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FACIL:50-387 Susquehanna Steam Electric Station, Unit 1, Pennsylva 05000387
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SUBJECT: LER 89-002-00:on 890112,operator error caused feedwater flow transient & reactor scram.

W/8 ltr.

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NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Susquehanna Steam Electric Station - Unit 1	0 5 0 0 0 3 8 7	1 OF 0 4

TITLE (4)
Operator Error Caused Feedwater Flow Transient and Reactor Scram

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)
										0 5 0 0 0
0 1	1 2	8 9	8 9	0 0 2	0 0	0 2	1 3	8 9		0 5 0 0 0

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

OPERATING MODE (9): 1
POWER LEVEL (10): 0 2 0

[ ] 20.402(b)	[ ] 20.405(c)	[X] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
Richard R. Wehry - Power Production Engineer - Compliance	7 1 7	5 4 2 - 3 6 6 4

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

At 0415 hours on January 12, 1989, with Unit 1 operating at approximately 20% power, a reactor scram occurred due to actuation of the Reactor Protection System (RPS). Operations was in the process of transferring from Startup Level Control to Auto Feedwater Level Control when control of level was lost due to a rapid increase in feedwater flow rate. The reactor level reached the +54" level which results in a trip of the main turbine. The large cold water addition caused reactor power to increase past 24%, which resulted in the RPS actuation upon turbine trip. The required plant equipment response during the transient was per design. The cause of the event was attributed to cognitive operator error. A cooldown of 101°F was experienced over the first hour following the scram. This exceeded the Tech Spec maximum cooldown rate of 100 degrees F per hour during a one hour period, but was not immediately identified. As a result, Tech Spec Action requirements were not properly implemented. An engineering analysis concluded that no adverse effects on the reactor coolant system structural integrity occurred as a result of the temperature deviation. Training was conducted for all licensed operators prior to assuming shift duties before the next startup. Enhancements for clarity purposes were made to the operating procedure. The post transient evaluation procedure was revised to add an additional review of post transient reactor coolant temperatures by the Shift Technical Advisor. Operations Training, stressing the importance of paying close attention to detail when monitoring reactor cooldown rates, has been scheduled.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Unit 1 Susquehanna Steam Electric Station	0 5 0 0 0 3 8 7	8 9	— 0 0 2 —	0 0	0 2 OF 0 4

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

DESCRIPTION OF EVENT

At 0415 hours on January 12, 1989, with Unit 1 operating at approximately 20% power, a reactor scram occurred due to actuation of the Reactor Protection System (RPS; EIIS Code: JC). Operations was in the process of transferring from Startup Level Control to Auto Feedwater Level Control (EIIS Code: JB) when control of level was lost due to a rapid increase in the feedwater flow rate to the vessel. The reactor water level reached the +54" level which results in a trip of the main turbine via control valve fast closure. The large cold water addition caused reactor power to increase past 24%, which enabled the RPS trip on control valve fast closure, resulting in the scram upon turbine trip. Because reactor power increased past 24%, both Reactor Recirculation Pumps (EIIS Code: AD) tripped via the Recirc Pump Trip (RPT) breakers. The immediate actions of EO-100-101, Reactor Scram, were performed. No Safety Relief Valves lifted since reactor pressure stayed below 940 psig. No level based isolations occurred. Required equipment response during the transient was per design and no ESF systems were challenged. A cooldown of 101°F was experienced over the first hour following the scram. (A maximum cooldown of 137°F was experienced during the first 45 minutes.) This exceeded the Tech Spec maximum cooldown rate of 100 degrees F per hour during a one hour period. Due to the tripping of the Reactor Recirculation Pumps and Control Rod Drive (EIIS Code: AA) flow not being reduced to 20-25 gpm, some thermal stratification occurred in the bottom head area resulting in the temperature deviation. Following the restart of the Reactor Recirc Pumps, bottom head drain temperature stabilized at approximately 430 degrees F and a normal reactor cooldown followed. The deviation was discovered on January 16, 1989. As a result, Tech Spec 3.4.6.1 Action requirements were not properly implemented.

CAUSE OF EVENT

The reactor scram was caused by cognitive personnel error (utility - licensed operator). Errors in three specific areas contributed to the reactor vessel level transient and caused the scram:

1) Proper automatic feedwater level control was not established/verified in accordance with the operating procedure.

2) Reactor Feed Pump discharge pressure was not controlled in accordance with the operating procedure.

3) The Reactor Feed Pumps were not placed in service in accordance with the operating procedures. Two Reactor Feed Pumps were placed in service feeding the vessel at the same time. The procedure requires having one feed pump in service at the conditions immediately preceding the event. The second feed pump should not have been placed in service until 30% reactor power:

Concurrent with the scram, the Reactor Recirculation Pumps tripped via the EOC-RPT breakers and CRD flow increased to approximately 200 gpm. For approximately ten minutes following the scram, there was insufficient level in the vessel to support natural circulation, until such time that the 'B' Reactor Feed Pump was restarted. Reactor Recirc Pumps were not restarted until approximately one hour after the scram. These two conditions contributed to thermal stratification in the bottom head area. CRD flow was not decreased to 20-25 gpm as required by EO-100-101, Reactor Scram, when a Recirc Pump is not started. Flow was set at 60 gpm, adding additional cold water to the bottom head area increasing the temperature deviation. The temperature deviation was not detected until subsequent review by Shift Supervision four days later. The failure to properly follow the procedure and the failure to identify and immediately report the out-of-limit temperature condition were considered to be cognitive personnel errors (utility - licensed operator).

REPORTABILITY/ANALYSIS

This event was determined reportable per 10CFR50.73(a)(2)(iv) in that an unplanned Engineered Safety Feature (ESF) actuation occurred when the Reactor Protection System (RPS) initiated an automatic reactor scram. The plant was safely shut down and there were no safety consequences or compromise to public health or safety. Required equipment response during the transient was per design and no ESF systems were challenged.

The failure to identify that the Tech Spec maximum cooldown rate of 100 degrees F in a one hour period was exceeded, thus failing to immediately comply with the required Tech Spec actions until four days later, was determined to be reportable per 10CFR50.73(a)(2)(i)(B). The vessel bottom head drain temperature was restored within the Tech Spec limits within 30 minutes of when the limit had been exceeded. However, since the out-of-limit condition was not identified until approximately four days later, the remaining actions (i.e., perform an engineering evaluation to determine the effects on structural integrity of the reactor coolant system and determine that the system remains acceptable for continued operations) were not completed until that time. Thus, a violation of Tech Spec 3.4.6.1 ACTION requirements occurred. Upon identification of the cooldown rate deviation on 1-16-89, a preliminary engineering evaluation was performed. This initial assessment concluded that the structural integrity of the reactor coolant pressure boundary was not compromised and authorization to continue power ascension was given. The formal evaluation, which followed, determined that fatigue usage due to the event remains within design limits and that brittle fracture was not a concern. It was concluded, therefore, that the structural integrity of the reactor coolant system remains within design limits and that the unit is acceptable for continued operation. There were no safety consequences or compromise to public health or safety as a result of the out-of-limit condition.

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

CORRECTIVE ACTIONS

Operations Supervision discussed the incident with the operator involved and conducted training for all licensed operators prior to their assuming shift duties before the Unit 1 startup. The Nuclear Training Center provided additional training and practice with feedwater controls, focusing on the transfer to automatic feedwater level control evolution. Enhancements for clarity purposes were made to the applicable operating procedure concerning the establishment of automatic feedwater level control. The post reactor transient procedure was revised to include a review of Operations' reactor vessel cooldown surveillance and a review of reactor coolant temperature indicators by the Shift Technical Advisor for a 1 to 2 hour post transient period. Training for Operations personnel has been scheduled concerning the out-of-limit cooldown temperature incident. This training will stress the importance of paying close attention to detail when monitoring cooldown/heatup rates and the immediate notification of Shift Supervision if any limit is exceeded.

ADDITIONAL INFORMATION

LER 85-023-00 described an event on Unit 2 (Docket No. 50-388) in which Unit 2 scrammed from 87% power due to an inadvertent action by an I&C Technician (utility - non-licensed) during performance of a surveillance test on the feedwater level control system. The cognitive personnel error caused a rapid feedwater flow increase, resulting in a turbine control valve fast closure trip and a RPS actuation.

LER 83-097-00 described an event on Unit 1 in which the Tech Spec heatup/cooldown rate of 100°F per hour was exceeded in the Reactor Recirculation Loops.

Pennsylvania Power & Light Company
Two North Ninth Street • Allentown, PA 18101 • 215/770-5151

February 13, 1989

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

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 89-002-00
FILE R41-2
PLAS - 352

Docket No. 50-387
License No. NPF-14

Attached is Licensee Event Report 89-002-00. This event was determined reportable per 10CFR50.73(a)(2)(iv) and 10CFR50.73(a)(2)(i)(B) in that the Reactor Protection System actuated upon a Turbine Control Valve Fast Closure and the Technical Specification Action requirements for Tech Spec Section 3.4.6.1 were not properly implemented following the scram.

JA Blakeslee
for R.G.Byram
Superintendent of Plant - Susquehanna

RRW/mjm

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