NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

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 6

TITLE (4) Condition Outside the Design Basis of the Plant which Resulted from a Modification to the Feedwater Heaters

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 2	3 1	8 7	8 7	0 3 1	0 2	0 8	0 1	8 8	PBAPS - Unit 3	0 5 0 0 0 2 7 8
										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)	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)	

X: 50.73(a)(2)(ii)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
W. C. Birely, Senior Engineer - Licensing Section	2 1 5 8 4 1 - 5 0 4 8

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)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

Abstract:

During the Unit 2 refueling outage in 1982 and the Unit 3 refueling outage in 1983, extraction steam block valves were installed on the 3rd, 4th and 5th feedwater heaters in each of the three feedwater strings. It was recently recognized that these valves have a common electrical feed and will fail closed on a loss of power. On December 31, 1987 with Unit 2 in Cold Shutdown and Unit 3 in the Refueling Mode with the core offloaded, it was determined that a loss of power to these block valves could result in a Loss of Feedwater Heating (LOFWH) event outside the design basis of the plant. Calculations performed assuming the failure indicate that there would have been no change in plant operating limits and that all design and licensing criteria would have been satisfied with no reduction in safety. In addition, neither unit experienced a loss of power to these block valves.

The cause of the event was an error during the design process. A review of a sample of other modifications made to nonsafety-related systems concluded that this was an isolated incident and not a programmatic problem. A plant modification which will preclude a single failure from resulting in a LOFWH event outside the design basis of the plant will be implemented on each unit prior to startup of that unit. This event is reportable pursuant to 10 CFR 50.73(a)(2)(ii)(B).

NRC Form 366 (9-83)

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Unit Conditions Prior to the Event:

Unit 2 - Cold Shutdown

Unit 3 - Refueling Mode with Core Offloaded

Description of the Event:

During the Unit 2 refueling outage in 1982 and the Unit 3 refueling outage in 1983, air-operated extraction steam block valves were installed on the 3rd, 4th and 5th feedwater heaters in each of the three feedwater strings for the purpose of protecting against turbine water induction.

While conducting a factory acceptance review of the new simulator in December of 1987, it was discovered that the block valves have a common electrical feed and will fail closed on a loss of power. On December 31, 1987, it was determined that loss of power to these block valves could result in a Loss of Feedwater Heating (LOFWH) event outside the design basis of the plant. The accident analysis in Section 14.5.2.3 of the Updated Final Safety Analysis Report (UFSAR) assumes a loss of 100 degree F of the feedwater heating capability. Conservative estimates indicate a potential 130 degree F LOFWH event due to the isolation of extraction steam to the 3rd, 4th and 5th feedwater heaters in all three feedwater strings under the modification.

The EIIS codes for the affected systems are: JC-Plant Protection System, SE-Extraction Steam System, AC-Reactor Core and SJ-Feedwater System. The EIIS codes for the affected components are: HX-Heat exchanger (feedwater heater), TRB-turbine and V-valve (block valve).

Consequences of the Event:

The modifications were made prior to Cycle 6 operation of each unit. The discovery of the potential for a 130 degree F LOFWH event was made following completion of each unit's Cycle 7 operation. Using the NRC-approved methodology for analysis of cold water transients (REDY transient code), it was found that had the licensing basis been 130 degrees F for the LOFWH event for either unit in Cycles 6 or 7, there would have been no change in plant operating limits and that all design and licensing criteria would have been satisfied with no reduction in safety. In addition, neither unit experienced a loss of power to the extraction steam block valves during Cycles 6 and 7.

While analyzing the assumed 100 degree F LOFWH event for Units 2 and 3, Cycles 6 and 7, calculations using the REDY code indicated that no reactor scram would occur since the neutron flux did not increase enough to reach the trip setpoint. For the LOFWH event, the most severe delta CPR (change in critical power ratio) occurs when the neutron flux rises to just under the scram setpoint. The values of the surface heat flux at the scram level were used to determine the estimated delta CPR for the 130 degree F LOFWH event. These results are listed below.

Plant	Cycle	100 degrees F LOFWH delta CPR	Estimated 130 degrees F LOFWH delta CPR
PB2	6	0.150	0.154
	7	0.146	0.151
PB3	6	0.158	0.162
	7	0.149	0.160

The calculated MCPR (minimum critical power ratio) for the LOFWH events in Cycles 6 and 7 would increase by 0.01 or less for transients in the 100 to 130 degree F range. The REDY calculations also determined that the operating limit MCPR for Cycles 6 and 7 would not have changed had the LOFWH analyses been performed for any value of feedwater temperature reduction between 100 and 130 degrees F since the limits were established by more severe events (i.e. rod withdrawal error and load rejection without bypass).

The original licensing analyses for the upcoming Cycle 8 of both units indicated that, for a 100 degree F LOFWH event, the reactor would scram on high neutron flux. As previously stated, the most severe delta CPR for a LOFWH transient occurs when the neutron flux rises to just under the scram setpoint. Therefore, a 130 degree F LOFWH analysis would also result in a high neutron flux scram and would produce heat flux results no more severe than the 100 degrees F analysis.

Cause of the Event:

The cause of the event was an error during the design process. The consideration of the impact of this modification on a safety-related system was not properly addressed. An investigation indicated that the procedure, which was in place at the time the extraction steam block valve modification was reviewed, was not specific enough to promote an indepth evaluation of all potential safety concerns.

Corrective Actions:

Upon discovery of the potential condition outside the design basis of the plant, an investigation was conducted to assess the potential consequences. A plant modification will be implemented to preclude a single failure from resulting in a LOFWH event outside the design basis of the plant. The modification for each unit will be completed prior to unit startup.

Actions Taken to Prevent Recurrence:

A sample of nonsafety-related modifications was reviewed to determine whether this incident was an isolated occurrence or a programmatic problem. The scope of this review is defined in an Attachment to this LER. Thirty-nine modifications which were made to nonsafety-related systems were reviewed. For the samples selected, the modifications adequately addressed all safety concerns. Four of the 39 could have had an impact on safety if they had not been performed correctly. Based on this review, it was concluded that this incident was an isolated occurrence, and no further corrective actions are warranted. It should be noted that Philadelphia Electric procedures for the performance of safety evaluations, especially with regard to unreviewed safety questions, have been greatly enhanced since the time of this modification to the extraction steam block valves. In addition, Philadelphia Electric is presently developing corporate standards for 10 CFR 50.59 evaluations which will further enhance these procedures and their associated training.

Previous Similar Occurrences:

LER 3-86-15 concerned an electrical design error which affected the Reactor Water Cleanup System.

Tracking Code: D - Procedure Deficiency

Attachment
LER 2-87-31
Page 1 of 2

SCOPE OF PROGRAM FOR REVIEWING NON-SAFETY RELATED MODIFICATIONS

1. Sample Selection

a) The adequacy of the PECo engineering procedures for the performance of safety evaluations will be determined by reviewing the initial document and all subsequent revisions. The evaluation will determine when an adequate procedure became effective. The sample will be selected from modifications issued prior to this date.

b) The non-safety related systems involved in the transient and accident analyses included in Section 14 of the UFSAR are most likely to have an impact on safety related issues. Modifications to these systems will be reviewed.

c) Modifications which involve multi-discipline interface will be reviewed to determine the extent to which inadequacies in this area contributed to errors of this type.

d) A sample of 5-10% of the modifications satisfying these guidelines will be selected for review. However, no fewer than 20 and no more than 50 will be reviewed in the initial sample.

2. Review and Evaluation

For the samples selected, a review of the modification package, design documents, and safety evaluation will be performed to determine if the design adequately addressed all safety related concerns. If a selected modification cannot be completely reviewed due to lack of retrievable documentation, another sample will be selected and reviewed in its place. Modifications which cannot be reviewed for this reason will be identified and evaluated for appropriate actions.

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	8 7	— 0 3 1	— 0 2	0 6 OF 0 6

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

Attachment
LER 2-87-31
Page 2 of 2

The results of the initial sample review will be evaluated to determine if more problems were found of either a similar or different nature. This evaluation may result in reassessment of the root cause and/or expansion of the review population. This process will continue until it is determined that the extent of this condition is fully evaluated.

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E. P. FOGARTY
MANAGER
NUCLEAR SUPPORT DIVISION

10 CFR 50.73

August 1, 1988

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 revised LER concerns a plant modification which resulted in a condition outside the design basis of the plant.

Reference:	Docket Nos. 50-277 and 278
Report Number:	2-87-31
Revision Number:	02
Event Date:	December 31, 1987
Report Date:	August 1, 1988
Facility:	Peach Bottom Atomic Power Station RD 1, Box 208, Delta, PA 17314

This revised LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(ii)(B). Changes are indicated by a vertical bar in the right-hand margins. Significant changes appear in the "Abstract", "Cause of the Event" and "Actions Taken to Prevent Recurrence" sections. The purpose of this revision is to satisfy a commitment made in Revision 1 to the LER to report the results of a program to review a sample of nonsafety-related modifications.

Very truly yours,

E P Fogarty

E. P. Fogarty
Manager
Nuclear Support Division

cc: W. T. Russell, Administrator, Region I, USNRC
T. P. Johnson, NRC Senior Resident Inspector
T. E. Magette, State of Maryland
INPO Records Center

IE22
1/1