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10CFR 50.73

November 15, 2006

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Peach Bottom Atomic Power Station (PBAPS) Unit 3
Facility Operating License No. DPR-44
NRC Docket No. 50-277

Subject: Licensee Event Report (LER) 2-06-02

This LER reports a condition involving Automatic Depressurization Valve deficiencies that was discovered during a recent Refueling Outage. 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. If you have any questions or require additional information, please do not hesitate to contact us.

Sincerely,

Joseph P. Grimes
Plant Manager
Peach Bottom Atomic Power Station

JPG/djf/IR 539277/538660/537127

Attachment
cc: PSE&G, Financial Controls and Co-owner Affairs
R. R. Janati, Commonwealth of Pennsylvania
INPO Records Center
S. Collins, US NRC, Administrator, Region I
R. I. McLean, State of Maryland
US NRC, Senior Resident Inspector

CCN 06-14086

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose 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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Peach Bottom Atomic Power Station Unit 2	05000 277	1 OF 5

4. TITLE
Automatic Depressurization System Safety Relief Valve Deficiencies

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
09	28	2006	06	- 02 -	0	11	15	2006	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 5

10. POWER LEVEL: 0

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

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☒ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☒ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
PBAPS Unit 2, James Armstrong, Regulatory Assurance Manager	717-456-3351

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
E	LK	FSV	A613	Y	X	SB	RV	T020	Y

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

Based on a review of testing performed on Safety Relief Valves (SRVs) during the P2R16 Refueling Outage, Site Engineering personnel determined that the 71B and 71G SRVs did not meet their allowable leak rate for the pneumatic actuation controls for the Automatic Depressurization System (ADS) feature of the SRVs. Additionally, the 71C SRV, Serial Number (S/N) 83, did not properly re-close on the fourth actuation during laboratory testing. The cause of the 71B and 71G ADS SRV pneumatic leakage is attributed to leakage from each of the SRV's actuator diaphragm and solenoid valve. These leaks only occurred when the SRV solenoid valves were energized. The diaphragms and solenoid valves associated with the 71B and 71G ADS SRVs were replaced. As-left leak testing was performed and the valves were restored to an operable condition prior to plant startup from the P2R16 Refueling Outage. A refurbished SRV was installed in the 71C SRV location to replace the S/N 83 SRV. There were no actual safety consequences associated with this event. This event was determined to not be risk significant.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Peach Bottom Atomic Power Station, Unit 2	05000277	06	- 02	- 00	2 OF 5

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

Unit Conditions Prior to Discovery of the Event

Unit 2 was in Mode 5 for its 16th Refueling Outage when the event was discovered. Deficiencies with Safety Relief Valves (EIIS: RV) were discovered during the time period 9/27/06 – 9/28/06 based on testing performed during the Refueling Outage. Unit 2 had been recently shutdown on 9/15/06 for the P2R16 Refueling Outage. Prior to the shutdown, Unit 2 had been operating for the previous 367 days. At the time of discovery (Mode 5), the Safety Relief Valves were not required to be operable. There were no structures, systems or components out of service that contributed to this event.

Description of the Event

Based on a review of testing performed on Safety Relief Valves (SRVs) during the P2R16 Refueling Outage, Site Engineering personnel determined that:

1. The 71B and 71G SRVs did not meet their allowable leak rate for the pneumatic actuation controls for the Automatic Depressurization System (ADS) feature of the SRVs. The as-found leak rates for the 71B and 71G SRVs were documented as off-scale (only when the SRV was actuated) and therefore, exceeded the leak rate limit of 100 cc/min.
2. The 71C SRV, Serial Number (S/N) 83, did not properly re-close on the fourth actuation during laboratory testing. This ADS SRV actuated and re-seated properly on the first three laboratory tests involving actuation on over-pressure. The fourth actuation was performed using the pneumatic (manual) system. The valve failed to re-close after this actuation.

This report is being submitted pursuant to 10CFR 50.73(a)(2)(i)(B) involving a condition prohibited by Technical Specifications and 10CFR 50.73(a)(2)(vii) involving multiple inoperable trains in a single system due to a single condition. Because there were deficiencies identified with multiple ADS SRVs, there is an indiciation that the discrepancies occurred over a period of time. Therefore, in accordance with NUREG-1022, this event is considered as reportable as a condition prohibited by Technical Specifications. Technical Specification 3.5.1 requires that the ADS function of five SRVs be operable. If one ADS valve is inoperable, Technical Specification 3.5.1, Condition F requires the valve to be restored to an operable status within 14 days. If two or more valves are inoperble, Required Action H requires the plant to be brought to Mode 3 in 12 hours.

This report is also being submitted pursuant to 10CFR 50.73(a)(2)(v) involving a condition that could have prevented the fulfillment of a safety function. Technical Specification 3.5.1 Bases state that at least four ADS SRVs are required to provide the required depressurization. Two SRVs (71B and 71G) had pneumatic system leak rates that exceeded leak rate limits.

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

Cause of the Event

The cause of the 71B and 71G ADS SRV pneumatic leaks was attributed to a failure of the associated diaphragm (EIIS: PC) and solenoid valve (EIIS: FSV) for each of the SRVs. When energized, the solenoid valve switches ports resulting in pneumatic pressure being applied to the SRV diaphragm. This results in the opening of the ADS SRV. The diaphragm and solenoid valve leakage only occurred when the particular SRV was actuated and therefore, the leakage would not have been detectable during normal plant operations. Potential underlying causes are being evaluated further in accordance with the Corrective Action Program. The solenoid valve leak was through the exhaust port seat. The diaphragms were found to show some signs of degradation (hard / brittle).

The 71C SRV, Serial Number (S/N) 83, would not properly re-close on the fourth actuation during laboratory testing. This ADS SRV actuated and re-seated properly on the first three laboratory tests involving actuation on over-pressure. The fourth actuation was performed using the pneumatic (manual) system. The valve failed to re-close after this actuation. The SRV is currently quarantined and the causal analysis will be completed in accordance with the Corrective Action Program based on finalized laboratory analysis.

The SRVs are manufactured by Target Rock Co. and are 'three-stage' relief valves. The solenoid valves were manufactured by the Automatic Valve Co. (Model No. C5450-5-110).

Analysis of the Event

There were no actual safety consequences associated with this event. There exists a total of 13 SRVs / Safety Valves (SVs) installed on the four Main Steam (EIIS: SB) Lines. The 11 installed SRVs exhaust steam through discharge lines to a point below the minimum water level in the Suppression Pool. The two installed SVs discharge steam directly to the Drywell. The SRVs and SVs are located on the four Main Steam Lines within Primary Containment (i.e. Drywell). The SRVs are 'three-stage' valves consisting of a main valve disc and piston (third stage) operated by a second stage disc and piston displaced by either a first stage pressure-sensing pilot (for overpressure protection) or a pneumatically-operated mechanical push rod (for the ADS function or for remote-manual operation).

The pneumatic leaks on the 71B and 71G SRVs and the failure of the 71C SRV to re-close following the fourth lift during laboratory testing would have had no impact on the ability of the SRVs / SVs to lift on an actual overpressure condition. The ASME Boiler and Pressure Vessel Code requires that the Reactor Pressure Vessel (EIIS: RCT) be protected from overpressure during upset conditions by self-actuated relief valves. As part of the nuclear pressure relief system, the size and number of SRVs and SVs are selected such that the peak pressure in the nuclear system will not exceed the ASME Code limits for the Reactor Coolant Pressure Boundary. During P2R16, the 70A SV was determined to have its as-found set points in excess of the Technical Specification allowable $\pm$ 1% tolerance (but well within the ASME Code allowable $\pm$3% tolerance). The required set point of the 70A SV is between 1247 psig and 1273 psig. The as-found set point was 1288 psig (1.2% higher than the Technical Specification allowable, but well within the ASME Code allowable set point). Since Technical Specification 3.4.3 only requires 11 of a total of 13 SRVs / SVs to be operable for over-pressure protection, the over-pressure protection function of the SRVs / SVs was not adversely affected by the conditions reported in this LER.

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

Analysis of the Event, continued

The ADS system consists of 5 of the 11 SRVs (71A, 71B, 71C, 71G and 71K). ADS is designed to provide depressurization of the reactor coolant system during a small break loss-of-coolant accident if the High Pressure Coolant Injection (HPCI) system fails or is unable to maintain required water level in the Reactor Pressure Vessel (RPV). ADS operation reduces the RPV pressure to within the operating pressure range of the low-pressure Emergency Core Cooling System (ECCS) subsystems so that the ECCS low-pressure subsystems can provide coolant inventory makeup.

In addition to the normal instrument nitrogen supply to each of the SRVs, the ADS SRVs are equipped with a nitrogen accumulator and associated inlet check valves. In the event that normal instrument nitrogen pneumatic supply is lost during a design basis event, the accumulator provides the pneumatic power to actuate the valves to mitigate design basis events. To meet the NUREG-0737 TMI Action Plan Requirements, the pneumatic accumulator supply is designed to hold a volume equivalent to five valve operations (at atmospheric pressure) following failure of the pneumatic supply to the accumulator. This capability is assured by a leak rate limit on the pneumatic system of less than 100 cc/min. As-found testing was satisfactory for the pneumatic systems for the 71A, 71C, and 71K ADS SRVs. Therefore, these ADS SRVs were fully capable of de-pressurizing the RPV for design basis events. The 71B and 71G ADS SRV diaphragms and solenoid valves would have only leaked when the associated ADS SRV was actuated in either the ADS or remote-manual mode. In the unlikely event that a design event occurred involving loss of the normal pneumatic supply and involved the need for ADS actuation, it is expected that the 71B and 71G ADS SRVs would have been able to be actuated initially until leakage would have resulted in loss of pneumatic actuation capability. Because of significant margin in design basis events analyses and the availability of other SRVs, it is not expected that this condition would have resulted in any significant threats to nuclear safety.

Concerning the ADS SRV 71C (S/N 83) failure to re-close during laboratory testing, it was determined that there were no actual safety consequences associated with this condition. During Cycle 16 operations, the 71C SRV operated and properly re-seated during a plant scram involving a main turbine trip on 7/10/05 (LER 2-05-01). If the SRV had been challenged by another plant transient during Cycle 16 operations and the valve did not re-close, the event would be bounded by the design basis event entitled, 'Inadvertent Opening of a Relief or Safety Valve.'

The conditions reported in this LER are not considered risk significant.

Corrective Actions

The diaphragms and solenoid valves associated with the 71B and 71G ADS SRVs were replaced. As-left leak testing and SRV functional testing involving the solenoid valves and diaphragms was performed. The valves were restored to an operable condition prior to plant startup from the P2R16 Refueling Outage. Further corrective actions involving ADS SRV diaphragms and solenoid valves are being considered in accordance with the Corrective Action Program.

Corrective Actions, continued

In accordance with preventive maintenance program, a refurbished SRV was installed in the 71C SRV location to replace the S/N 83 SRV. Further corrective actions in addition to those involving events reported previously (LER 2-03-04, LER 3-05-04) will be evaluated in accordance with the Corrective Action Program.

Previous Similar Occurrences

There was one previous LER identified involving the failure of an SRV to re-close following automatic operation. LER 2-03-04 identified an event on Unit 3 that involved a failure of SRV S/N 18 to re-close in September 2003. A laboratory failure analysis determined that tightly adhered foreign material on the pilot valve disc may have prevented the first stage pilot valve disc from properly re-closing. LER 2-03-04 also identified a condition involving a failure of a Unit 3 SRV to manually open as a result of a degraded actuator diaphragm. Corrective actions identified through the Corrective Action Program as a result of LER 2-03-04 included the improved decontamination, cleaning, refurbishment and examination processes. LER 3-05-04 also identified a condition discovered during laboratory testing in September 2005 where SRV S/N 193 failed to re-close after testing. The cause of the failure to re-close was due to vendor human performance issues during SRV maintenance. Because SRV S/N 83 was installed in the 71C SRV location in September 2002, previous corrective actions would not be expected to have prevented the SRV S/N 83 failure to re-close concern.