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 8

TITLE (4) Relay Trip Causing Breaker Trip that Resulted in Containment Isolations and Inoperability of Fire Pumps

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	0 2	8 8	8 8	0 0 4	0 1	0 4	0 8	8 8		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)	X 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)	

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
X	E K	R L Y	G 0 8 0	N					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 7	2 9	8 8

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

Abstract:

On March 2, 1988 at 2104 hours while in Cold Shutdown, the E-224 emergency load center transformer supply breaker tripped due to the unexpected trip of an undervoltage relay, which caused outboard Group II and III primary containment isolations. The systems involved include Shutdown Cooling, Reactor Water Cleanup, Reactor Building/Refuel Floor Ventilation and Standby Gas Treatment. Both fire pumps were inoperable as a result of the breaker trip. A 'B' channel half scram signal was also generated. The breaker trip was caused by the unexpected trip of the undervoltage relay (General Electric Company, Model No. 12HGA14AH6A) on the E-22 bus, which feeds the E-224 load center. The relay coil sustained severe heat damage which apparently allowed the relay contacts to close again 30 minutes later, initiating closure of the breaker. The relay, or the relay coil will be shipped to the manufacturer for a failure analysis. This LER will be updated to provide more details about the failure within 45 days following receipt of the analysis results.

There were no adverse safety consequences associated with this event. A temporary feed was installed for the E-224 load center. The failed relay was replaced on March 6, 1988 at approximately 1700 hours, and the E-22 bus was returned to service. Actions to prevent recurrence will be determined following the failure analysis and will be included in the updated LER.

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

Unit Conditions Prior to the Event:

- Unit 2 in Cold Shutdown Condition. Reactor Coolant Temperature - 145 Degrees F.
- Diesel Engine Driven Fire Pump Blocked Out of Service.
- Shutdown Cooling System Out of Service, But Available.

Description of the Event:

On March 2, 1988 at 2104 hours, the E-224 emergency load center transformer supply breaker tripped due to the unexpected trip of the E-22 bus undervoltage relay, which caused several engineered safety feature actuations. When the E-224 breaker tripped, the "E-22 Bus Undervoltage" alarm annunciated, and a 'B' channel half-scram signal was actuated, Group II and III Primary Containment Isolation System (PCIS) outboard isolation occurred, and the 'C' Reactor Water Cleanup (RWCU) System pump tripped. The 'B' Reactor Protection System (RPS) motor generator (M-G) set tripped on loss of power which caused the half-scram signal and Group III PCIS Outboard Isolation. The 'B' RPS M-G set provides power to the 'B' channel RPS logic and PCIS Group III outboard logic, both of which are normally energized and actuate when de-energized. A Group III Outboard Isolation includes the start of the 'B' Standby Gas Treatment System (SBGTS) fan and filter train initiation. The fan did not start because its power source is through the E-224 breaker.

The Group II outboard isolation occurred because its logic, which is normally powered through the E-224 breaker, was de-energized when the breaker tripped. The 'C' RWCU pump tripped because one of its suction valves closed as part of the Group II isolation. The Attachment to this LER provides details of the system responses that occurred.

The E-224 breaker trip also rendered the motor driven fire pump inoperable by removing its power source. Technical Specification 3.14.A.3 requires that the Commission be notified by telephone within 24 hours and in writing within one working day if both fire pumps are inoperable. Because the diesel engine driven fire pump was blocked out of service at the time, the Commission was notified by telephone and in writing (letter from J. F. Franz to W. T. Russell dated March 3, 1988) as required. Technical Specification 3.14.A.3 also requires that a special report be submitted within 14 days. Accordingly, a report was sent to the NRC Document Control Desk on March 16, 1988.

At 2134 hours the E-224 breaker closed unexpectedly restoring power to all affected loads. At this time the cause for the breaker trip had not yet been determined. When the breaker closed, the 'B' SBGTS fan started because the outboard Group III Isolation signal which was initiated when the breaker tripped had not been reset.

At 2135 hours the 'B' RPS M-G set was connected to its alternate power source because the E-224 breaker was not considered reliable. Between 2150 hours and 2202 hours the half-scram signal and isolation signals were reset, the 'B' SBGTS fan was shutdown and the 'C' RWCU pump was returned to service. The affected systems were isolated for approximately one hour.

Investigation of the breaker trip revealed that the E-22 4kv emergency bus undervoltage relay (#127-16) had suffered severe heat damage. It was concluded that the trip of this relay caused the E-224 breaker trip. A temporary feed was installed to energize the E-224 load center from the E-424 load center, and the E-22 bus was subsequently removed from service.

It was determined that the 'B' RPS M-G set output breakers did not trip during this event contrary to the system's design. This failure is being investigated. The results of this investigation will be discussed in a revision to this LER (See Cause of Event Section regarding a revised LER).

Consequences of the Event:

The safety consequences of this event were minimal. Because Shutdown Cooling and RWCU isolated, the normal means of decay heat removal were temporarily unavailable. The isolations were reset and RWCU was returned to service within one hour. The decay heat load was very low because the reactor had been shutdown for eleven months and the core had been reloaded during that period. Shutdown Cooling was not in service at the time. RWCU was removing decay heat and controlling coolant level by operating in the "condenser dump" mode. Both fire pumps were unavailable for a short period of time. The motor driven pump was functional within 30 minutes.

If these isolations had occurred under the same circumstances when the decay heat load and coolant temperature were greater, the consequences would have been more significant. However, even in the unlikely event the isolations could not be reset, several alternative methods of coolant makeup and energy removal could have been used. As long as coolant inventory was maintained at

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normal levels, adequate core cooling would be provided. Station Procedure GP-12, "Core Cooling Procedure", outlines alternative sources of makeup, such as Condensate Transfer from stay-full lines, Core Spray from Condensate Storage Tank or Torus, Control Rod Drives (CRDs) from Condensate Storage Tank, and Residual Heat Removal (RHR) from Torus. The procedure also outlines several methods of energy removal including draining to the Torus through RHR piping. Each of the alternative methods of coolant makeup/energy removal listed above was available during this event. There are several other methods outlined in GP-12 which, depending on reactor pressure and equipment status, could be used under other circumstances.

If this event had occurred while at power, the consequences may have been more significant. The trip of the Reactor Building Ventilation Fans and loss of power to 'B' SBGTS Fan could have resulted in high main steam line tunnel temperature. Also, loss of power to main condenser hotwell make-up and reject valves could have resulted in condensate pump trips and, consequently, feedwater pump trips. Consequently, a scram would have been probable either due to high main steam line tunnel temperature or loss of feedwater flow. The Shutdown Cooling System would have been isolated; however, the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems would have been available for coolant make-up and decay heat removal following the scram. The isolation of RWCU, RHR Head Spray and Instrument Nitrogen, for a brief period, would not have resulted in adverse consequences. Head Spray is not required to mitigate the consequences of an accident. The Instrument Nitrogen Compressors would have been unavailable because the suction valve closed; however, the Instrument Air System would have been available as a backup. Reactor coolant chemistry would not be significantly affected by isolation of RWCU for several hours unless an abnormal chemistry transient occurred.

Cause of the Event:

It was concluded that the breaker trip was caused by the trip of the undervoltage relay (General Electric Company, Model No. 12HGA14AH5A) on the E-22 bus. It is not believed that an actual undervoltage condition existed because no other undervoltage relays that monitor the same voltage tripped. The relay coil was found to have sustained severe heat damage which apparently allowed the relay contacts to close again, initiating closure of the breaker. The relay, or the relay coil will be shipped to the manufacturer for a failure analysis. This LER will be updated to provide more details about the failure within 45 days following receipt of the analysis results.

Corrective Actions:

The failed relay was replaced on March 6, 1988 at approximately 1700 hours, and the E-22 bus was returned to service. The temporary power feed between the E-224 and E-424 load centers was then removed. The E-22 bus was out of service for approximately 3 1/2 days. The replacement relay was taken from the Unit 3 E-33 bus because a spare was not available. Since the Unit 3 reactor vessel was defueled, the Unit 2 E-22 bus loads were considered more important with regards to plant safety. Temporary feeds were installed to provide power to the E-33 bus loads. Replacement relays and relay coils are on order.

Actions to Prevent Recurrence:

The manufacturer's failure analysis results will be evaluated to assess the need for actions to prevent recurrence. If additional actions are identified, they will be incorporated into the updated LER.

EIIS Codes:

The EIIS Codes for the affected systems are BH (SBGTS), KP (Fire Protection), SD (Condensate), SJ (Feedwater), JC (RPS), BM (Core Spray), BJ (HPCI), BN (RCIC), AA (CRD), CE (RWCU), BO (RHR/Shutdown Cooling), SB (Main Steam), LD (Instrument Nitrogen (Air) Supply), JM (Primary Containment Isolation), WD (Liquid Radwaste), IG (Incore Monitoring (TIP)) and SH (Condenser Vacuum).

The EIIS Codes for the components involved are BKR (breaker), BU (bus), P (pump), ENG (engine), FAN (fan), RLY (relay), MO (motor), MG (motor-generator set), FLT (filter), RCT (reactor), CL (coil) and ISV (isolation valve).

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Previous Similar Occurrences:

No previous LERs concerning the failure of an HGA relay were identified. LERs 2-84-5 and 3-87-7 concern the failure of General Electric Type CR120 relays.

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Summary of System Responses

Valves That Isolated		PCIS Group
*MO-2-10-17	Outboard Shutdown Cooling Suction	II
*MO-2-10-18	Inboard Shutdown Cooling Suction	II
AO-2969B	Instrument Nitrogen Drywell Supply	II
AO-2968	Instrument Nitrogen Torus Supply	II
MO-2-12-18	Outboard Reactor Water Cleanup Suction	II
AO-2-20-95	Drywell Equipment Drain Sump	II
MO-2-20-83	Drywell Floor Drain Sump	II
*MO-2-12-68	Reactor Water Cleanup Return to Vessel	II
**MO-2-2-77	Outboard Main Steamline Drain	I
*MO-2-10-33	Outboard RHR Head Spray	II
*MO-2-14-71	Torus Water Filter Pump Suction	II
*AO-4235	Nitrogen Compressor Suction	III
*AO-2519	Drywell Purge Inlet	III
*AO-2521A	Torus Air Purge	III
*AO-2505	Air Purge Supply Inlet	III
*AO-2523	Drywell & Torus Makeup Inlet	III
*SV-2978 A thru G	Oxygen Analyzer Sample Lines	III
*AO-2507	Vent to SBGTS	III
*AO-2512	Torus Vent	III
*AO-2514	Torus Vent Relief	III
*AO-2510	Drywell Vent Relief	III
*SV-8101	Containment Atmosphere Dilution Gas Sample Line	III
AO-2462	Refuel Floor Exhaust	III
AO-2452	Refuel Floor Supply	III
AO-2457	Reactor Building Supply	III
AO-2464	Reactor Building Exhaust	III
AO-2468	Equipment Cell Exhaust	III

* These valves received isolation signals, but did not move because they were in the closed position for maintenance blocking purposes or as part of normal operations.

** The MO-2-2-77 is a Group I valve, but it also receives an isolation signal during a Group II outboard isolation.

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Other Equipment Responses:

The Traversing In-Core Probe (TIP) received a withdrawal signal (PCIS Group II); however, no movement occurred because the TIP was in the withdrawn position. The main condenser mechanical vacuum pump received a trip signal (PCIS Group I), but it was not operating. Normal Reactor Building/Refuel Floor Ventilation Fans tripped and the 'B' SBGTS fan received a start signal (PCIS Group III).

PHILADELPHIA ELECTRIC COMPANY
2301 MARKET STREET
P.O. BOX 8699
PHILADELPHIA, PA 19101
(215) 841-4000

April 8, 1988

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 concerns a breaker trip which caused containment isolations and resulted in the simultaneous inoperability of both fire pumps. It was discovered after submitting Revision 0, that one isolation valve involved was inadvertently not included in the LER. This revision corrects the error by adding AO-2514 to page 7 as indicated by a vertical bar in the margin. We apologize for this error and any inconvenience it may have caused.

Reference:	Docket No. 50-277
Report Number:	2-88-04
Revision Number:	01
Event Date:	March 2, 1988
Report Date:	April 8, 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)(iv).

Very truly yours,

R. H. Logue

R. H. Logue
Assistant to the Manager
Nuclear Support Division

cc: W. T. Russell, Administrator, Region I, USNRC
T. P. Johnson, USNRC Senior Resident Inspector

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