Exelon Generation.

10CFR 50.73

December 20, 2017

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

Peach Bottom Atomic Power Station (PBAPS) Unit 3
Renewed Facility Operating License No. DPR-56
NRC Docket No. 50-278

Subject: Licensee Event Report (LER) 3-17-001

Enclosed is a Licensee Event Report concerning a condition prohibited by Technical Specifications due to a reactor pressure boundary leakage from a weld. In accordance with NEI 99-04, the regulatory commitment contained in this correspondence is to restore compliance with the regulations. The specific methods that have been 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 Jim Kovalchick at 717-456-3351.

Sincerely,

Pat D. N

Patrick D. Navin
Site Vice President
Peach Bottom Atomic Power Station

PDN/dnd/IR 4065691

Enclosure

cc: US NRC, Administrator, Region I
US NRC, Senior Resident Inspector
R. R. Janati, Commonwealth of Pennsylvania
S. Gray, State of Maryland
B. Watkins, PSE&G, Financial Controls and Co-owner Affairs

CCN: 17-106

NRC FORM 366 (04-2017) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Peach Bottom Atomic Power Station Unit 3	05000278	1 OF 3

4. TITLE

Reactor Pressure Boundary Leakage Due to Weld Failure in One-Inch Diameter Instrument Line

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
10	23	2017	2017	- 001	- 00	12	21	2017	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
3	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☒ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
000	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A	

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
James M. Kovalchick, 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
B	AD	PSF	N/A	N					

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 10/23/2017, during a walkdown of containment at the start of the Unit 3 refueling outage, a leak was identified in a socket weld for a 1-inch diameter instrument line. The line is connected to discharge piping for the 'B' recirculation pump and is part of the reactor coolant system pressure boundary. Because the leak was misting, the leakage rate could not be quantified. However, the reactor coolant system unidentified leakage prior to plant shutdown was 0.18 gpm. RCS pressure boundary leakage while in Mode 1 is a violation of Technical Specification 3.4.4 and is a reportable condition.

The cause of the event was a lack of fusion defect in the weld when it was done in the late 1980's. Normal vibration of the line since it was installed resulted in the crack initiating at the weld defect and propagating to the surface. The section of pipe and associated fitting were replaced, along with welds in similar sections of piping. There were no actual safety consequences as a result of this event.

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NARRATIVE

Unit Conditions Prior to Discovery of the Event

Unit 3 was in Mode 3, Hot Shutdown, when the condition was discovered on 10/23/2017 at approximately 4:00 am. The unit entered Mode 3 at 1:08 am in preparation for the planned refueling outage. There were no structures, systems, or components out of service that contributed to this event.

Description of the Event

On 10/23/2017, with the Unit in Mode 3, at the beginning of a refueling outage, personnel entered the drywell to perform an inspection. At approximately 4:00 am, leakage was identified on a one-inch diameter instrument line socket (EIIS:PSF) weld. The instrument line is for a pressure transmitter from the 'B' Recirculation Pump (EIIS:AD) discharge. Because the leak was misting, the leakage rate could not be quantified. However, the reactor coolant system unidentified leakage prior to plant shutdown was 0.18 gpm.

This line is part of the ASME Class 1 primary coolant pressure boundary. Technical Specification (TS) 3.4.4 does not allow any pressure boundary leakage in Modes 1, 2 or 3. As a result, TS 3.4.4 Condition C was entered, which requires the plant to be in Mode 3 in 12 hours and Mode 4 in 36 hours. The plant was in Mode 3 at the time of discovery and entered Mode 4, Cold Shutdown, at 12:54 pm on 10/23/2017 (approximately 9 hours after time of discovery).

Prompt notification to the NRC was made in accordance with 10 CFR 50.72(b)(3)(ii)(A) on 10/23/2017 at 9:35 am (Event Notification #53031).

Analysis of the Event

The leak was determined to be from a crack in the weld connecting the pipe to a tee fitting. The crack extended approximately 60 degrees around the circumference of the weld. Although it could not be determined when the crack developed, it is assumed it existed while operating in Mode 1, prior to shutdown. During such time, the leakage would have been contributing to the total unidentified leakage of 0.18 gpm. This is well below the 5 gpm limit for unidentified leakage as stated in TS 3.4.4.

This event is being reported in accordance with the following:

1. 10 CFR 50.73(a)(2)(i)(B) – Conditions Prohibited by Technical Specifications - TS Limiting Condition for Operation (LCO) 3.4.4 requires there to be no RCS pressure boundary leakage while in Modes 1, 2 and 3. If leakage exists, TS 3.4.4 Condition C requires the unit to be in Mode 3 in 12 hours and Mode 4 in 36 hours. Since the leak existed while in Mode 1 for greater than 36 hours, a condition prohibited by TS existed.

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NARRATIVE

2. 10 CFR 50.73(a)(2)(ii)(A) – Degradation of the RCS – Because of the RCS pressure boundary leakage, one of the principal safety barriers of the plant was degraded.

Cause of the Event

The section of the one-inch pipe containing the cracked socket weld was sent to an offsite laboratory for analysis. The pipe was sectioned and lack of fusion defects were identified at the root of the weld. The crack initiated at the location of these defects. The defects would have occurred when the weld was performed during a recirculation system pipe replacement in the late 1980's. It is unknown when the crack began, but normal vibration for the piping likely caused it to propagate to the surface of the weld, resulting in the identified leak.

Corrective Actions

The section of pipe and the associated fitting were replaced. Instrument lines connected to the suction and discharge lines for both of the recirculation pumps with similar configuration and subject to vibration were also replaced during the refueling outage. The new welds were performed with a 2:1 profile, which reduces their susceptibility to vibration-induced failures.

Previous Similar Occurrences

A similar event occurred in September of 2005. A crack in a socket weld caused by a lack of fusion defect resulted in a 1 gpm leak in a one-inch equalizing line for a check valve on the 'A' Residual Heat Removal (RHR) injection line. The event is documented in LER 2005-003, dated 10/28/2005. Additional information on this previous occurrence is contained in the corrective action program.