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U. S. Nuclear Regulatory Commission
Washington, DC 20555

Docket Nos. 50-277 & 278

SUBJECT: Licensee Event Report, Peach Bottom Atomic Power Station Unit 2 & 3

This LER concerns a Technical Specification violation when licensed thermal power was slightly exceeded due to unaccounted for control rod drive water flow.

Reference:	Docket No. 50-277 & 278
Report Number:	2-96-001
Revision Number:	00
Discovery Date:	1/18/96
Report Date:	2/20/96
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)(i)(B).

Sincerely,

JC McElwain for GDE

GDE\GAJ:gaj

enclosure

cc: B. Gorman, 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
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CCN#96-14013

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Peach Bottom Atomic Power Station Unit 2	05000 277	1 OF 4

TITLE (4)
Licensed Thermal Power Slightly Exceeded Due to Unaccounted for CRD Water Flow

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 NAME	DOCKET NUMBER
									PBAPS Unit 3	05000278
1	18	96	96	-- 001 --	00	2	20	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9):
POWER LEVEL (10):

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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 (Include Area Code)
Anthony J. Wasong	717-456-7014

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 15 single-spaced typewritten lines) (16)

On January 18, 1996 at approximately 1500 hours, engineering personnel determined that Unit 2 and Unit 3 had potentially operated marginally above 100% of rated Core Thermal Power (CTP) (i.e., 3458 Megawatts thermal power (MWt)). Specifically, a portion of the flow from the Control Rod Drive (CRD) system was not properly accounted for in the Nuclear Steam Supply System (NSSS) heat balance and CTP calculation. The CRD system flow not properly accounted for is approximately 6 gpm which may have resulted in an actual reactor power that exceeded indicated power by no more than approximately 1.05 MWt. No actual safety consequences occurred as a result of this event. The root cause for omission of the additional CRD flow to the CTP calculation and heat balance for the RCS pump seal injection flow, RWCU pump seal purge flow and reactor level instrumentation sensing lines was failure to consider the impact of this additional flow on core thermal power during the design process for these modifications. The computer program that calculates CTP has been modified to account for the additional CRD water flow into the reactor vessel. One previous similar event was identified.

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	05000277	95	-- 001 --	00	2 OF 4

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

Requirements of the Report

This report is submitted pursuant to 10 CFR 50.73 (a)(2)(i)(B) as a result of a condition prohibited by Technical Specification (Tech Spec) due to the shift average licensed power level being slightly exceeded.

Unit Conditions at Time of Discovery

Units 2 and 3 were in Mode 1 (RUN) at approximately 100% of thermal reactor power. There were no systems, structures, or components that were inoperable that contributed to the event.

Description of the Event

On January 18, 1996 at approximately 1500 hours, engineering personnel determined that Unit 2 and Unit 3 had potentially operated marginally above 100% of rated Core Thermal Power (CTP) (i.e., 3458 Megawatts thermal power (MWt)). Specifically, a portion of the flow from the Control Rod Drive (CRD) system was not properly accounted for in the Nuclear Steam Supply System (NSSS) heat balance and CTP calculation. The CRD system flow not properly accounted for is approximately 6 gpm which may have resulted in an actual reactor power that exceeded indicated power by no more than approximately 1.05 MWt.

Investigation into this event has identified the following previously unaccounted for sources of CRD water flow into the reactor vessel.

1. The General Electric (GE) design of the CRD system requires approximately 3 gpm to be provided to each Reactor Recirculation System (RCS) pump for seal staging flow. Approximately 0.75 gpm per pump is diverted to seal staging flow, resulting in an unmonitored 4.5 gpm flow to the reactor vessel through both pumps. A review of GE Nuclear Energy Group (GE-NEG) documentation and discussion with GE personnel indicated that the flow from the CRD system to the RCS pumps, historically, has never been considered in the NSSS heat balance and CTP calculations for any BWR plant. GE was unable to determine why this value was not considered. In addition, GE determined that there was no margin in the CTP calculation methodology which would offset the error introduced by this modification.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Peach Bottom Atomic Power Station Unit 2	05000 277	95	-- 001 --	00	3 OF 4

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

2. A modification installed in 1993 diverted approximately 0.04 gpm of CRD water to provide backfill to the reactor level instrumentation sensing lines. Since the tap for this CRD water supply is upstream of the CRD flow transmitter, this additional flow of CRD water to the reactor vessel is not accounted for in the CTP calculations.

3. A modification was completed in 1995 which diverted approximately 1.5 gpm of CRD water to provide seal purge flow to the Reactor Water Cleanup (RWCU) pumps. Again, since the tap for this CRD water supply is upstream of the CRD flow transmitter, this additional flow of CRD water to the reactor vessel is not accounted for in the CTP calculations.

Cause of the Event

The root cause for omission of the additional CRD flow to the CTP calculation and heat balance for the RCS pump seal injection flow, RWCU pump seal purge flow and reactor level instrumentation sensing lines was failure to consider the impact of this additional flow on core thermal power during the design process for these modifications.

A contributing cause to this event is an unawareness of the bases of the core thermal power calculation and NSSS heat balance on the part of modification design engineers. Another contributing cause is that the design input procedure which requires evaluation of system interactions does not specifically identify that core thermal power or NSSS heat balance inputs be considered.

Analysis of the Event

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

Plant accident and transient analyses account for power excursions above 100% power. Therefore, there was no impact on the ability of the plant to respond had a design basis event occurred. Additionally, no other plant reliability or functional concerns were created with this minor power excursion.

Corrective Actions

The computer program that calculates CTP has been modified to account for the additional CRD water flow into the reactor vessel.

An evaluation is being performed to thoroughly assess the complete heat balance calculation for units 2 and 3 to ensure that each addition and subtraction from the CTP is accounted for.

The PBAPS modification design process will be enhanced to specifically include consideration of changes to the core thermal power calculation and NSSS heat balance. In addition, training of modification design engineers will be developed to increase awareness of the potential impact on the core thermal power calculation and NSSS heat balance of design changes.

Previous Similar Events

One previous similar event was identified. LER 2-92-014 involved a Tech Spec violation when steady state power was exceeded due to uncertainties involving the measurement of feedwater flow. These uncertainties were in the unconservative direction and resulted in a slightly higher than allowable level of thermal power. Corrective actions involved adjusting the feedwater flow input into the process computer. Since the current event involves unaccounted for CRD water input to the reactor vessel, corrective actions from LER 2-92-014 could not have been expected to prevent this event.