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U. S. Nuclear Regulatory Commission
Washington, DC 20555

Docket No. 50-278

SUBJECT: Licensee Event Report
Peach Bottom Atomic Power Station - Unit 3

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

Reference:	Docket No. 50-278
Report Number:	3-95-005
Revision Number:	00
Discovery Date:	11/08/95
Report Date:	12/08/95
Facility:	Peach Bottom Atomic Power Station 1848 Lay Road, 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, Senior Resident Inspector
A. F. Kirby III, DelMarVa Power
H. C. Schwemm, VP - Atlantic Electric

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CCN 95-14106

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 WTH 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 3	0 5 0 0 0 2 7 8	1 OF 0 4

TITLE (4): Main Steam Relief Valve Set Point 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 1	0 8	9 5	9 5	0 0 5	0 0	1 2	0 8	9 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

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

	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 1 0 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)	NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

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

On 11/08/95, it was identified that three of the eleven MSRVs did not lift within the Tech Spec ± 1 % tolerance. All of the MSRVs and the SVs were removed during the tenth Unit 3 Refueling Outage for testing and refurbishment. Three MSRVs lifted within the range of + 1.4 % to + 3.0 % of their nameplate setpoints. All of these valves had been within tolerance when installed. This resulted in a condition where more than 2 independent trains in a single system may not have been capable of performing their safety function. End-of-cycle testing is performed to determine whether the MSRVs and the SVs setpoints are in compliance with Tech Spec limits. The overall safety function of the MSRVs to provide over pressure protection was maintained. 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. The ability of the MSRVs to successfully maintain a 1 % tolerance is due to 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 then installed for all the MSRVs and SVs. No actual safety consequences occurred as a result of this event. Previous similar events have been identified.

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) drifted out of tolerance.

Unit Conditions at Time of Discovery

Unit 3 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 11/08/95, Peach Bottom Atomic Power Station personnel reviewed ST-M-01G-450-3 "Main Steam Safety and Relief Valve Replacement" which documents compliance with ASME Code Testing and Tech Spec Section 2.2.

It was identified that three of the eleven MSRVs did not lift within the Tech Spec ± 1 % tolerance required by Tech Spec 2.2. 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 3 Refueling Outage for testing and refurbishment. Three MSRVs lifted within the range of + 1.4 % to + 3.0 % of their nameplate setpoints. The attached table provides the test results. All of these valves had been within tolerance when installed.

This resulted in a condition where more than 2 independent trains in a single system may not have been capable of performing their safety function. This is due to the fact that three MSRVs were not within the 1% tolerance of their nameplate setpoint. End-of-cycle testing is performed to determine whether the MSRVs and the SVs setpoints are in compliance with Tech Spec section 2.2. The overall safety function of the MSRV/SVs to provide over pressure protection was maintained. 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 with the as found setpoints.

Cause of the Event

The ability of the MSRVs to successfully maintain a 1 % tolerance is due to 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 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. Had a design basis transient occurred, it has been concluded that there was no overall impact on the safety function of the MSRVs because the other MSRVs/SVs were found to be very precisely set at their nominal setpoints. This allowed the three MSRVs with offsets greater than 1% to be absorbed in the overall MSRV system 1% setpoint tolerance.

Corrective Actions

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

Previous Similar Events

Previous similar events (LERs 2-92-21, 3-93-08, 2-94-10) have occurred which is consistent with industry experience. The implementation of Improved Tech Specs is expected to help reduce future reportable events from occurring.

Table: Pressure Setpoint Test Results

MSRV & SV #	Nameplate Setpoint (psig)	As Found Setpoint (psig)	% Drift
MSRV-71A (ADS)	1125 ±1%	1144	+ 1.7
MSRV-71B (ADS)	1125 ±1%	1121	< 1.0
MSRV-71C (ADS)	1105 ±1%	1110	< 1.0
MSRV-71D	1105 ±1%	1116	< 1.0
MSRV-71E	1105 ±1%	1100	< 1.0
MSRV-71F	1105 ±1%	1120	+ 1.4
MSRV-71G (ADS)	1115 ±1%	1122	< 1.0
MSRV-71H	1115 ±1%	1110	< 1.0
MSRV-71J	1115 ±1%	1119	< 1.0
MSRV-71K (ADS)	1125 ±1%	1159	+ 3.0
MSRV-71L	1115 ±1%	1110	< 1.0
SV-70A	1230 ±1%	1227	< 1.0
SV-70B	1230 ±1%	1224	< 1.0

Note: Automatic Depressurization System (ADS)