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MAY 18 2001

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 50-387/2001-001-00
PLA - 5315 FILE R41-2

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

Attached is Licensee Event Report (LER) 50-387/2001-001-00, which discusses a design deficiency in the Standby Liquid Control System that would prevent the system from meeting the criteria in the ATWS rule, 10CFR50.62. This report is submitted as a voluntary LER, in order to communicate the issues and corrective actions associated with what is currently considered a beyond-design basis scenario in the Susquehanna FSAR.

Bryce L. Shriver
Vice President – Nuclear Site Operations

Attachment

cc: Mr. H. J. Miller
Regional Administrator
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

cc: Mr. S. L. Hansell
Sr. Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 35
Berwick, PA 18603-0035

IE22

NRC FORM 366 (1-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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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TITLE (4)
Standby Liquid Control System Unable to Meet Requirements of the ATWS Rule for a LOOP / ATWS Event

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME: Susq. SES - Unit 2	DOCKET NUMBER: 05000388
03	01	2001	2001	001	00	05	18	2001	FACILITY NAME:	DOCKET NUMBER: 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)							
			20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
POWER LEVEL (10)	100		20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
			20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
			20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)	X	OTHER Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
			20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
			20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
			20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
			20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Gerard M. Machalick - Nuclear Licensing	570 / 542-3861

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) **(16)**

During Unit 1 and Unit 2 operation at 100% power, a discrepancy was discovered between the Standby Liquid Control System (SLCS) design pressure and the maximum pressure expected during a Loss Of Offsite Power / Anticipated Transient Without Scram (LOOP/ATWS) event. In the LOOP/ATWS scenario, the SLCS would not be able to inject the 82.4 gpm required by the ATWS rule, 10CFR50.62. The SLCS design is acceptable for all other ATWS scenarios. The cause of the SLCS design deficiency was a lack of coordination between the ATWS analysis and the SLCS design evaluation. This allowed a disparity to exist between system design and expected system performance. Corrective actions include modification of Unit 1 and 2 SLCS to allow injection of sodium pentaborate solution into the reactor at rated flow during a LOOP/ATWS. An assessment was performed to evaluate the ability of the SLCS to achieve the objectives of the ATWS rule for a LOOP/ATWS event. The assessment shows that there is reasonable assurance that the ATWS rule objectives will be achieved, and that there were no adverse consequences to the health and safety of the public as a result of this event.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

EVENT DESCRIPTION

A discrepancy was discovered between the Standby Liquid Control System (SLCS)(EIIS: BR) design pressure and the maximum pressure expected during a Loss Of Offsite Power / Anticipated Transient Without Scram (LOOP/ATWS) event. Specifically, the SLCS is designed to inject against a reactor steam dome pressure of 1106 psig, which corresponds to the lowest Main Steam Relief Valve (MSRV) (EIIS: SB) set point in the relief mode. In this mode of operation, the MSRVs require a compressed gas supply. During the LOOP/ATWS event, the compressed gas supply is assumed to be exhausted due to multiple valve actuations, and the MSRVs operate in the safety mode at a higher pressure, up to 1195 psig. With reactor pressure in the range of the MSRV safety set points, SLCS pump discharge pressure could be high enough to cause lifting of the pressure relief valves on the pump discharge piping. This would allow sodium pentaborate solution to be recycled back to the pump suction rather than being injected into the reactor. In the LOOP/ATWS scenario, the SLCS would not be able to inject the 82.4 gpm required by the ATWS rule, 10CFR50.62.

CAUSE OF EVENT

The cause of the SLCS design deficiency was a lack of coordination between the ATWS analysis and the SLCS design evaluation. This allowed a disparity to exist between system design and expected system performance.

Contributing factors were:

- Original SLCS design was based on the MSRV relief mode for maximum pressure.
- SLCS design requirements evolved with the implementation of the ATWS rule. The implementation was incorrectly based on original (MSRV relief mode) design assumptions.
- 1993 Susquehanna Power Uprate evaluations for SLCS and ATWS were performed separately, each utilizing different pressure limit assumptions (MSRV relief/safety mode). Different work groups within GE and PPL were responsible for the two evaluations, and the connection was not made between the MSRV safety mode pressure and SLCS maximum discharge pressure.
- Although design basis documents have been prepared for many plant systems, the topical document for ATWS design considerations has not been completed.

ANALYSIS / SAFETY SIGNIFICANCE

An assessment was performed to evaluate the ability of the SLCS to achieve the objectives of the ATWS rule for a LOOP/ATWS event. These objectives are to bring the reactor to hot shutdown conditions while maintaining fuel integrity, reactor pressure vessel integrity and primary containment integrity. The assessment shows that there is reasonable assurance that the ATWS rule objectives will be achieved, based on the following:

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

- One of two SLCS pump discharge relief valves will lift during the event at reactor steam dome pressure corresponding to MSRV safety settings. The second SLCS pump will inject into the reactor, and all ATWS objectives will be met with a maximum suppression pool temperature of 180 degrees Fahrenheit (F).
- If both SLCS pump discharge relief valves are assumed to lift in the event, the peak suppression pool temperature is 219 F. Although the ATWS criteria of 190 F is exceeded, this temperature is less than the design limit of 220 F and containment pressure is maintained well within the design value of 53 psig. Operation of the Residual Heat Removal pumps at the elevated temperatures was evaluated and found to be acceptable.

The risk significance of operating in a condition where the SLCS will inject sodium pentaborate solution into the RPV at an injection rate less than that assumed in the licensing basis analysis for an ATWS/LOOP event has been considered. The frequency of an ATWS/LOOP event is 9.5E-7 per cycle. The Integrated Plant Evaluation (IPE) cumulative Core Damage Frequency (CDF) for one SLCS pump injection is 2.2E-9 per cycle. This safety assessment demonstrates that SLCS injection with one pump will be achieved. Since the IPE already considers one-pump injection, there is no impact on the IPE cumulative CDF for one SLCS pump injection. Based on the results of the above assessments, there were no adverse consequences to the health and safety of the public as a result of this event.

The evolution of system design that occurred for the SLCS has not occurred for other plant systems. It appears that the SLCS design disparity is an isolated issue at Susquehanna.

This report is submitted as a voluntary LER, in order to communicate the issues and corrective actions associated with what is currently considered a beyond-design basis scenario in the Susquehanna FSAR.

The issue of non-compliance with the ATWS rule due to a SLCS design deficiency for the ATWS/LOOP event has not emerged previously in the industry. Other plants of similar design have been informed of this issue via the Boiling Water Reactor Owner's Group, and this voluntary LER will further the industry-wide communication and understanding of this issue.

CORRECTIVE ACTIONS

The Unit 2 SLCS has been modified to allow injection of sodium pentaborate solution into the reactor at rated flow during a LOOP/ATWS. The modification consisted of raising the design SLCS pump discharge pressure to accommodate the reactor steam dome pressure associated with MSRV safety mode operation.

Corrective actions that remain to be completed are:

- Modify Unit 1 SLCS to allow injection of sodium pentaborate solution into the reactor at rated flow during a LOOP/ATWS. This action will be completed during the next outage of sufficient duration, no later than the 2002 refueling outage.
- Complete the topical design basis documentation for ATWS.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

ADDITIONAL INFORMATION

Past Similar Events: None

Failed Component: None