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 4

TITLE (4) Design Deficiency that could permit Diesel Generator Trips during a Seismic Event

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	1 7	8 7	8 7	0 2 8	0 2	0 8	0 1	8 8	PBAPS 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)	0 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)	X 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)	50.73 (a)(2)(vi)
		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)
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: 2-87-28 Revision 2

A design deficiency was discovered which could result in Diesel Generator trips during a Loss of Offsite Power (LOOP) event. The Diesel Generator Room Carbon Dioxide Fire Suppression (Cardox) System control circuits, which are not classified as safety-related or seismic, could initiate Diesel Generator trip signals during a LOOP event if actuated by seismic conditions. There are four Diesel Generators common to Unit 2 and Unit 3, and each Diesel Generator could be tripped individually by the Cardox System. The original design does not prevent a seismic-induced diesel generator trip signal from the Cardox System. On December 17, 1987 it was determined that this condition was reportable pursuant to 10 CFR 50.73 (a)(2)(vi). The condition was discovered approximately one month earlier.

This condition compromised the ability to safely shut down the plant during a LOOP event concurrent with a seismic event. Both Peach Bottom units are shutdown. To correct this condition, the Cardox System will be upgraded with seismically qualified safety-related components to preclude inadvertent diesel generator trips. This modification will be complete prior to restart.

IE22 1/1

NRC Form 366 (9-83)

Description of the Event:

On December 17, 1987 it was determined that a recently discovered design deficiency was reportable pursuant to 10 CFR 50.73 (a)(2)(vi). The Diesel Generator Room Carbon Dioxide Fire Suppression (Cardox) System is not safety-related or seismically qualified. The Cardox System generates a diesel generator trip signal as a result of detection of a fire in a diesel generator room. The diesel generators are intentionally tripped upon actuation of the Cardox System because the diesel generators receive combustion air from within the diesel generator rooms and could draw in a significant amount of carbon dioxide resulting in reduced fire protection capability in the room and suffocation of the diesel engine. Various components of the Cardox System are not safety related or seismically qualified and could fail and cause a diesel generator trip signal to be generated or result in an inadvertent discharge of carbon dioxide into the diesel room possibly suffocating the diesel engine.

There are four Diesel Generators at Peach Bottom common to Unit 2 and Unit 3. Both Peach Bottom units are shutdown.

This condition was discovered by an investigation prompted by the discovery of a 10 CFR 50, Appendix R non-compliance at Limerick Generating Station, Unit 1 as reported to the NRC in LER 87-055 on Docket No. 50-352.

Additional time (beyond the required 30-day reporting period) was needed to determine the cause of the condition reported by this LER and to carefully assess the significance of the condition. Extensive engineering review of the "Significance of the Event" section was necessary to ensure that it accurately and completely addresses the requirements of 10 CFR 50.73. Further, it was determined late in the LER preparation process that finalizing the corrective actions would require additional engineering evaluation.

Significance of the Event:

The safety objective of the Diesel Generators and standby ac power supply and distribution system is to provide a reliable source of ac electrical power, independent of offsite sources, for the safe shutdown of the reactors. The condition being reported compromised that objective by posing a potential for tripping Diesel Generators at a time when they are needed. The probability that this condition could have actually impacted reactor safety is very small because a seismic event would have to have occurred shortly prior to or during a Loss of Offsite

Power (LOOP) event. Two independent and reliable offsite sources supply power to Peach Bottom. The sources are physically separated; therefore, a seismic event would not necessarily affect both sources.

When a LOOP occurs both reactors scram due to loss of power to the Reactor Protection System, resulting in a loss of auxiliary power from the main generators. Consequently, the only source of ac power to shut down the reactors would be the Diesel Generators. If a seismic-induced trip of the Diesel Generators was to occur, there would be no ac power available. This situation is referred to as a station blackout.

If a station blackout occurs during power operation or shortly after a reactor shutdown, reactor steam would be available to drive the Reactor Core Isolation Cooling (RCIC) System pump and/or High Pressure Coolant Injection (HPCI) System pump to control reactor level and, in conjunction with the Main Steam Relief Valves, control pressure. The Automatic Depressurization System (ADS), which uses five of the Main Steam Relief Valves, would be available to manually control reactor pressure. The HPCI, RCIC and Automatic Depressurization Systems use dc power from emergency batteries (except for the HPCI/RCIC turbine steam supply valves inside containment which are normally open and remain open after loss of ac power). It is expected that the Diesel Generator trip signals would be removed and the Diesel Generators would be placed in service before the emergency battery power was depleted.

By operating HPCI/RCIC, coolant is added to the reactor vessel while energy is removed with the steam that drives the HPCI/RCIC turbines. The reactor fuel would be protected from overheating in this manner. HPCI/RCIC could be cycled on and off to maintain sufficient coolant inventory until ac power is restored to the normal shutdown cooling systems. Fuel failure would not occur during this blackout scenario as long as coolant level is maintained above two-thirds active fuel.

If a station blackout occurs when the reactor is shut down and there is no reactor steam available, there would be no external systems available to remove decay heat or add coolant to the core. However, the heat-up and boiling of coolant inventory would protect the reactor fuel from overheating, as long as level does not decrease below two-thirds active fuel. It is expected that the Diesel Generator trip signals would be removed and the Diesel Generators would be placed in service before coolant level decreased below two-thirds active fuel.

Cause of the Event:

The cause of this condition is a deficiency in the original design. During the original system design it was not recognized that seismic-induced failure of control components or actuation of the Cardox System constituted a common cause which could make more than one Diesel Generator unavailable. Components of the Cardox System were not classified and procured as seismic or safety-related.

Corrective Actions:

To eliminate this design problem, the Cardox System will be upgraded with seismically qualified safety-related control components. This will prevent diesel generator trips resulting from a seismic event, will increase the reliability of the Cardox System and will decrease the potential for inadvertent actuation. This modification will be complete prior to restart.

EIIS Codes:

The EIIS Codes for the systems referred to in this LER are BJ (HPCI), BN (RCIC), BO (LPCI), BM (Core Spray), AC (Reactor Core), EK (Emergency Onsite Power Supply/Diesel Generators), LW (Cardox), KP (Fire Protection), SB (Main Steam/ADS), JC (Reactor Protection), CE (Reactor Water Cleanup) and FK (Switchyard/Offsite Power Sources). The EIIS Codes for the components referred to in this LER are P (pump), RV (relief valve), ISV (isolation valve), V (valve), TRB (turbine), RPV (reactor vessel), DG (diesel generator), BTRY (battery) and RLY (relay).

Previous Similar Occurrences:

LER 3-86-15 reported a design error associated with electrical wiring in the Reactor Water Cleanup System.

Tracking Code: B99 - Design Deficiency, general

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E. F. 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 is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(vi) and concerns a design deficiency which could have resulted in diesel generator trips during a Loss of Offsite Power event. Revision 0 and Revision 1 of this LER did not specify the final solution to the design problem because an engineering review was in progress. This revision of the LER provides the chosen corrective action to eliminate this design deficiency. This revision also clarifies the situation by providing more details and removing some details. Revisions (from Revision 1) are indicated by vertical bars in the page margins.

Reference:	Docket Nos. 50-277 and 50-278
Report Number:	2-87-28
Revision Number:	02
Event Date:	December 17, 1987
Revised Report Date:	August 1, 1988
Facility:	Peach Bottom Atomic Power Station RD 1, Box 208, Delta, PA 17314

Very truly yours,

E P Fogarty

E. P. Fogarty
Manager
Nuclear Support Division

cc: W. T. Russell, Administrator, Region I, USNRC
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IE22
1/1