REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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FACIL: 50-323 Diablo Canyon Nuclear Power Plant, Unit 2, Pacific Ga 05000323

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SUBJECT: LER 87-005-01: on 870410, RHR flow interrupted when both trains became inoperable due to airbound RHR pumps. Caused by inadequate communications. Procedures OP A-2: II re draining of RCS & OP AP-16 re RHR flow loss revised. W/871029 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 9
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
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	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	1	0	NRR/DEST/CEB	1	1
	NRR/DEST/ELB	1	1	NRR/DEST/ICSB	1	1
	NRR/DEST/MEB	1	1	NRR/DEST/MTB	1	1
	NRR/DEST/PSB	1	1	NRR/DEST/RSB	1	1
	NRR/DEST/SGB	1	1	NRR/DLPQ/HFB	1	1
	NRR/DLPQ/QAB	1	1	NRR/DOEA/EAB	1	1
	NRR/DREP/RAB	1	1	NRR/DREP/RPB	2	2
	NRR/DRIS/SIB	1	1	NRR/PMAS/ILRB	1	1
	REG FILE 02	1	1	RES DEPY GI	1	1
	RES TELFORD, J	1	1	RES/DE/EIB	1	1
	RGN5 FILE 01	1	1			
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
	LPDR	2	2	NRC PDR	1	1
	NSIC HARRIS, J	1	1	NSIC MAYS, G	1	1

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LICENSEE EVENT REPORT (LER)

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TITLE (4) INTERRUPTION OF RHR FLOW DURING RCS MIDLOOP OPERATION

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 4	1 0	8 7	8 7	0 0 5	0 1	1 0	2 9	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

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)(v)

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

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
DAVID P. SISK, REGULATORY COMPLIANCE ENGINEER	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
X	A B	S H V	G 2 5 5	NO					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

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

ABSTRACT (16)

On April 10, 1987, at 2123 PDT, with the unit in Mode 5 (Cold Shutdown) during a refueling outage, residual heat removal (RHR) flow was interrupted when both RHR trains became inoperable due to airbound RHR pumps. The 4-hour nonemergency event report required by 10 CFR 50.72 was made at 2230 PDT, April 10, 1987.

The reactor coolant system (RCS) had been drained to midloop level to facilitate removal of steam generator (SG) primary manways for nozzle dam installation. In addition, preparations were in progress for local leak rate testing of a seal water return line (including draining of the penetration). Due to a leaking valve used as a clearance point in the piping to the penetration being drained, RCS inventory was lost to the reactor coolant drain tank. This loss of inventory caused a decrease in RCS water level, vortexing in the pumps' suction line, and air entrainment in the RHR pumps.

At 2251 PDT, after verification that the SG manways were still installed and after venting of the RHR pumps, the RCS was flooded from the refueling water storage tank and an RHR pump started. RHR flow was interrupted for approximately 1 hour and 28 minutes. This resulted in some localized boiling but no damage to the core or significant radiological release. The unit was stable at 0130 PDT, April 11, 1987, and was returned to normal Mode 5 midloop operation.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

I. Initial Conditions

Unit 2 was in Mode 5 (Cold Shutdown) with the reactor coolant system (RCS) vented to the atmosphere and a reactor coolant average temperature of approximately 90 degrees Fahrenheit.

II. Description of Event

A. Event:

On April 10, 1987, at 2123 PDT, with the unit in Mode 5 (Cold Shutdown) during a refueling outage, residual heat removal (RHR) flow was interrupted. Operators had observed amperage fluctuations on RHR pump 2-2 (BP)(P) and shut it down. RHR pump 2-1 (BP)(P) was started. Within 60 seconds, indications of vortexing were observed on this pump also, and it was shut down. This resulted in both RHR trains (BP) being inoperable. The 4-hour nonemergency event report required by 10 CFR 50.72 was made at 2230 PDT, April 10, 1987.

The reactor coolant system (RCS)(AB) had been drained to midloop level to facilitate removal of steam generator (SG) primary manways for nozzle dam installation. In addition, preparations were in progress for a leak rate test on a seal water return line (including draining of the penetration). Due to a leaking valve used as an isolation point in the piping to the penetration being drained, RCS inventory was lost to the reactor coolant drain tank (RCDT). This loss of inventory caused a decrease in RCS water level, vortexing in the pumps' suction line, and air entrainment in the RHR pumps.

At 2251 PDT, after verification that the SG manways were still installed and after venting of the RHR pumps, the RCS was flooded from the refueling water storage tank (RWST) and an RHR pump started. RHR flow had been interrupted for approximately 1 hour and 28 minutes. This resulted in some localized boiling but no damage to the core or significant radiological release. The unit was stable at 0130 PDT, April 11, 1987, and was returned to normal Mode 5 midloop operation.

During this event, small amounts of reactor coolant were lost to containment via steam through the reactor head vent and via leakage of the SG manways when the RCS was flooded from the RWST. Releases of radioactive material to the atmosphere via open containment hatches and valves were well within that allowed by the Code of Federal Regulations.

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

1. Valve 8976 (common safety injection pump suction from the RWST) was cleared electrically (could be opened manually).

2. All core exit thermocouples had been determinated in preparation for reactor vessel head removal.

3. Postaccident monitoring panels 1 and 2 were out of service for modifications.

4. All four accumulators had been cleared and drained.

5. The charging system flow meter was inoperable.

6. The equipment hatch and personnel hatches and three manual vent valves outside of containment were open.

C. Dates and approximate times for major occurrences:

1. April 10, 1987, at 2043 PDT: An engineer opens valve to drain reactor coolant pump seal return penetration in preparation for local leak rate test. Shift foreman was not notified that draining had started.

2. April 10, 1987, at 2051 PDT: Control room operator noticed volume control tank (VCT) level trending down, therefore increased RCS letdown to VCT to increase level.

3. April 10, 1987, at 2123 PDT: Event date - RHR pump 2-2 was stopped due to indication of vortexing and pump 2-1 was started. RHR pump 2-1 was stopped 60 seconds later when indication of vortexing was observed.

4. April 10, 1987, at 2138 PDT: Operators closed valve to stop inventory loss from the VCT. Level decrease in VCT stopped.

5. April 10, 1987, at 2230 PDT: Health Physics on 140-foot level noticed increase in airborne activity.

6. April 10, 1987, at 2230 PDT: Four-hour nonemergency event report made in accordance with 10 CFR 50.72.

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7. April 10, 1987, at 2251 PDT: RCS level increased from RWST. RHR pump started.

8. April 11, 1987, at 0130 PDT: Unit stable in Mode 5, returning to normal midloop operation.

9. April 11, 1987, at 0320 PDT: Reactor vessel water level above midloop 108 feet 4 inches. SG manway leakage stopped.

D. Other systems or secondary functions affected:

None

E. Method of discovery:

Control room operators observed pump amperage fluctuations.

F. Operator actions:

After securing the second RHR pump, an auxiliary operator was dispatched to vent the RHR pumps in preparation for restarting. Another auxiliary operator was sent to check the status of the SG manway removal prior to reflooding the reactor coolant system and to verify RCS level from a visual standpipe. When manway covers were verified to be in place, the reactor coolant system was flooded by gravity feed from the RWST, and one RHR pump was restarted to establish flow through the core.

G. Safety system responses:

None

III. Cause of Event

A. Immediate cause:

PG&E concluded that actual vortexing began at the conclusion of the drain-down to midloop which was accomplished earlier during the day shift on April 10, 1987. During the drain-down, RCS water level was lowered several inches below the midloop point of 107 foot. The RHR flow of 3,000 gallons per minute and the low RCS water level resulted in the formation of a vortex that continued until the RHR pump was removed from service. An analysis of the reactor vessel refueling level indication system (RVRLIS) indications during this period concluded that RCS water level was not raised high enough to stop the vortex.

Since the charging system flow meter was inoperable, the operator had to use RCS level and VCT level to balance charging and letdown flows. Due to the VCT's smaller surface area, flow imbalances were noted in the VCT level considerably faster than in the RCS level; thus operators relied heavily on the VCT level to balance letdown and charging flow.

The RCS inventory loss that resulted from the leaking isolation valve lowered the VCT level. Believing an imbalance had developed between letdown flow and charging flow, operators increased letdown flow and thus reduced RCS water level, increasing the vortex effect in such a way that air entrainment resulted in pump motor amperage fluctuations. The operator decided to secure the pumps to prevent possible damage to them.

B. Root cause:

1. Vortex conditions were not thoroughly understood by operating personnel.

2. Personnel other than operators were operating valves to drain and vent a system (engineer conducting leakrate test) inside the clearance boundary.

3. Inadequate communications:

   a. The engineer performing the leak rate test did not inform the operator.

   b. Inadequate communication caused delays in identification and closure of the leak, in reporting the status of SG manways, and in direct visual verification of RCS water level via the tygon tube.

IV. Analysis of Event

Analysis shows that in the absence of recovery actions, no core damage would have occurred for about 47 hours after the initial loss of RHR. Analyses performed by Westinghouse show that the rapid heatup and cooldown associated with this event did not reduce the integrity of the reactor vessel. In addition, the amounts of reactor coolant lost to containment via steam from the head vent and via the steam generator manway were small, and releases to the atmosphere were well within that allowed by the Code of Federal Regulations. Thus the health and safety of the public were not affected by this event.

V. Corrective Actions

1. The procedure for draining the reactor coolant system, OP A-2:II, has been revised to (1) strengthen the requirements for maintaining the containment building in a condition such that pathways to the environment during midloop operations are capable of being closed in a timely manner; (2) require that the narrow range RVRLIS be in service prior to entering midloop operations; (3) provide precautions relating RHR flow to RCS level to preclude significant air entrainment due to vortex formation; and (4) provide checklists and walkdowns to ensure proper alignment of the reactor head and RVRLIS vent systems. PG&E is evaluating a change to OP A-2:II to provide guidance on RHR flow reduction to a value (to be determined with Westinghouse) consistent with adequate decay heat removal and other considerations.

2. The abnormal procedure for loss of RHR flow, OP AP-16, has been revised to (1) include requirements for not starting the second RHR pump if the first pump cavitates until adequate reactor vessel level is restored (OP A-2:II also provides guidance on this topic); (2) provide a table whereby the operator may determine the amount of time until the RCS will reach 200 degrees Fahrenheit without forced flow; (3) include recovery actions to be taken in the event that RCS level or flow decreases below acceptable values; (4) require that major pathways to the environment be closed if RHR flow is interrupted; and (5) provide contingencies for RCS feed and bleed if forced RHR flow cannot be reestablished.

3. Emergency Procedure EP G-1, "Accident Classification and Emergency Plan Activation," has been revised to require operators to declare an unusual event if RHR flow is not restored within 10 minutes. In addition, an alert will be declared if RCS temperature exceeds 200 degrees Fahrenheit (measured or projected). In addition, an alert will be declared if RHR flow is not restored within one hour.

4. A narrow range level transmitter was installed on Unit 2 before resumption of midloop operations after the April 10, 1987, interruption of RHR flow event, to sense the loop 3 hot leg level, referenced to the reactor vessel head. This provided indication in the control room through an accumulator level indicator with high and low alarm capabilities. Also, wide range indication sensing of the loop 2 crossover leg level, referenced to the pressurizer vapor space, has been added to Unit 2 and will be added to Unit 1. This is indicated in the control room through an accumulator level indicator with high and low alarm capabilities.

5. Additional training has been given to operating crews with respect to midloop operations as described in OPs A-2:II and AP-16. The training also covered RCS venting and RHR system venting (pumps and piping). Training for the midloop mode of operation will be included in the formal operator training program.

6. All outage activities were reviewed to identify any work item that might have provided a drain path from the RCS or opened a vent path between containment atmosphere and the environment. Work on these items was deferred until midloop operations were complete.

7. A knowledgeable engineer or manager was put on shift for midloop operations during this outage.

VI. Additional Information

A. Failed components:

Valves

1. CVCS-2-8396A

   Type: Diaphragm
   Manufacturer: Grinnell
   Model No.: 2466-10-M

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2. CVCS-2-8380

   Type: Diaphragm
   Manufacturer: Grinnell
   Model No.: 2466-10-M

B. Previous LERs on similar events:

   None

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PACIFIC GAS AND ELECTRIC COMPANY

PG&E 77 BEALE STREET • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-4211 • TWX 910-372-6587

October 29, 1987

PG&E Letter No.: DCL-87-256

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Re: Docket No. 50-323, OL-DPR-82
Diablo Canyon Unit 2
Licensee Event Report 2-87-005-01
Interruption of RHR Flow During RCS Midloop Operation

Gentlemen:

Pursuant to 10 CFR 50.73(a)(2)(v), PG&E is submitting the enclosed revision to Licensee Event Report 2-87-005 concerning the interruption of residual heat removal (RHR) flow during reactor coolant system (RCS) midloop operation. This revision is being submitted to provide additional information on the event, its cause, and corrective actions taken to preclude recurrence. This event did not affect 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,

W.B. Kaefer for

J. D. Shiffer

Enclosure

cc: J. B. Martin
M. M. Mendonca
P. P. Narbut
B. Norton
B. H. Vogler
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