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FEB 0 1 2012

U. S. Nuclear Regulatory Commission
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SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 50-387/2010-003-02
LICENSE NO. NPF-14
PLA-6805 **Docket No 50-387**

Attached is supplemental Licensee Event Report (LER) 50-387/2010-003-02. This supplement is being submitted because PPL Susquehanna, LLC (PPL) determined that the original investigation of this reportable event did not comprehensively address the organizational, programmatic, and safety culture contributors to the event and, as a result, established a root cause investigation team to supplement the original root cause evaluation. This supplement reflects the results of that additional evaluation.

The original LER 50-387/2010-003-00 was submitted to the Nuclear Regulatory Commission (NRC) on September 14, 2010. Supplement 1 to the LER (50-387/2010-003-01) was subsequently submitted to the NRC on March 1, 2011.

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

No commitments were identified in this submittal.

F. A. Kearney
Site Vice President

Attachment

Copy: NRC Region I
Mr. P. W. Finney, NRC Sr. Resident Inspector
Mr. R. R. Janati, DEP/BRP
Mr. B. K. Vaidya, NRC Project Manager

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013
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: 80 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 F53), 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.

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4. TITLE
Unit 1 Manual Reactor Scram due to Leakage from the Unit 1 Circulating Water System and Subsequent Flooding of the Unit 1 Condenser Bay

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
07	16	2010	2010	- 003 -	02	02	01	2012	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*			
1	☐ 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)
10. POWER LEVEL	☐ 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)
89%	☐ 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)
Brenda W. O'Rourke, Senior Engineer - Nuclear Regulatory Affairs	(570) 542-1791

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 July 16, 2010, at approximately 1641 EDT, the Susquehanna Steam Electric Station Unit 1 reactor was manually scrammed due to a large unisolable circulating water system leak in the main condenser area. All control rods fully inserted. Reactor water level lowered to -28 inches causing Level 3 (+13 inches) isolations. The Integrated Control System (ICS) Feedwater Level Control (FWLC) system detected the scram condition and automatically entered setpoint setdown mode, which placed the non-lead RFPs in idle mode, and initiated transition to startup level control mode. During this transition, ICS FWLC did not transfer to single element control due to a higher than expected steam flow signal, and concurrent feedwater flow oscillations resulted in an increase in reactor water level. Reactor water level reached Level 8 (+54 inches) which resulted in the trip of all three RFP turbines, the High Pressure Coolant Injection (HPCI) system, and the Reactor Core Isolation Cooling (RCIC) system to shutdown. The circulating water system was shut down and the main steam isolation valves were manually closed. Reactor water level was restored and maintained within normal operating range using the RCIC system. Pressure control was initiated using the HPCI system in the pressure control mode. All safety systems operated as expected. No steam relief valves opened. It was estimated that approximately one million gallons of water leaked into the condenser bay area.

The cause of the unisolable circulating water system leak was due to the condenser waterbox manway gasket rolling out of position. Investigation concluded that the gasket reached the point where it could no longer maintain system pressure and rolled out of position due to gasket creep (i.e., inadequate gasket preload to maintain joint integrity). The gasket extrusion was the result of inadequate preload, rather than a system pressure transient or a material defect. Corrective actions taken for Unit 1 included inspection and replacement of gaskets.

The root causes were determined to be less than optimal system manway and isolation valve design, less than adequate risk informed decision making resulted in the failure to adequately address previous Circulating Water (CW) system leaks, and inadequate procedure quality, use and adherence resulted in the loss of CW pressure boundary integrity and inadequate mitigation of the CW leak.. Planned actions to prevent recurrence included revising procedures to address gasket installation procedure deficiencies, revising procedures to address isolating individual waterboxes, and developing revisions to processes and procedures to improve risk informed decision making.

There were no adverse consequences to the health and safety of the public as a result of this event. This event is being reported under 10 CFR 50.73(a)(2)(iv)(A) due to the manual actuation of the RPS, RCIC and HPCI system.

NARRATIVE

CONDITION PRIOR TO THE EVENT

Unit 1 - Mode 1, 89 percent Rated Thermal Power due to ambient conditions

EVENT DESCRIPTION

On July 16, 2010, at approximately 1641 EDT, the Susquehanna Steam Electric Station (SSES) Unit 1 reactor was manually scrammed due to a large unisolable circulating water system leak in the main condenser area. During attempts to isolate the leak, the operators lowered reactor power from approximately 89 percent to about 39 percent. It was identified that the leak was coming from two separate condenser waterbox manway door seals. Due to delays in the identification of the leak's location and the failure of the condenser waterbox isolation valves to close electrically, the condenser bay continued to flood. Based on rising water level in the condenser area and the unsuccessful isolation of the source of the leakage, Operations manually scrammed Unit 1 by placing the mode switch in shutdown. All control rods fully inserted. Reactor water level lowered to -28 inches causing Level 3 (+13 inches) isolations.

The Integrated Control System (ICS) Feedwater Level Control (FWLC) system detected the scram condition and automatically entered setpoint setdown mode, which placed the non-lead RFPs in idle mode, and initiated transition to startup level control mode. During this transition, ICS FWLC did not transfer to single element control due to a higher than expected steam flow signal, and concurrent feedwater flow oscillations resulted in an increase in reactor water level. Reactor water level reached Level 8 (+54 inches) which resulted in the trip of all three RFP turbines, the High Pressure Coolant Injection (HPCI) system, and the Reactor Core Isolation Cooling (RCIC) system to shutdown.

The Circulating Water (CW) System was shut down and the main steam isolation valves (MSIVs) were manually closed. Reactor water level was restored and maintained within normal operating range using the RCIC system. Pressure control was initiated using the HPCI system in the pressure control mode. All safety systems operated as expected. No steam relief valves opened. It was estimated that approximately one million gallons of water leaked into the condenser bay area.

An ENS notification (# 46103) was made to the NRC in accordance with 10 CFR 50.72(b)(2)(iv)(B) for an event or condition that resulted in the actuation of the RPS when the reactor was critical, and 10 CFR 50.72(b)(3)(iv)(A) due to the manual actuation of the RPS, RCIC and HPCI system.

CAUSE OF THE EVENT

PPL completed a root cause investigation shortly after the event occurred. The root cause identified that the circulating water system leak occurred due to the condenser waterbox manway gasket rolling out of position. The neoprene rubber gaskets tend to "creep" after installation, resulting in the torque on the gasket hold-down bolts to drop below the required torque values as defined in the installation procedure. The root cause determined that the lower preload torque resulting from the gasket "creep" would have been detected and the failure likely precluded had the installation procedure required a re-check of the manway hold-down bolt torque. The procedure does require a leak check to be performed at system operating conditions, but no requirement to re-check the torque values is required by the procedure. This represents a missed opportunity to identify and evaluate previous CW system leaks by the corrective action program.

Other contributing causes of the failure include use a bolt torque value not in accordance with current manufacturer recommendations, irregularities in the manway seating area, and less than optimal manway cover design.

Investigation concluded that the gasket reached the point where it could no longer maintain system pressure and rolled out of position due to gasket "creep" (i.e., inadequate gasket preload to maintain joint integrity). The gasket extrusion was the result of inadequate preload, rather than a system pressure transient or a material defect.

During the flooding, Operations was not able to positively determine which waterbox was leaking due to limited access to the condenser area and the inability to identify equipment due to the water spray from the leak. This difficulty was compounded by the lack of procedure direction supporting operator response to CW system leaks. It is possible that if the correct waterbox was initially selected and isolated, the total leakage into the condenser bay would have been significantly reduced.

PPL subsequently determined that the original investigation did not comprehensively address the organizational, programmatic, and safety culture contributors to the event and established a root cause investigation team to supplement the original root cause evaluation. The root causes identified by the supplemental root cause evaluation include:

- Less than adequate risk informed decision making resulted in the failure to adequately address previous CW system leak events.
- Less than optimal design of CW system equipment (condenser waterbox manway and waterbox inlet/outlet valves' motors compartments) resulted in the failure to maintain the CW system pressure boundary integrity and impacted the flooding mitigation capability.
- Less than adequate procedure quality, use and adherence to maintenance procedures for main condenser leak and CW system leak detection, and operations procedure for immersed heat exchanger internals epoxy lining, resulted in the loss of the CW pressure boundary integrity and inadequate mitigation of the CW leak.

ANALYSIS / SAFETY SIGNIFICANCE

Circulating Water System

A circulating water system rupture is an anticipated event as discussed in Section 10.4.1.3.3 in the SSES Final Safety Analysis Report (FSAR). The presence of any water accumulation in the condenser bay is detected by level switches which are mounted on the shielding wall at various points around its perimeter. These switches did alarm in the control room during the flooding event. The condenser bay is designed to contain the water from a circulating water system rupture. The concrete shielding walls which surround the condenser bay are designed to withstand the possible 20 feet of differential water pressure they could experience in the event of a major rupture. Water level reached approximately 12 feet in the condenser bay during the event, which is well within the design flood level.

The 656 foot elevation doors, which provide access through the shielding walls, are pressure resistant. While not watertight, these doors do restrict water leakage out of the condenser bay in the event of flooding. Also, while penetrations through the walls are not watertight, they are filled with a sealant for radiation shine which serves to limit the quantity of water leaking out of the condenser bay in the event that it becomes flooded.

There is no safety-related equipment in the Turbine Building below grade (676 foot elevation), and the penetrations below grade, between the Turbine building and the Reactor Building, are designed to prevent flooding of the Reactor Building. There was no significant water accumulation in the turbine building, outside of the condenser bay, during this event – the flooding was effectively contained within the condenser bay.

Neither a major rupture of the CW system nor a rupture of the condenser hotwell will have an effect on any safety-related system since no safety-related systems are located within this area.

ICS Feedwater Level Control System

The ICS FWLC logic is designed to automatically perform a series of actions following a reactor scram in order to prevent overfeeding the reactor vessel and transition to startup level control mode. The ICS FWLC system is a cascade level controller that implements level control by controlling flow in a three element controller. The ICS FWLC system is designed to detect a reactor scram condition based on a reactor vessel level of less than +13 inches and a total main steam flow of less than 0.7 million pounds mass per hour. Upon detecting a scram condition, the ICS FWLC is designed to automatically initiate setpoint setdown mode, which sets the reactor vessel level setpoint to +18 inches, places the non-lead reactor feed pumps in idle mode and initiates transition of the FWLC system to startup level control mode.

Potential Consequences

The SSES Plant Analysis group prepared a risk assessment for the plant shutdown initiating event. This risk assessment takes into consideration the plant conditions immediately leading to shutdown, the equipment out of service for maintenance at the time of shutdown, and the conditions at the time of shutdown. Flooding in the condenser bay is not considered an initiating event in the SSES Probabilistic Risk Assessment model. It is assumed in flood calculations that a flood in the condenser bay will eventually result in a plant shutdown due to loss of the main condenser. It is anticipated that the shutdown would occur by either an automatic or manual reactor scram. During the event, a manual scram was executed from approximately 39 percent power. The circulating water system was shutdown and the MSIV's were isolated. Equipment in the area that may have been out of service as a result of the flooding was non-safety related power generation equipment and therefore was not required for safe shutdown of the unit.

Actual Consequences

The leak from the CW system resulted in an internal flooding event, and a subsequent manual scram, isolation of the reactor and loss of the condenser heat sink for Susquehanna Unit 1. Although actual plant safety was not compromised, Unit 1 was unnecessarily placed in a higher level of risk as a result of this event. With the manual scram, Unit 1 was safely shut down. A reactor manual scram is consistent with expected operator actions for flooding events of this magnitude. The MSIVs were manually closed and CW was isolated. HPCI tripped and RCIC shut down by design when high vessel water level was received immediately following the scram. Both systems were subsequently reset. HPCI was used for pressure control while RCIC was used for level control. No mitigating systems were out of service during this event. While the reactor vessel level reaching +54 inches following reactor scram was not expected for the ICS FWLC system, the system's basic protection function to prevent overfeeding of the reactor vessel worked as designed. Additionally, although safety systems were required to function as a result of the event, they operated per design, and there were no safety impacts to safety-related equipment due to the flooding nor was Unit 2 impacted as a result of the Unit 1 event. There was no impact to the health and safety of the public.

CORRECTIVE ACTIONS

Completed Actions:

The following actions have been completed on Unit 1 and Unit 2:

- All manway gaskets were inspected and replaced as required.
- The epoxy coating on the manway at the gasket seating surface has been re-worked to provide a smooth and uniform surface where the gasket fits over the end of the manway.
- The manway hatch epoxy coated surfaces have been roughened to increase friction between the hatch and the gasket contact surface.
- An installation check has been incorporated into the work packages to ensure proper seating of the gasket prior to installing the hatch.

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- The manway hatch bolt torque for new installation has been increased from 60 ft-lbs to 110 ft-lbs. Testing demonstrated that 110 ft-lbs torque provides the desired 50% crush on the gasket as per vendor recommendation.
- Bolt torquing was performed in a star pattern at 25% increments, let idle for a minimum of 2 hours, and then torque checked at 110 ft-lbs to address any gasket creep.
- A design change was implemented to the ICS FWLC system logic to automatically transfer FWLC to single element upon initiation of setpoint setdown mode following a scram.
- Revised procedures to address gasket installation procedure deficiencies.
- Enhanced the inspection procedure to ensure compliance with Manway coating specification.
- Revised procedures to address isolating individual waterboxes for leak between waterbox inlet and outlet valves and add diagrams to aid in leak isolation if required.
- Developed revisions to processes and procedures to improve risk informed decision making.
- Established a written policy for risk-informed decision making using a consistent systematic methodology.

Key Planned Corrective Actions

- Design and implement a modification of the Unit 1 and Unit 2 condenser manway joint connections to eliminate flooding potential of these joints.
- Design and implement a modification of the Unit 1 and Unit 2 main condenser waterbox inlet and outlet valves motor compartments to establish reliability for valve operation during main condenser manway cover leak events.
- Identify list of components that need labeling and provide labels (such as condensers, piping areas/access doors) to improve condenser leak identification capability.