Exelon Nuclear
Peach Bottom Atomic Power Station
1848 Lay Road
Delta, PA 17314-9032

Telephone 717.456.7014
www.exeloncorp.com

Exelon Nuclear

10CFR 50.73

October 28, 2005

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-56
NRC Docket No. 50-278

Subject: Licensee Event Report (LER) 3-05-03

This LER reports a leak in the Reactor Coolant System identified during the 15th Refueling Outage for Unit 3. 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/CR 375299

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 05-14099

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)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Peach Bottom Atomic Power Station, Unit 3	05000 278	1 OF 5

4. TITLE
Residual Heat Removal System Small Bore Piping Leak due to Weld Deficiency

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	20	2005	05	- 03 -	0	10	28	2005	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 3

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 3, James Mallon, 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

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)*

On 9/20/05 at approximately 0500 hours, Licensed Operations personnel determined that a small amount of Reactor Coolant System (RCS) pressure boundary leakage existed on Unit 3. This determination was made based on review of a Primary Containment inspection performed subsequent to a Reactor shutdown on 9/19/05 for a Refueling Outage. The RCS pressure boundary leakage was determined to exist on a one-inch equalizing line for the 'A' Residual Heat Removal injection testable check valve located within the normally inaccessible Primary Containment. The leak was at a pipe coupling socket weld. The leak was determined to exist during Unit 3 Cycle 15 operations. The peak Containment unidentified leakage rate totaled approximately 1 gpm during the operating cycle. The cause of the weld deficiency was due to lack of fusion associated with the weld. The welded joint was replaced and appropriately inspected and tested successfully during the Unit 3 Refueling Outage. An extent of condition evaluation was performed for additional inspections associated with similar welds on Unit 3. Other evaluations will be performed for Unit 2 similar welds. There were no actual safety consequences associated with this event.

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

Unit Conditions Prior to Discovery of the Event

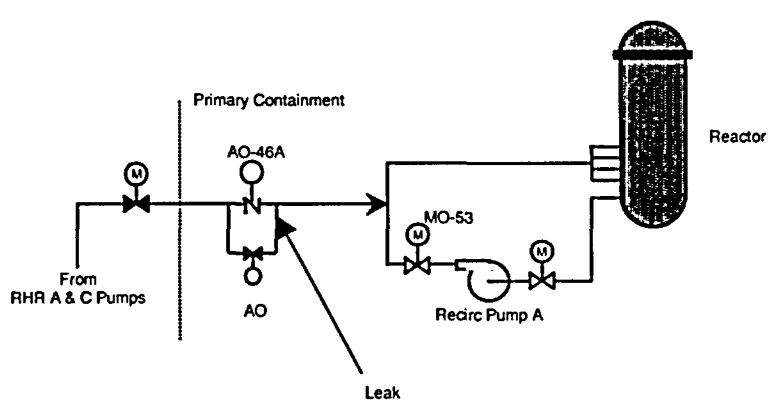
Unit 3 was in Mode 3 when the event was discovered on 9/20/05. As planned, Unit 3 had been manually scrammed on 9/19/05 at approximately 2141 hours from approximately 11% power in preparation for the 15th Refueling Outage for Unit 3. There were no structures, systems or components out of service that contributed to this event.

Description of the Event

On 9/20/05 at approximately 0500 hours, Licensed Operations personnel determined that a small amount of Reactor Coolant System (RCS) pressure boundary leakage existed on Unit 3. This determination was made based on review of a Primary Containment inspection performed subsequent to a Reactor shutdown on 9/19/05 at approximately 2141 hours for a Refueling Outage. The initial Primary Containment entry had commenced at approximately 2315 hours.

The RCS pressure boundary leakage (see below diagram) was determined to exist on a one-inch equalizing line for the 'A' Residual Heat Removal (RHR)(EIIS:BO) Loop testable air-operated check valve (AO-46A) (EIIS: V). The AO-46A valve is located on the 'A' RHR loop injection line. This line is located within the normally inaccessible Primary Containment and injects into the 'A' Recirculation (Recirc) line on the downstream side of the A Recirc line discharge motor-operated valve (MO-53A). The leak was located on a pipe coupling (EIIS: PSF) socket weld. This portion of the equalizing line is connected to the AO-46A check valve body on the downstream (Reactor) side of the check valve. The leaking pipe coupling is ASME Class 1 piping and therefore, is part of the RCS pressure boundary.

Simplified Diagram of 'A' RHR Loop



Description of the Event (continued)

Once the ASME Class 1 pressure boundary leakage was determined to exist by Licensed Operations personnel at 0500 hours on 9/20/05, the 'A' Low Pressure Coolant Injection (LPCI) subsystem and the 'A' Shutdown Cooling (SDC) subsystem were promptly declared inoperable. Both the LPCI and SDC systems are sub-modes of the overall RHR system. The 'B' SDC subsystem was placed in service by 1049 hours on 9/20/05. The Unit 3 Reactor entered Mode 4 (Cold Shutdown) by 1058 hours on 9/20/05.

NRC prompt notifications were completed pursuant to 10CFR 50.72(b)(3)(ii)(A) on 9/20/05 at approximately 1248 hours to report the RCS operational leakage (Event Notification #41832). This report is being submitted pursuant to:

1. 10CFR 50.73(a)(2)(i)(B) - Conditions Prohibited by Technical Specifications - Technical Specification Limiting Condition for Operation (LCO) 3.4.4.a requires that no RCS pressure boundary leakage exists. The leak existed during the applicability of the LCO (i.e. Modes 1, 2, and 3) during Unit 3 Cycle 15 operations. Also, as a result of the ASME Class 1 boundary leakage, the 'A' LPCI subsystem was declared inoperable. Since the leak developed during Unit 3 Cycle 15 operations, a condition prohibited by Technical Specification LCO 3.5.1 Condition A existed since the 'A' LPCI subsystem was inoperable for greater than 7 days.
2. 10CFR 50.73(a)(2)(ii)(A) - Degradation of the RCS – Because of the RCS leak, one of the principal safety barriers of the plant (i.e. the RCS) was degraded.

Analysis of the Event

There were no actual safety consequences associated with this event.

The discovered leakage rate was determined to be small. The peak Containment unidentified leakage rate totaled approximately 1 gpm during the operating cycle. Based on a review of Containment unidentified leakage data during the 15th Unit 3 operating cycle, it appears that the leak may have developed in the 4th quarter of 2003. At this time, licensed Operations personnel identified a step increase in unidentified leakage, although well below any Technical Specification action thresholds. No plant shutdowns occurred on Unit 3 since September 2003. The leakage was closely trended throughout the operating cycle. This leak was well below the 5 gpm unidentified leakage limit during Unit 3 Cycle 15 operations. The allowable RCS operational leakage limits are based on the predicted and experimentally observed behavior of pipe cracks. The evidence from experiments supports that, for leakage even greater than the specified unidentified leakage limits, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. The total leak rate increase since the 4th quarter 2003 was small.

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

Analysis of the Event (continued)

If a break of the one-inch line would have occurred, the leakage would be well within the makeup capabilities of the Emergency Core Cooling System (ECCS) network. The ECCS network is designed to provide effective core cooling regardless of the size or location of the piping break. The leakage from the one-inch pipe break during Reactor operations is well within the makeup capability of the High Pressure Coolant Injection (HPCI)(EIIS: BJ) system (part of the ECCS network). Other available ECCS systems include the Core Spray System (EIIS: BM), LPCI System (EIIS: BO) and Automatic Depressurization System (EIIS: RV). These systems were also capable of providing core cooling. In addition, the worst case leak was also well within the capability of the normal Feedwater system (EIIS: SJ) and the Reactor Core Isolation Cooling system (EIIS: BN).

This event is bounded by the Updated Final safety Analysis Report design event entitled 'Pipe Breaks Inside Primary Containment'.

The RHR system is comprised of 4 pumps, 4 heat exchangers and associated piping and valves. The modes of RHR (depending on valve lineups) include LPCI, SDC, Suppression Pool Cooling, Suppression Pool Spray and Drywell Spray. As a result of the leak, there was no impact on the Suppression Pool Cooling, Suppression Pool Spray or Drywell Spray modes of RHR since they use different flow paths than LPCI and SDC.

In accordance with site procedures, the 'A' LPCI subsystem was considered inoperable as a result of the ASME Class 1 boundary leak. NRC Inspection Manual Part 9900, Technical Guidance (9/26/05) states that the leaking component must be declared inoperable.

The SDC mode of RHR is not required to be operable per Technical Specifications until the Reactor is in Mode 3 with Reactor steam dome pressure less than the RHR SDC isolation pressure (70 psig). This plant condition did not exist between the time period of the 4th quarter of 2003 and 9/20/05. The 'B' and 'D' SDC subsystems were operable on 9/20/05.

Based on the above analyses, this event is not considered as risk significant.

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

Cause of the Event

The RCS leakage was due to a socket weld crack in the one-inch pipe coupling associated with the 'A' RHR loop testable check valve equalizing line. The crack extended approximately 160 degrees around the circumference of the weld. The cause of the weld failure was due to lack of fusion (i.e. a weld flaw) associated with the root weld. The lack of fusion was the crack initiator. The lack of fusion extended approximately 120 degrees around the weld. Laboratory analysis indicated that the crack originated from the weld interior at the weld root. The lack of fusion at the root of this weld reduced the strength of the weld. This weld was completed as part of a Recirc / RHR pipe replacement project performed on Unit 3 in the 1987-1989 time period.

Further cause analysis of the failed weld is being performed in accordance with the Corrective Action Program.

Corrective Actions

The welded joint was replaced and appropriately inspected and tested successfully during the Unit 3 Refueling Outage. The inspections included Non-Destructive Examinations (NDE).

An extent of condition evaluation was performed for additional inspections associated with similar welds on Unit 3. The evaluation identified other welds most susceptible to the failure identified. This sample size was beyond that which is required by the ASME code, but was considered prudent. Therefore, ultrasonic testing of similar small-bore welds was performed during the Unit 3 Refueling Outage. Additional indications were also identified on similar socket welds on the 'A' and 'B' loops of RHR. Repairs were completed for these indications and inspected / tested successfully during the Unit 3 Refueling Outage.

Further extent of condition reviews (including Unit 2) are being performed in accordance with the Corrective Action Program. This includes the consideration of enhanced programmatic strategies to identify small bore piping degraded welds.

Previous Similar Occurrences

There were no previous LERs identified involving RCS leakage involving similar one-inch piping.

Exelon Nuclear
Peach Bottom Atomic Power Station
1848 Lay Road
Delta, PA 17314-9032

Telephone 717.456.7014
www.exeloncorp.com

Exelon Nuclear

10 CFR 55.33

November 1, 2005

Mr. Samuel J. Collins
Regional Administrator
U. S. Nuclear Regulatory Commission Region I
475 Allendale Rd.
King of Prussia, PA 19406-1415

Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3
Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Request for Approval for Generic Fundamentals Examination (GFE) Administration

The purpose of this letter is to request approval to have the 14 individuals listed on Attachment 1 take the BWR Generic Fundamentals Examination section of the written operator licensing examination to be administered on 12/7/05. All of the listed individuals on Attachment 1 are enrolled in the PBAPS operator licensing training program and will have completed the generic fundamentals portion of this program by the examination date.

The following personnel will have access to the examinations before they are administered:

Name	Title
Peter Gigliotti	Operations Training - Exam Proctor
Richard Edens	Operations Training - Exam Coordinator / Reviewer
Bruce Hennigan	Operations Training - ILT Principle Instructor
Jack Popielarski	Operations Training - Manager

Please address the examinations to the overnight mail address below:

Bruce Hennigan, Operations Training – ILT Principal Instructor
Mail Stop: PBTC
Peach Bottom Atomic Power Station
1848 Lay Rd.
Delta, PA 17314

If you have any questions, feel free to contact Mr. Jack Popielarski at 717-456-3822.

Sincerely,

Robert C. Braun
Site Vice President
Peach Bottom Atomic Power Station

cc: F. L. Bower, Senior Resident Inspector, USNRC, PBAPS
R. R. Janati, Commonwealth of Pennsylvania
Document Control Desk, USNRC, Washington DC
R. J. Conte, USNRC Region I Chief, Operations Branch, DRS

CCN: 05-14102
Attachment