Jon A. Franke
Site Vice President

Susquehanna Nuclear, LLC
769 Salem Boulevard
Berwick, PA 18603
Tel. 570.542.2904 Fax 570.542.1504
Jon.Franke@talenenergy.com

TALEN ENERGY

JUN 1 0 2015

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

10 CFR 50.73

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 50-388/2015-004-00
UNIT 2 LICENSE NO. NPF-22
PLA-7337

Docket No. 50-388

Attached is Licensee Event Report (LER) 50-388/2015-004-00. The LER reports an event involving a degraded condition due to Reactor Coolant Pressure Boundary leakage in accordance with 10 CFR 50.73(a)(2)(ii)(A). This event is also being reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by Technical Specifications.

There were no actual consequences to the health and safety of the public as a result of this event.

This letter contains no new regulatory commitments.

for J. A. Franke

J. A. Franke

Attachment: LER 388/2015-004-00

Copy: NRC Region I
Mr. J. E. Greives, NRC Sr. Resident Inspector
Mr. J. A. Whited, NRC Project Manager
Mr. B. R. Fuller, PA DEP/BRP

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@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.

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4. TITLE Degraded Condition Due to Reactor Coolant Pressure Boundary Leakage Caused by Vibration and Stiff Pipe Connection

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
04	11	2015	2015	- 004	00	06	10	2015	FACILITY NAME	DOCKET NUMBER 05000

9. OEPRATING 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)(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)
10. POWER LEVEL	☐ 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)
000	☐ 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

LICENSEE CONTACT	TELEPHONE NUMER *(Include Area Code)*
C. E. Manges, Jr., Senior Engineer – Nuclear Regulatory Affairs	570 542 3089

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	P	FLOWSERVE	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)*

On April 11, 2015, during the initial drywell walk down following shutdown for a refueling outage, a leak was identified on the Unit 2 "A" Reactor Recirculation Pump seal piping at a seal flange weld associated with the pressure and vent piping to the upper seal chamber (Connection #2). Water was seen spraying out of the top of the pipe in a fan pattern.

This event was reported under 10 CFR 50.72(b)(3)(ii)(A) per the guidance of NUREG 1022, Revision 3, Section 3.2.4 as a degraded condition (EN 50976). This event is also being reported as a Licensee Event Report (LER) in accordance with 10 CFR 50.73(a)(2)(ii)(A) and 10 CFR 50.73(a)(2)(i)(B).

The direct cause of the weld crack was determined to be high cycle, low amplitude vibration fatigue. The apparent cause involved the combined effects of an unrecognized system vibrational mode and a stiff pipe connection to the seal flange. Causal factors included the presence of a stress concentrator (minor lack of fusion) on the root of the weld and two early life cavitation events in 1983. Piping configurations for the Unit 2 pumps were modified to add flexibility in the pipe run between the seal flange and first support and to minimize vibration modes coincident with known pump resonance frequencies 1X, 2X, and 5X. Piping configurations will be modified for Unit 1 during the next refueling outage (the affected connections on Unit 1 are not reactor coolant pressure boundaries).

There were no actual consequences to the health and safety of the public as a result of this event.

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NARRATIVE

CONDITIONS PRIOR TO EVENT

Unit 2 – Mode 3, 0 percent Rated Thermal Power

Other than the leaking weld itself, there were no structures, systems, or components that were inoperable at the start of the event that contributed to the event.

EVENT DESCRIPTION

On April 11, 2015, during the initial drywell walk down following shutdown for a refueling outage, a leak was identified on the Unit 2 "A" Reactor Recirculation (RXR) [EIIS System Identifier: AD] Pump [EIIS Component Identifier: P] seal piping at a seal flange weld associated with the pressure and vent piping to the upper seal chamber (Connection #2). Water was seen spraying out of the top of the pipe in a fan pattern. A timeline of relevant events follows:

In 1983, there were two cavitation events on the Unit 2 "A" RXR pump (2P401A).

In December 1999, an engineering change was implemented that added a pipe support to the pump motor stand, removed the spring can, added 2X1 welds, and replaced a 45 degree elbow with a bend.

In August 2000, an engineering change added a ¾" X 1" reducing coupling ten inches from the union, replaced the ¾" piping with 1" piping from the coupling to the flange, and modified the pipe support for a 1" pipe.

In May 2013, an engineering change was implemented to replace the 2P401A shaft and related components.

On April 10, 2015 at approximately 2347, the Unit 2 mode switch was placed in shutdown commencing the Unit 2 17th refueling and inspection outage (RIO).

On April 11, 2015 at approximately 0415, the initial drywell entry was commenced.

On April 11, 2015 at approximately 0555, the control room was notified of leakage from the 2A Reactor Recirculation Pump seal line.

On April 11, 2015 at approximately 0958, the control room was notified that the leakage was pressure boundary leakage.

This event was reported under 10 CFR 50.72(b)(3)(ii)(A) per the guidance of NUREG 1022, Revision 3, Section 3.2.4 as a degraded condition (EN 50976). This event is also being reported as a Licensee Event Report (LER) in accordance with 10 CFR 50.73(a)(2)(ii)(A) and 10 CFR 50.73(a)(2)(i)(B).

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NARRATIVE

CAUSE OF EVENT

The direct cause of the weld crack was determined to be high cycle, low amplitude vibration fatigue.

The apparent cause involved the combined effects of an unrecognized system vibrational mode and a stiff pipe connection to the seal flange. The vibrational mode was introduced in 2013 during a replacement of the motor, shaft, and seal that included piping changes to accommodate an elevation change. The stiff pipe configuration was introduced during piping and pipe support modifications made in 1999 and 2000.

Causal factors included the presence of a stress concentrator (minor lack of fusion) on the root of the weld and two early life cavitation events in 1983. The minor lack of fusion is a fairly common occurrence on this type of weld and does not, by itself, render the weld defective.

ANALYSIS/SAFETY SIGNIFICANCE

The small leak was a violation of the Unit 2 Technical Specification, Section (TS) 3.4.4, "Reactor Coolant System (RCS)." The RCS leakage shall be limited to no pressure boundary leakage.

The condition was identified after the unit was shutdown. During plant operation, unidentified drywell leakage was measured at approximately 0.25 gallons per minute (gpm). The TS allowable is 5 gpm. The weld failure was a significant contributor to the overall unidentified drywell leakage. There was no evidence available to plant operators at the time to substantiate that leakage was from the reactor coolant pressure boundary. The unit was shut down for a scheduled refuel and inspection outage. During containment walk-downs, the subject condition was identified and classified as pressure boundary leakage.

The potential consequence was a ¾ inch unisolable pipe leak from the Reactor Coolant Pressure Boundary.

Based on review of the Unit 2 TS Bases (Section 3.4.4):

- The allowable RCS operational leakage limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage were also considered. The evidence from experiments suggests 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 unidentified leakage flow limit allows time for corrective action before the Reactor Coolant Pressure Boundary (RCPB) could be significantly compromised. The five gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Crack behavior from experimental programs shows that leakage rates of hundreds of gallons per minute will precede crack instability

During the April 11, 2015 entry of the drywell, identified leakage from the 2A RXR Seal piping weld was described as a spray out of the top of the pipe in a fan pattern as opposed to a steady stream.

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NARRATIVE

CORRECTIVE ACTIONS

1. Unit 2: The piping configurations for Pipe Connection #2 and #6 on 2P401A and for Pipe Connection #1 and #2 on 2P401B were modified to add flexibility in the pipe run between the seal flange and first support and minimize vibration modes coincident with known pump resonance frequencies 1X, 2X, and 5X.
2. Unit 1: The stiffened connections on 1P401A and 1P401B (Connection #1 for each pump) are not reactor coolant pressure boundaries. The piping configurations for Pipe Connection #1 on 1P401A and on 1P401B will be modified to add flexibility in the pipe run between the seal flange and first support and minimize vibration modes coincident with known pump resonance frequencies 1X, 2X, and 5X.

COMPONENT FAILURE INFORMATION

The failed component was a seal flange weld associated with the pressure and vent piping to the upper seal chamber (Connection #2) for the Unit 2 RXR pump. The apparent cause involved the combined effects of an unrecognized system vibration and a stiff pipe connection to the seal flange.

PREVIOUS SIMILAR EVENTS

LER 50-387/2014-011-00: "Degraded Condition Due to Reactor Coolant Pressure Boundary Leakage Caused by Inadequate Weld," dated February 11, 2015. Although this was also reported as a degraded condition as a result of pressure boundary leakage, the cause was due an inadequate weld.

LER 50-387/2012-007-01: "Unplanned Shutdown Due to Elevated Drywell Unidentified Leakage," dated November 20, 2012.

Susquehanna identified eight condition reports (CRs) involving cracked welds on small bore piping. Seven out of the eight CRs identify vibration as the cause of the cracked weld, and one CR identified lack of weld fusion as the cause of the cracked weld. The seven CRs identifying socket weld failures occurred between 1992 and 2004. Susquehanna completed an initiative to mitigate the effects of system vibration on the fatigue life of socket welds inside containment. Volumetric inspections are performed on the socket welds considered to be most vulnerable to vibration-induced fatigue. The goal of the inspections is to ensure that the socket welds are of good quality with no latent cracks. If the weld is of good quality then the existing weld is built up with a full circumference EPRI 2x1 socket weld overlay. If the weld is not of good quality then the entire weld is removed and a full circumference EPRI 2x1 socket weld is installed. Previous corrective actions appear to be effective given that no non-vendor weld failures have occurred since the 2004 time frame.