

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) SUSQUEHANNA STEAM ELECTRIC STATION - UNIT ONE DOCKET NUMBER (2) 0 5 0 0 0 3 8 7 1 OF 0 2 PAGE (3)

TITLE (4) VALVE PACKING LEAK LEADS TO REACTOR SCRAM SIGNAL

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)		
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES		DOCKET NUMBER(S)
04	12	86	86	01	6	00	05	12	86		05000
											05000

OPERATING MODE (8) 4

POWER LEVEL (10) 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more of the following) (11)

<input type="checkbox"/> 20.402(b)	<input checked="" type="checkbox"/> 20.408(a)	<input type="checkbox"/> 50.73(a)(2)(iv)	<input type="checkbox"/> 73.71(b)
<input type="checkbox"/> 20.408(a)(1)(i)	<input type="checkbox"/> 50.38(a)(1)	<input type="checkbox"/> 50.73(a)(2)(v)	<input type="checkbox"/> 73.71(a)
<input type="checkbox"/> 20.408(a)(1)(ii)	<input type="checkbox"/> 50.38(a)(2)	<input type="checkbox"/> 50.73(a)(2)(vi)	OTHER (Specify in Abstract Draw and in Text, NRC Form 366A)
<input type="checkbox"/> 20.408(a)(1)(iii)	<input type="checkbox"/> 50.73(a)(2)(i)	<input type="checkbox"/> 50.73(a)(2)(vii)(A)	
<input type="checkbox"/> 20.408(a)(1)(iv)	<input type="checkbox"/> 50.73(a)(2)(ii)	<input type="checkbox"/> 50.73(a)(2)(vii)(B)	
<input type="checkbox"/> 20.408(a)(1)(v)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)	

LICENSEE CONTACT FOR THIS LER (12)

NAME Jeffrey A. Hirt, Engineering Level 1 TELEPHONE NUMBER 717 542-3917

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPROS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPROS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (if you complete EXPECTED SUBMISSION DATE) NO

EXPECTED SUBMISSION DATE (15)

MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single space typewritten lines) (16)

On April 12, 1986, at 0100 hours, an automatic scram signal was generated on the Reactor Protection System when reactor vessel water level dropped to +13 inches. No control rods were required to move since all rods were inserted and a manual scram had already been initiated as a requirement for a hydrostatic test. The automatic scram signal resulted while operators were lowering water level to between 90 and 100 inches as part of procedure SE-100-002 'ASME CLASS 1 BOUNDARY SYSTEM LEAKAGE/HYDROSTATIC PRESSURE TESTING.' While the operator was monitoring vessel level, using the Shutdown Range Indicator, a packing leak developed on the equalizing valve. This caused a loss of inventory in the reference leg and produced an error of approximately 100 inches on the Shutdown Range Indicator. The operator thought he was reducing reactor vessel level from approximately +109 inches, when actually level was dropping near the scram setpoint of +13 inches. After the scram signal was received, Operations personnel raised water level by increasing CRD flow and isolating letdown to the condenser. The valve packing was adjusted and the reference leg refilled.

As an action to reduce the possibility of recurrence, the event will be reviewed by the operators during their next requalification training cycle.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) SUSQUEHANNA STEAM ELECTRIC STATION UNIT ONE	DOCKET NUMBER (2) 0 5 0 0 0 3 8 7	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
		8 6	0 1 6	0 0	0 2	OF 0 2

TEXT (If more space is required, use additional NRC Form 366A's) (17)

On April 12, 1986, at 0100 hours, an automatic scram signal was generated on the Reactor Protection System (RPS) (EIIS CODE: JC) when reactor vessel water level dropped to +13 inches. No control rods were required to move since all rods were inserted and a manual scram had already been initiated as a requirement for a hydrostatic test. At the time of the RPS actuation, Unit One was in Cold Shutdown (Condition 4), and reactor pressure was 0 psig. Water level was being controlled via Control Rod Drive (CRD) (EIIS CODE: CD) Make-up and condenser letdown through the Residual Heat Removal (RHR) (EIIS CODE: BO) System.

The reactor vessel was being depressurized per procedure SE-100-002 'ASME CLASS 1 BOUNDARY SYSTEM LEAKAGE/HYDROSTATIC PRESSURE TESTING' when the scram signal was generated. As part of the depressurization, operators were reducing vessel level from approximately 217 inches to between 90 and 100 inches. As level was reduced, a packing leak developed on the equalizing valve for the Shutdown Range Indicator. This caused a partial loss in the reference leg inventory and produced an error of approximately 100 inches on the indicator. The licensed operator, monitoring the water level, thought level was at +109 inches when actually it was approaching the scram setpoint of +13 inches. At approximately 0049 hours, as level decreased from +154 inches (as read from the Shutdown Range Indicator), the Narrow Range Indicator came on scale, reading about +59 inches (actual vessel water level). The Narrow Range Indicator which reads from 0 to +60 inches, utilizes a different reference leg than the Shutdown Range Indicator. At 0054 hours, when the water level dropped below +30 inches, the control room low level annunciator sounded. The licensed operators in the control room acknowledged the alarm. They reasoned that the level readings indicated on the narrow range monitor were inaccurate since the narrow range indicator is calibrated for use during normal reactor operation (when the reactor is flooded the operators are trained to use the shutdown range indicator). After concluding the narrow range indications were inaccurate and the low level alarm false (the low level alarm is triggered by the narrow range indication) the operators continued to lower vessel level until the scram setpoint was reached. In reality the wide range instrumentation was in error due to the packing leak on the equalizing valve. Based on the Narrow Range Indicator, the minimum water level reached was approximately +10 inches (approximately 14 feet above the top of the active fuel).

After the scram signal was received, operations personnel raised reactor water level by increasing CRD flow and isolating letdown to the condenser. Instrumentation and Control (I&C) technicians corrected the indication problem by adjusting the valve packing and refilling the reference leg with approximately 103 inches of water.

As an action to reduce the possibility of recurrence the event will be reviewed by the operators during their next requalification training cycle.



Pennsylvania Power & Light Company

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May 12, 1986

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 86-016-00
FILE R41-2
PLAS - 175

Docket No. 50-287
License No. NPF-14

Attached is Licensee Event Report 86-016-00. This event was determined reportable per 10CFR50.73 (a) (2) (iv), in that an automatic actuation of the Reactor Protection System resulted when a valve packing leak caused an error in vessel level indication.

T.M. Crimmins, Jr.
Superintendent of Plant-Susquehanna

JAH/cdn

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