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February 7, 2003

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

Docket No. 50-277
SUBJECT: Licensee Event Report, Peach Bottom Atomic Power Station Unit 2

This LER reports an automatic scram of Unit 2 as a result of a failed circuit card in the Electro-Hydraulic Control (EHC) system. Also, this LER reports a condition involving the Reactor Core Isolation Cooling (RCIC) system not being fully operable as required by Technical Specifications. In accordance with NEI 99-04, the regulatory commitment contained in this correspondence is to restore compliance with the regulations. The specific methods that are planned to restore and maintain compliance are discussed in the LER.

Reference:	Docket No.	50-277
Report Number:	2-02-001	
Revision Number.	00	
Event Date:	12/21/02	
Report Date:	02/7/03	

Facility: Peach Bottom Atomic Power Station Unit 2
1848 Lay Road, Delta, PA 17314-9032

Sincerely,

Gordon L. Johnston, Plant Manager

GLJ/djf/CR137110

Enclosure

cc: PSE&G, Financial Controls and Co-owner Affairs
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INPO Records Center
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U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)

Circuit Card Failure Results in a Primary Containment Group I Isolation and Plant Scram

EVENT DATE (5) MO	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REV NO	REPORT DATE (7) MO	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER
12	21	02	02	0 0 1	00	02	07	03	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 1

POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check all that apply) (11)

	20 2201(b)		20 2203(a)(3)(ii)		50 73(a)(2)(ii)(B)		50 73(a)(2)(ix)(A)
	20 2201(d)		20 2203(a)(4)		50 73(a)(2)(iii)		50 73(a)(2)(x)
	20 2203(a)(1)		50 36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73 71(a)(4)
	20 2203(a)(2)(i)		50 36(c)(1)(ii)(A)		50 73(a)(2)(v)(A)		73 71(a)(5)
	20 2203(a)(2)(ii)		50 36(c)(2)		50 73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20 2203(a)(2)(iii)		50 46(a)(3)(ii)		50 73(a)(2)(v)(C)		
	20 2203(a)(2)(iv)		50 73(a)(2)(i)(A)		50 73(a)(2)(v)(D)		
	20 2203(a)(2)(v)	X	50 73(a)(2)(i)(B)		50 73(a)(2)(vii)		
	20 2203(a)(2)(vi)		50 73(a)(2)(i)(C)		50 73(a)(2)(viii)(A)		
	20 2203(a)(3)(i)		50 73(a)(2)(ii)(A)		50 73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steven C. Beck - Regulatory Assurance	(717) 456-3243

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	IT	ECBD	G 0 8 0	Y	E	JM	FSV	A610	Y

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)

At approximately 2035 on 12/21/02, Unit 2 automatically scrammed as a result of Main Steam Line Isolation Valve closure due to a Group I Primary Containment Isolation System (PCIS) actuation. The Group I PCIS actuation was a result of low main steam line pressure caused by the opening of Main Steam Line Bypass Valves due to an Electro-Hydraulic Control (EHC) circuit malfunction. PCIS Group II / III isolations were received as expected. A secondary containment isolation damper did not initially close due to a sticking solenoid valve. The redundant damper closed as designed. The High Pressure Coolant Injection and Reactor Core Isolation Cooling (RCIC) systems automatically started and injected water into the reactor vessel. RCIC was used to control reactor water level although the flow controller operated erratically which was subsequently determined to have caused RCIC to be inoperable resulting in a condition prohibited by Technical Specifications. Additionally, the reactor cool down rate of 100°F/hr was exceeded at 2205 hours on the reactor bottom head resulting in a Technical Specification Required Action entry. The Required Action entry was exited upon conclusion of a satisfactory thermal cycles analysis. The EHC control circuit malfunction was due to a component failure on a circuit card. Corrective actions include replacing the defective circuit card, replacing the secondary containment isolation damper solenoid valve, re-adjusting the RCIC flow controller, and reinforcing the monitoring of cool down rates to licensed operators.

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

Unit Conditions Prior to the Event

Unit 2 was in Mode 1 and operating at 100% rated thermal power when the event occurred. At the time of the event, there were no activities in progress related to Main Turbine Bypass Valves (EIIS: V) or Electro-Hydraulic Control (EHC) system (EIIS: IT), including pressure set adjustments and/or EHC testing. No actual change in reactor pressure occurred which would have initiated the EHC system to respond to control reactor pressure.

Description of the Event

At approximately 2035 on 12/21/02, Unit 2 automatically scrammed as a result of Main Steam Line Isolation Valve (MSIV) closure due to a Group I Primary Containment Isolation System (PCIS) actuation. The Group I PCIS actuation was a result of low main steam line pressure caused by the opening of Main Steam Line Bypass Valves (BPVs) due to an EHC control circuit malfunction.

PCIS Group II and III isolations were received, as expected, as a result of reaching the Level 3 reactor water level set point. One secondary containment isolation damper (EIIS: DMP) did not close immediately on the isolation. However, the redundant isolation damper closed thereby ensuring the safety function was met. At approximately 2036 hours, the damper that initially remained opened went to the desired closed position.

Reactor water level decreased to the Level 2 set point resulting in the automatic initiation of the Alternate Rod Insertion (ARI) system, the High Pressure Coolant Injection (HPCI) system, and the Reactor Core Isolation Cooling (RCIC) system. Additionally, both Reactor Recirculation pumps automatically tripped as designed due to the Level 2 reactor water level signal.

As a result of HPCI and RCIC automatically starting, reactor water level was restored. These systems automatically turned off as designed on the Level 8 high reactor water level signal. At approximately 2050 hours, RCIC (EIIS: BN) was restarted to control reactor water level. The RCIC controller (EIIS: TC) was observed to have an inappropriate response while controlling level in that the flow was fluctuating (a condition that was subsequently determined to be a Technical Specification inoperable condition). The RCIC controller was subsequently placed in the manual mode and the reactor level was controlled with RCIC in manual. Reactor pressure was controlled with a combination of Main Steam Relief Valve (MSRV) and HPCI operation (Condensate Storage Tank (CST) to CST mode).

Three Main Steam Line BPVs did not initially close when the EHC system was secured at approximately 2100 hours. This resulted in not being able to re-open the MSIVs. The re-opening of the MSIVs would have allowed for the use of the reactor feed water system for level control.

At approximately 2105, a normal reactor cool down was commenced with the goal of achieving cold shutdown (Reactor Mode 4). At approximately 2205, the cool down rate exceeded 100^{o}F/hr on the reactor bottom head (a rate of approximately 120^{o}F/hr). The appropriate Technical Specification Required Action was entered to initiate an analysis to determine the affect of this cool down rate had on the plant.

Description of the Event, cont.

The PCIS Group I / II / III isolations were reset by approximately 2150 hours. NRC prompt notifications were completed on 12/22/02 at approximately 0020 hours. Cold shutdown (Reactor Mode 4) was reached on 12/22/02 at approximately 0605 hours.

This report is being submitted pursuant to 10CFR50.73 (a)(2)(iv)(A) due to valid actuations of the Reactor Protection System, the Primary Containment Isolation System, the HPCI System, and the RCIC System.

This report is also submitted pursuant to 10CFR50.73 (a)(2)(i)(B) to report a condition prohibited by Technical Specifications due to RCIC being inoperable in excess of 14 days. The RCIC flow controller had been improperly adjusted during post modification testing in May 1994.

Analysis of the Event

There were no actual safety consequences as a result of this event.

All control rods inserted on the reactor scram signal. The Group I / II / III PCIS isolations resulted in the primary containment isolation safety function being met. Even though one secondary containment isolation damper did not close, the redundant damper closed, thereby accomplishing the safety function. The damper that did not initially close was promptly closed as a result of manual actions.

HPCI, ARI and the Recirculation Pump Trip safety functions operated as designed with no concerns noted.

Although RCIC was operated and performed the overall intended function of controlling reactor water level for this event, subsequent analysis of the oscillations of the RCIC flow resulted in RCIC being considered technically inoperable. Technical Specification Surveillance Requirement (SR) 3.5.3.3 requires that RCIC be capable of developing a flow rate greater than or equal to 600 gpm. The oscillations in flow cycled above and below this flow value. Therefore, the SR requirement was considered to be technically not met, resulting in an inoperable condition. However, since RCIC was still able to be used manually to fulfill the function of controlling reactor water level, RCIC was considered available.

Concerning the variance in the cool down rate, a thermal cycles analysis was performed. This analysis determined that there were no detrimental affects to the reactor coolant system as a result of exceeding the cool down rate. This analysis (required by Technical Specifications) was completed prior to plant startup and determined that the reactor coolant system was considered acceptable for continued operations.

Analysis of the Event, cont.

Main Steam Line BPVs #2, #6, and #8 remained opened after the EHC system was removed from service. Closing of these valves is not a plant safety function. In addition, the EHC system (which controls the opening and closing of the BPVs) is not a safety related system. Following an event where the MSIVs close, it is preferable to re-open the MSIVs once the PCIS Group I isolation signal is reset. This would allow for the use of the reactor feed water system for level control. However, other equipment was used to ensure that the water level and pressure control actions were accomplished. The BPVs re-closed once the EHC control system was repaired and placed back into service.

A Conditional Core Damage Probability (CCDP) study was performed. This analysis accounted for equipment difficulties encountered during the event. The results of this analysis determined that this event was not risk significant.

This event is bounded by the design basis event entitled, 'Pressure Regulator Failure'. During this event the plant safety systems responded as necessary. This event did not involve operations that exceeded the design basis. Even though there were equipment deficiencies, none involved significant unexpected system interactions or loss of safety function.

Cause of the Event

The cause of the Group I isolation and subsequent reactor scram was due to a failed Steam Line Resonance Compensation Card in the non-safety related Turbine EHC system. As-found voltages were taken on the actual installed circuit card that failed and a failure analysis was performed. This analysis indicated that the card was not operating properly. These conclusions were discussed with technical experts including General Electric (GE). The cause of the card failure was due to a failed component on the Steam Line Resonance Compensation Card (i.e. an Operational Amplifier - Model AD504). The card had been replaced during a recent refueling outage to enhance overall reliability of the EHC system. However, an unknown manufacturing defect caused the failure approximately 3 months after the card was in service. Subsequent analysis determined that generic defects existed with a particular batch of Model AD504 Operational Amplifiers. These are not safety related components.

The cause of the Secondary Containment damper failing to automatically isolate was a sticking solenoid valve (EIIS: FSV). It was determined that the preventive maintenance program for this valve was less than adequate.

The cause of RCIC controller malfunction was due to improper adjustment performed during a modification acceptance test subsequent to a modification of the controller in 1994. At this time, the gain on the controller was set too high as a result of attempting to optimize RCIC operation in the CST to CST mode. This resulted in adverse affects on the controller operation for the CST to Reactor Mode.

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

Cause of the Event, cont.

The cause of exceeding the cool down rate on the reactor bottom head was due to thermal stratification, which occurred when both reactor recirculation pumps tripped, combined with cool water being introduced into the bottom head from the Control Rod Drive system. Human performance errors in evaluating recorded temperatures during cool down trending contributed to this concern.

The cause of three BPVs remaining open when the EHC system was secured is due to less than adequate maintenance on the BPVs. Without EHC fluid pressure, springs exist to close the BPVs. The spring closure feature of the BPVs was not able to close the valves upon loss of fluid pressure due to friction within the valve, actuator and / or hydraulic binding without the EHC fluid as the primary driver.

Corrective Actions

The failed EHC card was replaced. The replacement card was tested satisfactorily and placed into service. Other similar cards that were replaced during the recent refueling outage were reviewed for similar defective components. No similar defects were determined to exist. A formal root cause analysis is in progress to determine underlying reasons for allowing a card with a defective component to be placed in service. Additional corrective actions will be generated as appropriate based on the results of the root cause analysis.

The RCIC controller was readjusted and tested satisfactorily. The Unit 3 RCIC controller was reviewed for similar concerns and it was determined that the controller was operable. Although considered operable, the Unit 3 RCIC controller gain was adjusted to further improve system performance. Other similar controllers in the HPCI systems for Units 2 and 3 were reviewed and they were determined to be properly adjusted.

The affected Secondary Containment damper solenoid valve was replaced and the damper was satisfactorily tested. A program has been initiated to replace solenoid valves on similar dampers and to routinely perform preventive maintenance on similar equipment.

A thermal cycles analysis of the reactor pressure vessel was completed with satisfactory results. The procedure used for recording temperature data during plant cool downs will be reinforced through training to operations personnel. The procedure used for monitoring cool down rates will also be reviewed and revised as necessary for usability enhancements.

The BPVs were lubricated and verified to operate properly with EHC fluid pressure. The maintenance program is being upgraded for the BPVs.

Previous Similar Occurrences

There were no previous events identified involving a failed EHC card due to a manufacturing defect that resulted in a Group I isolation and full reactor scram.