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SEP 1 4 2010

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop OP1-17
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 50-387/2010-003-00
LICENSE NO. NPF-14
PLA-6648

Docket No 50-387

Attached is Licensee Event Report (LER) 50-387/2010-003-00. 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. This event was determined to be reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A) for an event that resulted in the manual actuation of the Reactor Protection System, Reactor Core Isolation Cooling and the High Pressure Coolant Injection system.

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

No commitments were identified in this submittal.

T. S. Rausch
Senior Vice President - Chief Nuclear Officer

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 (9-2007)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Susquehanna Steam Electric Station Unit 1	05000387	1 OF 4

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 -	00	09	14	2010	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 89%

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)
Jason Jennings, Senior Engineer - Nuclear Regulatory Affairs	(570) 542-3155

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 (SSES) 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. Reactor water level was restored and maintained within normal operating range using the Reactor Core Isolation Cooling (RCIC) system. No steam relief valves opened. The main steam isolation valves (MSIVs) were manually closed and the circulating water system was shut down. Pressure control was initiated using the High Pressure Coolant Injection (HPCI) system in the pressure control mode. All safety systems operated as expected. 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 an inadequate manway gasket installation process and inadequate control of information in controlled procedures. Planned actions to prevent recurrence include revising procedures to address gasket installation procedure deficiencies and revising procedures to address isolating individual waterboxes. An extent of condition inspection will be performed on Unit 2 during the next planned downpower.

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.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Susquehanna Steam Electric Station Unit 1	05000387	2010	--003--	00	2 OF 4

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. Reactor water level was restored and maintained within normal operating range using the Reactor Core Isolation Cooling (RCIC) system. No steam relief valves opened. The main steam isolation valves (MSIVs) were manually closed and the circulating water system was shut down. Pressure control was initiated using the High Pressure Coolant Injection (HPCI) system in the pressure control mode. All safety systems operated as expected. 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

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.

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 procedural deficiencies. It is likely that if the correct waterbox was initially selected to isolate the leak, the leak would have been effectively isolated with minimal flooding in the condenser bay.

ANALYSIS / SAFETY SIGNIFICANCE

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 circulating water 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.

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

There was no impact on plant safety as a result of this event. The reactor was manually scrammed and the unit safely shutdown. There was no impact on Unit 2 as a result of these actions. A manual reactor scram is consistent with expected operator actions for flooding events of this magnitude. The MSIVs were manually closed and circulating water was isolated. HPCI was used for pressure control, RCIC for level control. HPCI tripped and RCIC shut down by design when high vessel water level was received. No mitigating systems were out of service during this event.

Based upon the above discussion, the actual consequences of this event were minimal. 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:

- 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.
- 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.

Planned Corrective Actions

- Revise procedures to address gasket installation procedure deficiencies.
- Enhance the inspection procedure to ensure compliance with Manway coating specification.
- Revise procedures to address isolating individual waterboxes for leak between waterbox inlet and outlet valves and add diagrams to aid in leak isolation if required.
- Perform an extent of condition inspection on Unit 2 during next planned downpower.