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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

Based on subsequent investigation and interviews, LER 2-99-006 is being revised to update the information around the cause of the heatup rate event.

This LER reports a Unit 2 reactor scram on September 30, 1999, resulting from a generator lockout condition and subsequent turbine trip. This LER also reports a Reactor Coolant System heatup rate being exceeded during the recovery period after the scram. Accordingly, the LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(iv) for the Engineered Safety Feature actuations following the turbine trip and the requirements of 10 CFR 50.73 (a)(2)(i)(B) for exceeding the heatup rate specified by the Technical Specifications.

Reference: Docket No. 50-277
Report Number: 2-99-006
Revision Number: 01
Event Date: 09/30/99
Report Date: 10/30/99

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

Sincerely,

Mark E. Warner, Plant Manager

MEW/dmw

enclosure

cc: N. J. Sproul, Manager, Financial Controls and Co-owner Affairs
R. R. Janati, Commonwealth of Pennsylvania
INPO Records Center
H. J. Miller, US NRC, Administrator, Region I
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CCN 99-14106

IE22

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to the industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1) Peach Bottom Atomic Power Station Unit 2	DOCKET NUMBER (2) 0500 277	PAGE (3) 1 of 5
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TITLE (4)
This LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(iv) for the Engineered Safety Feature actuations following the turbine trip and the requirements of 10 CFR 50.73 (a)(2)(i)(B) for exceeding the heatup rate specified by the Technical Specifications.

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
09	30	99	99	006	01	1	13	00	Facility Name	Docket Number
OPERATING MODE (9)		1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more) (11)							
POWER LEVEL (10)		100	20.2201(B)	20.2203(a)(2)(v)		X	50.73(a)(2)(i)		50.73(a)(2)(viii)	
			20.2203(a)(1)	20.2203(a)(3)(i)			50.73(a)(2)(ii)		50.73(a)(2)(x)	
			20.2203(a)(2)(i)	20.2203(a)(3)(ii)			50.73(a)(2)(iii)		73.71	
			20.2203(a)(2)(ii)	20.2203(a)(4)		X	50.73(a)(2)(iv)		OTHER	
			20.2203(a)(2)(iii)	50.36(c)(1)			50.73(a)(2)(v)		Specify in Abstract below	
			20.2203(a)(2)(iv)	50.36(c)(2)			50.73(a)(2)(vii)		or in NRC Form 336A	

LICENSEE CONTACT FOR THIS LER (12)

NAME Donna Wielgopolski	TELEPHONE NUMBER (include area code) 717.456.4634
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COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

Cause	System	Component	Manufacturer	Reportable to EPIX	Cause	System	Component	Manufacturer	Reportable to EPIX
B	EI	INVT	T248	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO	EXPECTED Submission Date (15)	Month	Day	Year
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ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On Thursday, September 30, 1999 at approximately 19:06 hours, with Unit 2 operating at 100 percent power, a generator lockout and subsequent turbine trip occurred that resulted in a reactor SCRAM. The turbine trip caused a high reactor pressure condition resulting in three Main Steam Relief Valves (MSRV) lifting and an Alternate Rod Insertion (ARI) initiation. Additionally, Primary Containment Isolation System (PCIS) Group II and Group III isolations occurred due to low reactor water level following the SCRAM.

The generator lockout occurred when the Generator Over Frequency relay actuated the Accidental Energization Protection generator lockout relay. The Generator Over Frequency relay was actuated by a transient ground on the DC Power System.

On Saturday, October 2, 1999, at approximately 04:20 hours, with Unit 2 in Mode 3, the Reactor Recirculation System piping experienced a heatup rate of approximately 170° F within 45 minutes. This exceeded the 100° F/hr heatup rate as specified in Tech. Spec. Section 3.4.9. This occurred while flushing the Residual Heat Removal (RHR) system to equalize temperatures in the 2A Reactor Recirculation piping prior to restarting the 2A recirculation pump. These activities were being performed as part of the SCRAM recovery activities.

There were no safety consequences from these events and all safety systems were available.

The ARI initiation, the lifting of MSRVs, and the PCIS isolations are reportable per 10 CFR 50.73 (a)(2)(iv). Exceeding the heatup rate of the recirculation line specified in Tech. Spec. Section 3.4.9 is reportable per 10CFR 50.73 (a)(2)(i)(B).

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TEXT (If more space is required, use additional copies of NRC form 336A) (17)

Requirements of the Report

This LER is being submitted in accordance with 10 CFR 50.73(a)(2)(iv) as a result of unplanned Engineered Safety Feature (ESF) actuations: PCIS group II and III isolations (EIS:JM), three Main Steam Relief Valves lifting, and initiation of Alternate Rod Insertion (ARI).

This LER is also being submitted in accordance with 10 CFR 50.73 (a)(i)(2)(B) for exceeding the heatup rate of 100 degrees F/hr specified in Tech. Spec. Section 3.4.9 for the Reactor Coolant System's recirculation line.

Unit Conditions at Time of Event

Unit 2 was in Mode 1 (RUN) at 100 percent power (EIS: EA) at the time of the reactor SCRAM. Later, Unit 2 was in Mode 3 (Hot Shutdown) when the recirculation line heatup rate was exceeded.

Unit 3 was in Mode 3 (Hot Shutdown), in preparation for a scheduled refueling outage.

Description of the Event

On Thursday, September 30, 1999, at approximately 19:06 hours, the Unit 2 reactor SCRAMMED from 100 percent power. The initiating event was the actuation of a main generator protective relay, 2-50H-381 (Generator Over Frequency Relay), and the subsequent actuation of a generator lockout relay, 2-50H-386J (Accidental Energization Protection). Relay 2-50H-381 was actuated by a transient ground on the DC System.

Prior to the event, at approximately 17:15 hours, an Equipment Operator noticed that the ground detection relay in Ground Detection Panel 2AD036 was "buzzing." DC ammeters, located at the ground detection panel, indicated that a slight positive ground was present on the Unit 2 'A' and 2 'C' DC System, though no alarm was present. The ground current was not great enough to energize the ground detection relay or the alarm. Troubleshooting activities were initiated to locate and isolate the DC ground.

At approximately 17:30 hours, the control valve position indication for the Unit 2 'B' and 2 'C' Reactor Feedpump Turbines (RFPT) changed from 5% to 10% open and from 10% to 30% open, respectively. Troubleshooting by plant operators immediately after identifying the change in control valve position indication determined that the 'offset' condition had no impact on the actual RFPT performance only the control valve position indication.

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At approximately 19:06 hours, Unit 2 automatically SCRAMMED. All systems responded appropriately following the SCRAM with the exception of the following:

- The 2 'B' and 2 'C' Reactor Feedwater Pump Turbines (RFPTs) were removed from service as part of the operator's normal scram actions but could not be immediately reset. The inability to reset the RFPTs was a result of the Woodward Governor logic sensing an open control valve (2 'B' and 2 'C' RFPT control valve position indication 'offset' identified at 17:30 hours). The 2 'A' RFPT was reset and used to maintain reactor water level as designed during this period of time.

During the event, at approximately 19:10 hours, the alarm for the "2 A/C Battery Ground" was received. The DC ammeters indicated that there was a neutral ground on the system.

To preserve information for the purpose of troubleshooting, various components were quarantined and remained in their tripped condition until troubleshooting was completed. The generator lockouts were not immediately reset. The recirculation pumps could not be immediately restarted due to the bottom head drain temperature indication being erroneous.

Approximately 1-1/2 days after the SCRAM at 04:20 hours on October 2, 1999, the 2 'B' Residual Heat Removal (RHR) System (EHS:BO) pump was attempted to be put into service to allow restoration of the 2 'A' recirculation pump. The 2 'B' RHR system was being placed into service to restore Reactor Coolant and Recirculation Pump Loop temperatures within the limits Technical Specification Surveillance Requirement 3.4.9.4. This Surveillance Requirement requires the temperature difference between the recirculation loop and the reactor pressure vessel coolant temperature to be less than or equal to 50 degree F. As a result of placing the 2 'B' RHR pump in service, a heatup rate of 170 degrees F in 45 minutes occurred on the 2 'A' recirculation piping. This exceeded the 100 degree F/hr heatup rate for the Reactor Coolant System (RCS) as specified in Technical Specification Limiting Condition for Operation (LCO) 3.4.9.

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Causes of the Event

The reportable events were caused by the following issues:

- 1) A transient DC ground on the Unit 2 A/C 250 VDC power supply system, actuated the Generator Over Frequency Relay. The investigation determined that an inverter in the Post Accident Monitoring Panel (20C722A) failed and induced a ground on the Unit 2 A/C 250 VDC power system. The failure of the inverter was the source of the transient ground actuation of the over frequency relay. The investigation validated the cause by simulating the relay actuation by momentarily subjecting the relay to a ground.
- 2) Due to a malfunction of the bottom head drain temperature indication, Technical Specification Surveillance Requirement (SR) 3.4.9.3 to verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is less than or equal to 145 degrees F could not be met. Because this SR could not be met, the 2 'A' recirculation pump could not be placed in service. Once indication was restored, reactor vessel temperature stratification resulted in the differential temperatures being outside the limits of Technical Specification SR 3.4.9.4.

Analysis of the Event

There were no safety consequences due to this event. During and following the transient, all systems performed as designed with the exception of the inability to immediately reset the 2 'B' and 2 'C' RFPTs.

The inability to immediately reset the 2 'B' and 2 'C' RFPTs challenged the operators, since reactor level control could only be accomplished with the 2 'A' reactor feedwater pump (RFP). This condition does not represent a significant safety issue as the designed safety systems, High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems were available for reactor water level control and only one RFP is required in this condition.

The lifting of MSRVs RV-2-07-071D, E and F, the initiation of Alternate Rod Insertion (ARI), and the Group II and Group III PCIS isolations, were expected ESF actuations for a turbine trip event. There were no abnormal circumstances for these ESF actuations and the equipment performed as designed.

The reactor coolant temperature transient was analyzed and determined to be well within the design thermal cycle for the affected components.

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Corrective Actions

The corrective actions associated with the reportable events are:

1. The inverter which caused the DC ground was replaced. A failure analysis of the inverter identified a diode which had shorted to the grounded chassis. The ongoing event investigation will assure that generic implications are identified and corrective actions implemented as appropriate.
2. A Temporary Plant Alteration (TPA) was installed to remove the Accidental Energization Protection lockout relay from service when the generator is synchronized to the power grid. The purpose of the circuit is to trip and lockout the generator if one of its output breakers is inadvertently closed while the generator is off-line. Thus, the appropriate procedures have been changed to reinstall the circuit upon shutdown of the generator. The same TPA will be installed on Unit 3 during startup from the current Refueling Outage. Further evaluations will be performed to determine long term corrective actions.
3. A review of a previous thermal transient analysis to the recirculation pipe, pump, and valves, where the temperature difference exceeded over 210 degrees F in an hour, determined that there was a negligible impact on the affected components.

The following corrective action was taken for the 2 'B' and 2 'C' RFPT control valve indication 'offset':

1. Investigation into position indication changes of the RFPT control valves, determined that the most probable cause was associated with an induced electrical signal. The cause could not be validated because the 'offset' condition went away shortly after the plant scram and could not be reproduced. Since the exact cause could not be determined, compensatory changes were made to the configuration of signal wire shielding to reduce the amount of noise on the system. Additionally, a new operating procedure, AO 6D.3-2, "Reactor Feed Pump Operation Following a Scram With a Positive Offset in Control Valve Position Indication," was written and approved. The procedure directs the operators on how to reset the RFPTs when control valve position indication is greater than actual position 'offset'. All operating crews have been briefed on the new procedure.

Previous Events

There were no previous events in which a DC ground caused the actuation of the Generator Over Frequency Protection Relay. However, LER 2-97-009 reported a momentary loss of DC voltage occurring when a battery charger was placed in service. This transient did actuate the same relay causing a reactor SCRAM.

In the last 2 years, there were no events in which the recirculation system piping and components exceeded the 100 degree F/hour heatup rate.