 1 was in Mode 4 (Hot Shutdown) with reactor coolant system (RCS) pressure at approximately 900 psig and RCS temperature at 340 degrees Fahrenheit (F). The RCS heatup was on hold at this time pending resolution of the emergency core cooling system (ECCS) check valve test problems.

II. Description of Event

A. Event:

Control room operators had identified a potential problem with one of the steam line pressure channels (SB)(PIC) indicated by two steam line differential pressure bistables (515A&B) (SB)(PDS) being in the tripped condition for no apparent reason. I&C technicians had been notified of the problem and were investigating possible causes. Prior to troubleshooting the steam line pressure problem, a tailboard was held by the Shift Foreman and Control Room Operator with the I&C personnel performing the troubleshooting. The Shift Foreman questioned the I&C technicians as to whether any bistables other than the identified steam line differential pressure channels would need to be placed in the tripped position. The I&C technicians responded that only the 515 A&B channels would be tripped, and the Shift Foreman then authorized the I&C technicians to proceed with their troubleshooting efforts.

The I&C personnel removed the steam pressure channels from service, and were in the process of installing test equipment in accordance with STP I-12B6 "Steam Generator Feedflow, Steamflow and Pressure Channels". During the test equipment installation it was noticed that the pressure compensating signals to several steam flow channels would be affected (Flow channels 513 and 533). The technicians recognized a need to trip the high steam flow bistables associated with these steam flow channels. As required by STP I-12B1 "Removal From Service Steam Generator Feedflow, Steamflow, and Pressure Channels", permission from the Shift Foreman or an NRC licensed operator was necessary to trip these bistables with the unit in the current mode (RCS Tavg <543 F, SSPS active). They attempted to call the Shift Foreman, who was unavailable to speak on the phone at that time because of other activities. They then called the unit Control Room Operator with their request to trip additional bistables, however, the Senior Control Room Operator took the call.

The Senior Control Room Operator, who was not involved in the tailboard briefing with the I&C technicians, noted that steam line safety injection signal was blocked, and due to a miscommunication with the Control Room Operator authorized the I&C personnel to proceed.

Since all 4 Reactor Coolant Loop Tavg channels were less than 543 degrees F and, with the bistables tripped, two steam lines were in the "high steam flow condition", proper logic existed for the steam line isolation, which occurred immediately. All 4 Main Steam Isolation valves (MSIV)(SB)(ISV) and all 4 steam generator inside containment blowdown isolation valves (WI)(ISV) closed as designed.

B. Inoperable structures, components, or systems that contributed to the event:

Pressure Comparator 1PC 515 had failed, which required the troubleshooting described.

C. Dates and approximate times for major occurrences:

1. On June 25, 1988, at 2015 PDT: Shift Foreman authorizes I&C technicians to start work on steam line pressure channels.

2. On June 25, 1988, at 2101 PDT: Event date - The high steam flow bistables were tripped, steam line isolation occurred immediately.

3. On June 25, 1988, at 2101 PDT: Discovery Date - The problem was immediately known due to the status of control room alarms and indicators.

4. On June 25, 1988, at 2123 PDT: Recovery Date: The MSIVs and Steam Generator Blowdown isolation valves were reopened.

5. On June 25, 1988, at 2231 PDT: The event was reported to the NRC Operations Center in accordance with 10 CFR 50.72.

D. Other systems or secondary functions affected:

None.

E. Method of Discovery:

The problem was immediately known to operations personnel in the control room by alarms and valve status indicators.

F. Operator actions:

The Control Room Operator verified proper closure of all valves required for steam line isolation. Following the reinstatement of the high steam flow bistables, all main steam isolation valves and steam generator blowdown valves were reopened.

G. Safety system responses:

All four main steam isolation valves closed (FCVs-41, 42, 43, and 44).

All four steam generator inside containment blowdown isolation valves closed (FCVs 760, 761, 762, 763).

III. Cause of Event

A. Immediate Cause:

Tripping the high steam flow bistables made up the necessary logic (high steam flow with Low-Low Tavg in 2 of 4 RCS loops) to initiate the steam line isolation signal as designed.

B. Root Cause:

The root cause of this event is personnel error (cognitive), inadequate communications between the I&C technicians and the Control Room Operators. The Shift Foreman and Control Operator, who were involved in the previous tailboard with the technicians were not specifically made aware of the request to trip additional bistables. Communication used by the personnel involved was inadequate for this type of operation.

IV. Analysis of Event

The function of the steam line isolation system is to mitigate the effects of a steam line break downstream of the main steam isolation valves. Since the Auxiliary Feedwater System and the steam generator power operated relief valves were always available for Reactor Coolant System heat removal, this event had no impact on the safety and health of the public.

V. Corrective Actions:

A. Immediate Corrective Actions:

1. The control operator verified proper steam line isolation by reviewing the status lights.

2. The high steam flow bistables were reinstated and the MSIVs and steam generator blowdown valves were reopened.

B. Corrective Actions to Prevent Recurrence

1. Maintenance will develop a new procedure to provide guidelines for specific instructions, reviews, and approvals for troubleshooting activities.

2. Until the new troubleshooting procedure is issued, the I&C Department will issue a policy which requires that tripping of any protection or safeguards bistable be specifically approved in writing by the Unit Shift Foreman. Telephone requests will not be acceptable. This new policy will be reviewed with I&C personnel as soon as possible during routine tailboard meetings.

3. The Operations Training Department will review the steam line isolation logic and this event in the next available Licensed Operator Requalification Session.

4. An Operations Incident Summary will be issued to familiarize all Operations personnel with this event and its corrective actions.

VI. Additional Information

A. Failed components:

None

B. Previous LERs on similar events:

None

C. Additional Information:

None

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San Francisco, CA 94106
415/973-4684
TWX 910-372-6587

James D. Shiffer
Vice President
Nuclear Power Generation

July 25, 1988

PG&E Letter No. DCL-88-188

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 1-88-013-00
Main Steam Line Isolation Due To Inadequate Communications
Between I&C Technicians and the Control Room Operators

Gentlemen:

Pursuant to 10 CFR 50.73(a)(2)(iv), PG&E is submitting the enclosed Licensee Event Report (LER) regarding an inadvertent main steam line isolation.

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,

J. D. Shiffer

cc: J. B. Martin
M. M. Mendonca
P. P. Narbut
B. Norton
H. Rood
B. H. Vogler
CPUC
Diablo Distribution
INPO

Enclosure

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