CCN 90-14232

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION
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Delta, Pennsylvania 17314
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PEACH BOTTOM—THE POWER OF EXCELLENCE

December 19, 1990

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 revised LER is being submitted following a completed Failure Analysis
| on the Main Steam Isolation Drain Valves. The LER concerns the discovery of excessive as found Primary Containment leakage rate.

Reference:	Docket No. 50-277
Report Number:	2-90-003
Revision Number:	01
Event Date:	03/10/90
Report Date:	12/19/90
Facility:	Peach Bottom Atomic Power Station RD 1, Box 208, Delta, PA 17314

This LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(v) and 50.73(a)(2)(ii).

Sincerely,

John Doe

cc: J. J. Lyash, USNRC Senior Resident Inspector
W. T. Russell, USNRC, Region I

IE22
||

bcc: R. A. Burricelli, Public Service Electric & Gas
Commitment Coordinator
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D. M. Smith, Vice President PBAPS (2 copies)
J. Urban, Delmarva Power

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Peach Bottom Atomic Power Station - Unit 2

DOCKET NUMBER (2): 0 5 0 0 0 2 7 7

PAGE (3): 1 OF 0 3

TITLE (4): Discovery of Excessive Primary Containment As Found Leakage Rate Due to Incomplete Procedure

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 3	1 0	9 0	9 0	0 0 3	0 1	1 2	1 9	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 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)	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)	X 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: A. A. Fulvio, Regulatory Engineer

TELEPHONE NUMBER: AREA CODE 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 3/10/90 at approximately 1000 hours, surveillance testing indicated Primary Containment leakage rate limit (La), as established by Technical Specific .ions, had been exceeded due to excessive through seat leakage on Main Steam Line Drain Isolation Valves MO-74 and MO-77. No actual safety consequences occurred as a result of this event. The cause of the failure was that the manufacturer's required tolerances on critical dimensions were not included in the maintenance procedure. This resulted in high leakage through these valves from excessive clearance between the valve disc and seat assemblies when in the closed position. The valves were repaired and returned to service prior to start-up. The maintenance procedure has been revised to include manufacturers required tolerances. There were two previous similar events.

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Peach Bottom Atomic Power Station Unit 2	0 5 0 0 0 2 7 7	9 0	— 0 0 3 —	0 1	0 2 OF 0 3

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

Requirements for the Report

This report is required pursuant to 10 CFR 50.73(a)(2)(V) and (a)(2)(ii) because Primary Containment may not have been capable of controlling the release of radioactive material during design basis events.

Unit Status at Time of Discovery

Unit 2 was in cold shutdown due to a scheduled outage.

Description of Event

On 3/5/90 at approximately 1400 hours, during the performance of Local Leak Rate Test ST 20.029, an unacceptably high through seat leakage rate (greater than 125,000 cc/min) was discovered for the Main Steam (EIIS:SB) Drain Isolation Valves (EIIS:ISV) MO-74 and MO-77. Since the method of leak rate testing employed during this test involves pressurizing the volume between MO-74 and MO-77, individual leak rates for each valve could not be determined.

On 3/9/90, MO-74 was disassembled and manually gagged to prevent leakage. On 3/10/90, at approximately 1000 hours, ST 20.029 was again performed to determine if the boundary maintained by MO-77 was acceptable. When the test results again indicated a leak rate of greater than 125,000 cc/min, it was determined that the Primary Containment pressure boundary leakage rate limit (La), as established by Technical Specifications, had been exceeded. The La value for PBAPS Unit 2 is 125,417 cc/min. The exact amount of leakage was not determined because it was in excess of the upscale limit (125,000 cc/min) of the mass flow meter used during the test.

Cause of the Event

The cause of the failure was that the manufacturer's required tolerances on critical dimensions were not included in the maintenance procedure. This resulted in high leakage through these valves from excessive clearance between the valve disc and seat assemblies when in the closed position. These valves are manufactured by Anchor Darling and are type CCA-W8321811.

Analysis of the Event

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

In the event that an accident had occurred during the period of time these valves were degraded, La could have been exceeded thereby allowing offsite doses to be greater than those previously analyzed in the Updated Final Safety Analysis Report. A normally closed non-safety related motor operated valve (with a 1" restricting orifice in parallel) downstream of MO-74 and MO-77 could be made available to reduce the release rate.

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

Corrective Actions

MO-74 and MO-77 were rebuilt using new discs, and seat assemblies were machined to proper tolerances. These valves were leak tested and the leakage found to be within acceptable limits following the completion of the maintenance.

| The maintenance procedure has been revised to include the manufacturer's required
| tolerances for critical dimensions.

Previous Similar Events

There have been two previous similar LER's involving excessive through seat leakage (not in excess of La) on the Main Steam Drain Valves.

LER 2-86-15 reported, in part, excessive through seat leakage on MO-77. The cause of the excess leakage was attributed to normal valve wear, and the valve was reconditioned as appropriate.

LER 2-87-05 reported, in part, the excess through seat leakage on MO-74. This leakage was attributed to the accumulation of fine particles on the seating surface of the valve due to a previous replacement of an upstream valve. The valve was cleaned and returned to service satisfactorily.