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APR 28 2000

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

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 50-387/00-005-00
PLA - 5192 FILE R41-2

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

Attached is Licensee Event Report 50-387/00-005-00. This event was determined to be reportable per 10CFR50.73(a)(2)(ii) in that both isolation check valves on the CRD System exceeded their leakage acceptance criteria. These valves are designed to prevent back flow from the reactor vessel to piping and components outside of secondary containment. The check valves were replaced with check valves of a different design and the new valves successfully passed the leak test. There were no consequences to the health or safety of the public.

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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

JE22

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
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: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
CRD Seismic Island Check Valves Did Not Meet Acceptance Criteria

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 NAME	DOCKET NUMBER
										05000
03	29	00	00	-- 005 --	00	04	28	00	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	0	20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
		20.2203(a)(1)	20.2203(a)(3)(i)	X 50.73(a)(2)(ii)	50.73(a)(2)(x)
		20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
		20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
		20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
		20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Cornelius T. Coddington – Senior Engineer, Licensing	610 / 774-4019

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	AA	CKV	A391	Y					

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)

On March 29, 2000, at 0200 hours, with Unit 1 in Condition 5 (Refueling) at 0 percent power, both of the in-series Control Rod Drive System (CRD) Seismic Island Check Valves (non 10CFR50, Appendix J, Option B valves) failed their leakage acceptance criteria. Both check valves had leakage past the valve seat that was not quantifiable. The valves were disassembled. The cause of failure for the check valves is believed to have been due to oxide buildup on the valve body bore and the disc. This buildup resulted in the valve disc sticking in the lifted position. The valves were replaced with in-line check valves. A leak test was performed successfully on each of the new valves. Since no design basis accident occurred, there were no safety consequences or compromise to the public health or safety as a result of the check valves not passing their leak test.

EVENT DESCRIPTION

On March 29, 2000, at 0200 hours with Unit 1 in Condition 5 (Refueling) at 0 percent power, both of the in-series Control Rod Drive System (CRD) (EIIS Code: AA) Seismic Island Check Valves (non 10CFR50, Appendix J, Option B valves) (EIIS Code: AA) failed their leakage acceptance criteria. Both check valves had leakage past the valve seat that was not quantifiable. The valves were disassembled and inspected after being transported to a remote location. The valves were both found in the closed position. The valve internals (body and disc) and the seating surfaces had a light coating of oxide on them.

CAUSE OF EVENT

The root cause for the failure was unable to be determined due to significant disturbance to the valves prior to disassembly for inspection. The cause of failure for the check valves is believed to be oxide buildup on the valve body bore and the disc. This buildup probably resulted in the valve disc sticking in the lifted position.

REPORTABILITY/ANALYSIS

The CRD Seismic Island in each unit is designed to provide a 30-day water seal in the CRD supply line post-LOCA. The CRD Seismic Island eliminates a potential water bypass leakage pathway from the CRD insert/withdrawal lines that penetrate primary containment to the CRD supply line that penetrates secondary containment. In addition, the Seismic Island prevents the CRD purge supply lines to the reactor recirculation pumps from becoming an air bypass leakage pathway. During the performance of leak rate testing in Unit 1, both CRD Seismic Island Check Valves had leakage past the valve seat that was not quantifiable. The leak rate test pressure was unable to be achieved. This constituted a condition that involves a degradation of secondary containment bypass leakage for Unit 1. This event was determined to be reportable in accordance with 10CFR50.73(a)(2)(ii). If a Design Basis Accident-Loss of Coolant Accident (DBA-LOCA) with fuel failure had occurred in Unit 1, the CRD Seismic Island would not have been able to provide the 30-day water seal which could have resulted in the potential for offsite doses in excess of 10CFR100 limits. For a LOCA using realistic assumptions, the offsite doses would not exceed 10CFR100 limits. Since no design basis accident occurred, there were no actual safety consequences or compromise to the public health or safety as a result of not having the 30-day water seal.

As a result of Unit 1 historical test failures, the frequency of testing for both units has been increased from every second refueling outage to every refueling outage and 10CFR50 Appendix J, Option B rules applied. The design of the Unit 1 and Unit 2 CRD Seismic Island check valves are the same, and the valves were installed at approximately the same time. However, the Unit 2 leakage history is significantly better than the Unit 1 leakage history. Unit 2 as-found test results for September 1995, June 1998, and April 1999 are less than 1% of the recent Unit 1 test results. In each case on Unit 2, the test volume was able to reach test pressure and a recordable leak rate was achieved. There have been no instances on Unit 2 where a CRD Seismic Island check valve has stuck open. Based on the Unit 2 leakage history, in the event of a DBA-LOCA on Unit 2, the CRD Seismic Island check valves will perform their function to limit water leakage.

In accordance with the guidelines provided in NUREG-1022, Revision 1, Section 5.1.1, the required submission date for this report was determined to be April 28, 2000.

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

CORRECTIVE ACTIONS

The following corrective actions were identified and completed:

- The Unit 1 lift check valves were replaced with in-line check valves.
- A leak test was successfully performed on each of the new check valves.

The following corrective action was identified and will be completed:

- The CRD Seismic Island Lift Check Valves in Unit 2 will be replaced with in-line check valves.

ADDITIONAL INFORMATION

Past Similar Events: LER 98-007-00, Docket No. 387/License No. NPF-14

Failed Component: Check valves; 146026 and 146027

Manufacturer: Anchor Darling