CCN 92-14167

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

R. D. 1, Box 208

DELTA, PA 17314

(717) 456-7014

PEACH BOTTOM ATOMIC PHILADELPHIA ELECTRIC COMPANY POWER STATION

KEN POWERS
PLANT MANAGER

September 11, 1992

Docket Nos. 50-277
50-278

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

SUBJECT: Licensee Event Report
Peach Bottom Atomic Power Station - Units 2 and 3

This LER concerns a condition prohibited by the Technical Specifications when the core thermal power limit was exceeded due to feedwater flow inaccuracies as verified by Sodium tracer testing.

Reference:	Docket Nos. 50-277 50-278
Report Number:	2-92-014
Revision Number:	00
Event Date:	08/12/92
Report Date:	09/11/92
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)(i).

Sincerely,

Ken Powers

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

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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 - Units 2 and 3	0 5 0 0 0 2 7 7	1 OF 0 3

TITLE (4) Condition Prohibited by the Technical Specifications when the Calculated Core Thermal Power Limit was Exceeded

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)
0 8	1 2	9 2	9 2	0 1 4	0 0	0 9	1 1	9 2	Peach Bottom - Unit 3	0 5 0 0 0 2 7 8
										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)	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 Engineer	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 8/12/92, after review of the Unit 2 feedwater tracer tests, it was determined that the unit had exceeded a requirement specified in the Technical Specifications. Specifically, the unit was operated above its licensed limit of 3293 Megawatts thermal. The Station decided to take a conservative course of action and derate both PBAPS units by 5% in power. Subsequent Unit 3 feedwater tracer testing identified a similar condition on Unit 3. The cause of the event was a lower than actual feedwater flow input into the process computer due to a design verification error. Based on the results of the feedwater tracer tests on both units, the process computer software and feedwater flow controls will be changed as appropriate to incorporate the results of the feedwater tracer tests. No actual safety consequences occurred as a result of this event. No similar previous LERs were identified.

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

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

Requirements for this Report

This report is being submitted to satisfy the requirements of 10CFR50.73(a)(2)(i)(B) as a result of a condition prohibited by the Technical Specifications (Tech Specs).

Unit Conditions at Time of Discovery

Unit 2 and 3 have been in the RUN mode at 100% of rated thermal reactor (EIIS:RPV) power. There were no systems, structures, or components that were inoperable that contributed to this event.

Description of Event

On 8/12/92, after review of the Unit 2 feedwater tracer tests performed on 8/01/92, it was determined that the unit had exceeded a requirement specified in the Tech Specs. Specifically, the unit was operated above its licensed limit of 3293 Megawatts thermal (MWt). The feedwater tracer testing utilized a radioactive Sodium 24 tracer which was injected into the feedwater lines to allow calculation of the actual feedwater flow rates. Based on the feedwater tracer test data, the feedwater flow signal used to calculate core thermal power was lower than actual. This means that the maximum calculated core thermal power limit was exceeded when the calculated core thermal power exceeded approximately 99.2%. Subsequent Unit 3 feedwater tracer testing on 8/12/92 verified that the maximum calculated core thermal power limit was exceeded when the Unit 3 calculated core thermal power exceeded approximately 98.8%.

Based on preliminary results of the Unit 2 feedwater tracer tests on 6/10/92 and 6/12/92 in conjunction with an engineering study, the Station decided on 6/20/92 to take a conservative course of action and derate both PBAPS units by 5% power until such time that conclusive feedwater tracer test results could be obtained. The tests on 6/10/92 and 6/12/92 indicated that a difference existed between the indicated and actual feedwater flow rate. This created a difference between the calculated and the actual core thermal power. The feedwater flow rate is the dominant variable in the determination of core thermal power.

Cause of the Event

The cause of the event was a lower than actual feedwater flow input into the process computer. The feedwater flow rate is used by the process computer in a core thermal power calculation to generate core thermal power. An error in the measured feedwater flow resulted in an offset in the core thermal power.

The lower than actual feedwater flow input was a design verification error. This occurred during a modification in 1973 of the feedwater nozzle instrument tap locations. At this time, analytical recalibration data was provided and incorporated but a feedwater tracer test was not performed to validate the analytical data.

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

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

Analysis of the Event

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

The safety analyses which establish the basis for the operating license include accident analyses, containment response analyses, minimum critical power ratio (MCPR) safety limit evaluation, and transient analyses. The accident and containment response analyses, which are typically performed prior to plant startup, are conservatively evaluated at 3440 MWt (104.5% power). The transient analyses, which establish core thermal power limits, are evaluated on a cycle-by-cycle basis. These analyses are performed at rated core thermal power of 3293 MWt but include conservative adders to account for a 2% (102% power) uncertainty in core thermal power. Also evaluated on a cycle-by-cycle basis is the ASME overpressure analysis. This analysis is performed at 102% of rated core power. The MCPR safety limit evaluation (GETAB) accounts for a 1.76% uncertainty (low bias) in feedwater flow as well as an uncertainty in other parameters that influence the calculation of core thermal power. Thus, the safety analyses provide for a 2% uncertainty in core thermal power. Since the negative feedwater flow (and thus core power) bias was less than 2%, it is believed that no safety concern existed because of the inherent margins in the safety analyses.

Corrective Actions

On 6/20/92, both units were derated by 5% power. The Unit 2 and 3 feedwater tracer test results have been analyzed and new maximum core thermal power levels were established on each unit.

Based on the results of the feedwater tracer tests, the Unit 3 process computer software has been modified and the feedwater flow controls will be changed as appropriate. In addition, the Unit 2 process computer software and feedwater flow controls will be changed as appropriate during the upcoming Refueling Outage.

Previous Similar Events

No similar previous LERs were identified which involved exceeding the calculated core thermal power Tech Spec limit.