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U. S. Nuclear Regulatory Commission
Washington, DC 20555

Docket Nos. 50-277

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

This LER concerns setpoint drift associated with three Main Steam Relief Valves and two Safety Valves.

Reference:	Docket Nos.	50-277
Report Number:	2-94-010	
Revision Number:	00	
Discovery Date:	10/24/94	
Report Date:	11/22/94	
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)(vii).

Sincerely,

Garrett D. Edwards

GDE/GAJ:gaj
enclosure

cc: R.A.Burricelli, Public Service Electric & Gas
R. R. Janati, Commonwealth of Pennsylvania
INPO Records Center
T. T. Martin, US NRC, Administrator, Region I
R. I. McLean, State of Maryland
W. L. Schmidt, US NRC, Resident Inspector
A. F. Kirby III, DelMarVa Power
H. C. Schwemm, VP - Atlantic Electric

CCN 94-14175

IE22

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)
Peach Bottom Atomic Power Station - Unit 2	0 5 0 0 0 2 7 7	1 OF 0 4

TITLE (4): Main Steam Relief And Safety Valve Setpoint Drift

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
1 0	2 4	9 4	9 4	0 1 0	0 0	1 1	2 2	9 4		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): 1 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)	[X] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[ ] 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	
Anthony J. Wasong, Manager, Experience Assessment	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	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 (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On 10/24/94, following the completion of the tenth Unit 2 Refueling Outage, Peach Bottom Atomic Power Station personnel reviewed ST-M-01G-450-2 "Main Steam Safety and Relief Valve Replacement". This test documents the as found setpoint of these valves at the completion of the operating cycle in compliance with ASME Code Testing requirements and the Technical Specifications (Tech Spec). Test data indicated that three of the eleven MSRVs and two of the two SVs did not lift within the Tech Spec ± 1 % tolerance specified in Tech Spec Section 2.2.1.B and 2.2.1.C. The three MSRVs that were beyond their 1 % tolerances lifted within the range of -1.5 % to + 2.0 % of their nameplate setpoints. The two SVs lifted within the range of -2.52 % to -4.47 % below their nameplate setpoint. The ability of the MSRVs and the SVs to successfully maintain a 1 % tolerance is an industry wide concern. The event was caused by a small drift in the setpoint of the valves which is inherent in the valve's design. Refurbished valves have been properly setup at the test facility and were installed for these MSRVs and SVs. In addition, a Technical Specification change request is being evaluated by the PECO Energy Company to address MSRV and SV setpoint tolerance relaxation. No actual safety consequences occurred as a result of this event. Previous similar events have been experienced.

Requirements of the Report

This report is submitted pursuant to 10 CFR 50.73 (a)(2)(vii) as any event where a single cause or condition caused two independent trains to become inoperable in a single system. This occurred when the setpoints on three Main Steam Relief Valves (MSRV) (EIIS:RV) and two Safety Valves (SV) drifted out of tolerance.

Unit Conditions at Time of Discovery

Unit 2 was in the "RUN" mode at 100 % 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 10/24/94, following the completion of the tenth Unit 2 Refueling Outage, Peach Bottom Atomic Power Station personnel reviewed ST-M-01G-450-2 "Main Steam Safety and Relief Valve Replacement". This test documents the as found setpoint of these valves at the completion of the operating cycle in compliance with ASME Code Testing requirements and the Technical Specifications (Tech Specs).

Test data indicated that three of the eleven MSRVs and two of the two SVs did not lift within the Tech Spec ± 1 % tolerance specified in Tech Spec Section 2.2.1.B and 2.2.1.C. 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 setpoint of the safety valves shall be as specified in Specification 2.2". All of the MSRVs and the SVs were removed during the tenth Unit 2 Refueling Outage for testing and refurbishment. The three MSRVs that were beyond their 1 % tolerances lifted within the range of -1.5 % to + 2.0 % of their nameplate setpoints. The two SVs lifted within the range of -2.52 % to -4.47 % below their nameplate setpoint. The attached table provides the test results. All of these valves had been within tolerance when installed.

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 setpoint 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 of 1375 psig.

Cause of the Eve .

The ability of the MSRVs and the SVs to successfully maintain a 1 % tolerance is an industry wide concern. The event was caused by a small drift in the setpoint of the valves which is inherent in the valve's design.

Analysis of Event

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

The consequences are considered minimal due to the fact that an analysis concluded that reactor pressure could not have exceeded the ASME design pressure rating of 1375 psig with the MSRVs and SVs in this condition. In addition, MSRV setpoint drift would have had no impact on either the Automatic Depressurization System 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.

Corrective Actions

Refurbished valves have been properly setup at the test facility and were installed for these MSRVs and SVs.

In addition, a Technical Specification change request is being evaluated by the PECO Energy Company to address MSRV and SV setpoint tolerance relaxation. Corrective actions to resolve the concern of setpoint drift will be implemented as appropriate.

Previous Similar Events

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

NRC FORM 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

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

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Peach Bottom Atomic Power Station Unit 2	0 5 0 0 0 2 7 7	9 4	— 0 1 0	— 0 0	0 4 OF 0 4

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

Table: Pressure Setpoint Test Results

MSRV & SV #	Nameplate Setpoint (psig)	As Found Setpoint (psig)	% Drift
MSRV-71A (ADS)	1125 ± 1%	1134	< 1.0
MSRV-71B (ADS)	1125 ± 1%	1121	< 1.0
MSRV-71C (ADS)	1105 ± 1%	1101	< 1.0
MSRV-71D	1105 ± 1%	1125	+ 1.8
MSRV-71E	1105 ± 1%	1095	< 1.0
MSRV-71F	1105 ± 1%	1088	- 1.5
MSRV-71G (ADS)	1115 ± 1%	1106	< 1.0
MSRV-71H	1115 ± 1%	1105	< 1.0
MSRV-71J	1115 ± 1%	1124	< 1.0
MSRV-71K (ADS)	1125 ± 1%	1120	< 1.0
MSRV-71L	1115 ± 1%	1138	+ 2.0
SV-70A	1230 ± 1%	1175	- 4.47
SV-70B	1230 ± 1%	1199	- 2.52

Note: Automatic Depressurization System (ADS)