CCN 92-14129

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

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KEN POWERS
PLANT MANAGER

November 12, 1992

Docket No. 50-277

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Licensee Event Report
Peach Bottom Atomic Power Station - Unit 2

This LER concerns a Technical Specification violation due to Main Steam Relief and Safety Valve setpoint drift.

Reference:	Docket No. 50-277
Report Number:	2-92-021
Revision Number:	00
Discovery Date:	10/15/92
Report Date:	11/12/92
Facility:	Peach Bottom Atomic Power Station RD1, Box 208, Delta, PA 17314

This LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(i)(B).

Sincerely,

Ken Powers

cc: J. J. Lyash, US NRC Senior Resident Inspector
T. T. Martin, US NRC, Region I

JE22

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Technical Specification Violation due to Main Steam Relief and Safety Valve Setpoint Drift

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)
1 0	1 5	9 2	9 2	0 2 1	0 0	1 1	1 2	9 2		0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
Albert A. Fulvio, Regulatory Supervisor	7 1 7 4 5 6 - 7 0 1 4

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

CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

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

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On 10/15/92, Peach Bottom Atomic Power Station personnel identified that two of the eleven MSRVs and one of the two SVs did not lift within the Tech Spec 1% tolerance. Two MSRVs lifted within the range of - 2.1% to + 2.3% of the nameplate setpoint while one SV lifted at 4.2% below its nameplate setpoint. The cause of the event has been determined to be that two MSRVs and one SV as found setpoints were not within the 1% tolerance of their nameplate setpoints. This is an industry wide drift concern with these valves. No actual safety consequences occurred as a result of this event. Refurnished valves have been properly setup at the test facility and then installed for five MSRVs and one SV. Appropriate corrective actions to resolved the concern of setpoint drift are being developed through an industry program being coordinated by a Boiling Water Reactor Owners Group Committee. Previous similar events have been identified.

Requirements of the Report

This report is submitted pursuant to 10 CFR 50.73 (a)(2)(i)(B) as a result of a Technical Specification (Tech Spec) violation due two Main Steam Relief Valve (MSRV) (EIIS:RV) and one Safety Relief Valve (SV) setpoints drifting out of tolerance.

Unit Conditions at Time of Discovery

Unit 2 was in the "REFUEL" mode at 0 % of thermal reactor (EIIS:EA) power. There were no systems, structures, or components that were inoperable that contributed to the event.

Description of the Event

On 10/15/92, Peach Bottom Atomic Power Station personnel reviewed ST-M-01G-450-2 "Main Steam Safety and Relief Valve Replacement" which documents compliance with ASME Code Testing and Tech Spec section 2.2.1.B and 2.2.1.C. It was identified that two of the eleven MSRVs and one of the two SVs did not lift within the Tech Spec ± 1 % tolerance. Per Tech Spec 4.6.D.1, "At least one safety valve and 5 relief valves shall be checked or replaced with bench checked valves every 24 months. All valves will be tested every two cycles. The set point of the safety valves shall be as specified in Specification 2.2". Therefore five of the eleven MSRVs and one of the two SVs had been removed during the eighth Unit 2 Refueling Outage to support testing at the Westinghouse Safety Valve Test Facility and then refurbished. Two MSRVs lifted within the range of -2.1 % to +2.3 % of its nameplate setpoint while one SV lifted at 4.2 % below its nameplate setpoint. The attached table provides the test results.

This condition resulted in a Tech Spec violation of section 2.2.1.B and 2.2.1.C because two MSRVs and one SV were not within the 1% tolerance of their nameplate set points. End-of-cycle testing is performed to determine whether the MSRVs and the SVs are in compliance with Tech Spec section 2.2.1.B and 2.2.1.C. However, this testing does not provide information as to when the valves may have failed to satisfy the Tech Spec drift limits.

Reactor over pressure protection is provided by the nuclear pressure relief system which includes eleven pilot operated MSRVs manufactured by Target Rock Corporation and supplied by General Electric (GE). Nominal set pressures for the MSRVs are distributed as follows: four at 1105 psig, four at 1115 psig, and three at 1125 psig. In addition, there are two Svs with an opening setpoint of 1230 psig. The safety function of these valves is to prevent steam pressure excursions from causing the reactor coolant system pressure to exceed the ASME design pressure rating 1375 psig.

Cause of the Event

The cause of the event has been determined to be that two MSRVs and one SV as found set points were not within the 1% tolerance c their nameplate set points. There is an industry wide concern with drifting setpoints on these and similar relief valves.

Analysis of Event

No actual safety consequences occurred as a result of this event.

The consequences are considered minimal due to the fact that MSRV setpoint drift would have had no impact on either the ADS function or the manual actuation mode of the MSRVs. In the case of an over pressure condition, plant procedures instruct the Reactor Operator (RO) to reduce reactor pressure via manual MSRV operation. If reactor pressure increases above 1040 psig, a reactor high pressure alarm actuates and a reactor scram is automatically initiated if reactor pressure increases above 1055 psig. In the event that reactor pressure continues to increase, the RO has manual control of the MSRVs. A safety evaluation has concluded that reactor pressure could not have exceeded the ASME design pressure rating of 1375 psig. In addition, the as-found MSRV and SV setpoints were below the ASME Code +3% tolerance.

Corrective Actions

Refurbished valves have been properly setup at the test facility and then installed for five MSRVs and one SV.

Appropriate corrective actions to resolve the concern of set point drift are being developed through an industry program being coordinated by a Boiling Water Reactor Owners Group (BWROG) Committee. The Philadelphia Electric Company (PECO) is participating in the committee to resolve concerns with setpoint drift. As further information regarding corrective actions becomes available throughout the industry, these measures will be reviewed and implemented as appropriate.

Previous Similar Events

Previous similar events have been experienced which is consistent with industry experience. The corrective actions addressed above should resolve this concern.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Table: Pressure Setpoint Test Results

MSRV #		Nameplate Setpoint (psig)	As Found Setpoint (psig)	% Drift
MSRV-71A	(ADS)	1125 ±1%	not tested	not tested
MSRV-71B	(ADS)	1125 ±1%	not tested	not tested
MSRV-71C	(ADS)	1105 ±1%	1105	≤1.0
MSRV-71D		1105 ±1%	not tested	not tested
MSRV-71E		1105 ±1%	not tested	not tested
MSRV-71F		1105 ±1%	not tested	not tested
MSRV-71G	(ADS)	1115 ±1%	1112	≤1.0
MSRV-71H		1115 ±1%	not tested	not tested
MSRV-71J		1115 ±1%	1116	≤1.0
MSRV-71K	(ADS)	1125 ±1%	1101	-2.1
MSRV-71L		1115 ±1%	1141	+2.3
SV-70A		1230 ±1%	not tested	not tested
SV-70B		1230 ±1%	1178	-4.2

Note: Automatic Depressurization System (ADS)