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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Susquehanna Steam Electric Station - Unit 1	05000387	1 OF 7

TITLE (4)

Loss Of Both Loops Of Residual Heat Removal – Low Pressure Coolant Injection

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
02	28	99	99	-- 001 --	02	11	05	99	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	100

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

	20.2201(b)		20.2203(a)(2)(v)	X	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)	X	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	570 / 542-3294

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
B	BO	FCV	C600	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 February 28, 1999 at 2202 hours, with Unit 1 in Mode 1 (Power Operation) at 100 percent power, it was determined that Unit 1 had been operated for approximately 17 hours on 2/16/99 with the redundant loops of the Low Pressure Coolant Injection (LPCI) mode of Residual Heat Removal (RHR) System inoperable. Operation of the Unit with the loops of RHR LPCI inoperable, placed the Unit outside its design bases. Two diverse subsystems (Core Spray) were available to perform the low pressure safety function. During an investigation of a potential degradation in the 'B' RHR Loop keepfill system, a failed valve on the 'B' RHR Loop was discovered on 2/27/99. Pending further review, it is currently assumed that the 'B' RHR Loop valve had failed on 2/11/99 when the first indications that there was potential degradation in the 'B' RHR Loop keepfill system were noted. The 'A' Loop of RHR had been taken out of service for maintenance on 2/16/99. Both loops of RHR have been restored to operable status. The cause of the failure of the RHR valve was determined to be a result of environmentally assisted stress corrosion cracking due to the introduction of inadequate stem material incorporated by a 1985 modification. Taking the 'A' RHR Loop out of service and thus having both loops of RHR LPCI inoperable was due to a less than adequate investigation into the potential degradation of the 'B' RHR Loop keep-fill system. Corrective actions completed include: interim replacement of the broken and cracked stem/plug assembly in Unit 1 with new like-in-kind material, issuance of modification packages to replace the stem/plug assemblies with redesigned assemblies, replacement of the stem/plug assemblies in both Unit 2 RHR loops valves with the redesigned assemblies, review of specifications to ensure that other valves are not susceptible to environmentally assisted stress corrosion cracking, and training on this event with the appropriate personnel. Corrective actions to be completed include: replacement of the Unit 1 stem/plug assemblies with less susceptible material, revisions to specifications, and evaluation of the process for troubleshooting and investigating plant problems. A review of the Design Basis Accident analysis concluded that there were no safety consequences or compromises to the health and safety of the public. In addition, an expanded evaluation of the safety consequences of having the 'B' loop of RHR out of service from the last time the HV-151F017B valve was stroked (November 19, 1998) to March 1, 1999 was performed using the new NRC event evaluation methodology. In all cases, the significance color was green. Therefore, failure of valve HV-151F017B is a low safety consequence event.

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

EVENT DESCRIPTION

On February 28, 1999 at 2202 hours, with Unit 1 in Mode 1 (Power Operation) at 100 percent power, it was determined that Unit 1 had been operated for approximately 17 hours on February 16, 1999 with the redundant loops of the Low Pressure Coolant Injection (LPCI) mode of Residual Heat Removal (RHR; EIIS Code: BO) System inoperable. Operation of the Unit with the loops of RHR LPCI inoperable placed the Unit in a condition that was outside its design bases. The 'A' RHR Loop was taken out of service on February 16, 1999 to perform maintenance on the loop. Unknown at that time, the 'B' RHR Loop was also inoperable due to a failed closed HV151F017B valve. During an investigation of a potential degradation in the 'B' RHR Loop keep-fill system, the failed valve on the 'B' RHR Loop was discovered on February 27, 1999 when it was disassembled and found to have the stem sheared at the threaded connection to the disk. At this time, it is assumed that the F017B valve failed on February 11, 1999 when the first indications that there was a potential degradation in the 'B' RHR Loop keep-fill system were noted. Both loops of RHR have been restored to operable status.

CAUSE OF EVENT

The cause of the failure of the RHR F017B valve stem was determined to be a result of environmentally assisted stress corrosion cracking. The root causes for the inception of environmentally assisted stress corrosion cracking are that susceptible stem/plug assembly material was incorporated into the plant during a 1985 modification. The susceptible material was incorporated in the modification because: (1) There was an inadequate process in that during the valve modification (in 1985), the modification was not reviewed by persons with all the necessary expertise, (2) PP&L personnel did not comply with modification process requirements in effect at the time (circa 1985), in that Design Inputs were not used to generate the RHR valve specification, and (3) there was an inadequate vendor valve design, in that interference (and therefore unexpected stress intensities) could exist between the plug corner and stem corner radius when these parts were assembled. Taking the 'A' RHR Loop out of service on February 16, 1999 and thus having the redundant loops of RHR LPCI inoperable was due to a less than adequate investigation into the potential degradation of the 'B' RHR Loop keep-fill system on February 12 and 13, 1999.

REPORTABILITY/ANALYSIS

Background

A schematic diagram of the applicable portion of the Unit 1 'B' Loop of RHR is provided as the last page of this LER. ECCS systems at Susquehanna SES are continuously supplied with condensate water at approximately 150 psig by a Condensate Transfer system. This "keep-fill" system assures that the ECCS discharge lines are full of water thereby ensuring that the ECCS will perform properly by injecting its full capacity into the Reactor Coolant system upon demand. This system also prevents water hammer following an ECCS initiation signal. Due to this design, the RHR discharge piping is pressurized to approximately 150 psig. For the Unit 1 B Loop of RHR this pressure is measured at a pressure instrument located at the RHR heat exchanger. (PT E11-1N026B on the diagram.) Pressure is continuously displayed in the control room.

The automatic RHR LPCI injection valve (F015) is normally closed. The RHR LPCI throttle valve (F017) is normally open. The Condensate Transfer system taps into the RHR system between the normally closed F015 valve and the normally open F017 valve. The piping system that contains the F015 valve and F017 valve is the injection flow path for RHR LPCI and the return path for RHR Shutdown Cooling.

Description of Key Events

In June 1983, PP&L experienced problems in Unit 1 with the originally installed RHR LPCI throttle valves (F017A and B). These valves were Anchor/Darling globe valves. The valve disc nut had separated from the disc, thus allowing the disc to fall onto the valve seat. This resulted in isolation of the keep-fill system from the RHR discharge header. A temporary modification to these valves was implemented to strengthen the disc nut to disc connection. A design change was implemented to completely replace the Anchor/Darling globe valves with CCI drag valves. The Unit 1 and Unit 2 RHR F017 valves were replaced during the Unit 1 and Unit 2 First Refueling Outages in the Spring of 1985 and Spring of 1986. During this replacement and unknown to PP&L, the valve stem/plug assembly material in the new CCI valves was susceptible to environmentally assisted stress corrosion cracking.

Electrical maintenance was performed on the Unit 1 'B' loop of RHR on February 11, 1999 and the appropriate Limiting Condition for Operation (LCO) was entered. The RHR F017B valve was stroked shut and then open on February 11, 1999. Stroke times were normal. When the RHR System was filled and vented in preparation for the full flow test on February 11, 1999, system pressure did not respond as expected. A condition report was written on February 11, 1999, documenting the fact that loop pressure dropped unexpectedly and was slow to recover when the fill and vent evolution was performed. The condition report suggested that the cause was low flow in the keep-fill system. On the morning of February 12, 1999, the pressurization problem was investigated. It was determined that the most likely cause of this condition was blockage in the small diameter keep-fill piping supplying the 'B' Loop of RHR. The most likely location of the blockage was determined to be at two 2" spring actuated check valves located in the keep-fill piping. Written operability assessments of the keep fill pressurization problem were performed before the system was returned to service.

Later on February 12, 1999 the Unit 1 'B' loop was taken out of service again to inspect and repair, as necessary, the keep-fill check valves which were believed to be the location of the blockage. No obstructions or flow restrictions were found. Additional investigation was performed in an attempt to locate the blockage. The investigation was unsuccessful in determining the source of the obstruction. RHR Loop 'B' continued to take a longer time than normal to repressurize following venting. Operations and Engineering personnel agreed that the RHR system remained operable (Written documentation was prepared on February 14, 1999) since it was believed that the anomaly did not affect the capability of the RHR LPCI system to inject its full capacity into the Reactor Coolant system upon demand or to avoid water hammer following ECCS initiation signal. The system was returned to service on February 13, 1999.

On February 16, 1999, the 'A' loop of the Unit 1 RHR system was taken out of service for scheduled maintenance. The duration of the LCO was approximately 17 hours. During the course of the maintenance, the quarterly valve exercising was performed. The heat exchanger vent valves were cycled. In order to compare the pressure response with that of the Unit 1 'B' loop. Pressure dropped from 150 psig to 130 psig, as indicated in the control room. When the loop fill and vent evolution was performed, no pressure drop was observed on the control room indicator. Given the difference in response between the 'A' and 'B' RHR loops, it was reinforced that there was a keep-fill supply problem that was isolated to the 'B' loop of RHR.

On February 26, 1999, additional troubleshooting of the 'B' Loop of RHR keep-fill system began. During the troubleshooting with additional instrumentation, there was an indication that a substantial differential pressure (about 70 psid) existed across the RHR F017B valve which was indicating open. The calibration and response of the RHR loop pressure instrument was confirmed. At this point, with the additional information obtained in the troubleshooting effort, operability of the RHR LPCI could not be assured. The RHR 'B' LPCI Loop was declared inoperable effective as of 1600 hours.

On the morning of February 27, 1999, Maintenance personnel began disassembly of the RHR F017B valve. The craftsmen found the stem sheared at the threaded connection to the disc. Maintenance proceeded to repair the valve. The reason for the slow repressurization of the RHR system to keep-fill pressure, initially observed on February 11th, was then understood to be caused by the F017B valve disc separated from the stem and seated in the valve body.

On February 28, 1999, it was determined that both loops of RHR LPCI, two subsystems of the low pressure ECCS function, had been inoperable for 17 hours and 15 minutes on February 16, 1999. The Technical Specifications define that both subsystems of RHR and both subsystems of Core Spray comprise the low pressure ECCS subsystems. Since the diverse subsystems, two loops of Core Spray, remained operable, the overall safety function of low pressure ECCS would have been met based on analysis. Per the guidance of NUREG-1022, Rev. 1, however, the event is reportable per 10CFR50.73(a)(2)(v) regardless of whether or not an alternate safety system could have been used to perform the safety function of low pressure ECCS.

Since the 'B' loop of RHR LPCI was incapable of performing its design function for an extended period of time during operation, it was determined to also be reportable per 10CFR50.73(a)(2)(ii) as being outside the design basis, because the RHR LPCI system did not have suitable redundancy (in accordance with the guidance of NUREG-1022, Rev. 1).

Additionally, since the RHR F017B valve was inoperable from February 11, 1999 and the Technical Specification LCO 3.5.1 could not be met, this event is also reportable in accordance with 10CFR50.73(a)(2)(i).

The Unit 1 RHR 'B' LPCI Loop was returned to service and applicable LCOs were cleared at 1745 on March 1, 1999.

<u>Safety Analysis</u>

The result of the RHR F017B stem failure was that the 'B' loop of RHR LPCI would have been unavailable had an actual demand occurred and RHR Shutdown Cooling would not have been able to be established with the 'B' Loop of RHR after the IST stroke testing on February 11, 1999. No other modes of RHR were affected. However, since no LPCI or RHR Shutdown Cooling operation had occurred from February 11, 1999 to March 1, 1999, by which time the valve had been repaired and declared operable, there were no safety consequences to this event.

The Safety Significance of Unit 1 operation with the redundant loops of RHR LPCI inoperable is determined based on an examination of those analyses that credit LPCI as a mitigating system. The analyses of interest here are those pertaining to the design basis Loss of Coolant Accident (LOCA) and those evaluated in the Independent Plant Evaluation (IPE). With respect to LOCA, the occurrence of a LOCA on Unit 1 while both Loops of RHR LPCI were inoperable would not have resulted in exceeding the 10CFR50.46 Peak Clad Temperature (PCT) limit of 2200 °F. Evaluation of the LOCA analysis indicated that loss of RHR LPCI would be bounded by other analyzed events, and therefore, would not produce a limiting Peak Clad Temperature.

The exact date of failure of the RHR F017B valve could not be determined even after extensive investigation. It is assumed for reportability of this event that the RHR F017B valve failed on February 11, 1999 when the first indications that there was degradation in the 'B' RHR Loop keep-fill system were noted. A conservative assessment of safety significance of having the 'B' loop of RHR out of service from the date of the performance of the last stroke time test of the valve (November 19, 1998) to the date the valve was declared operable (March 1, 1999) was performed. This assessment used the new NRC event evaluation methodology. The assessment included a consideration of the most significant system outages coupled with failure of valve HV-151F017B. In all cases, the significance color was green. Therefore, failure of valve HV-151F017B is a low safety consequence event.

Based on the above, there were no safety consequences or compromises to the health and safety of the public.

In accordance with the guidelines provided in NUREG-1022, Revision 1 Section 5.1.1, the required submission date for the original report was determined to be March 30, 1999.

CORRECTIVE ACTIONS

Corrective actions that have been completed include:

- Replacement of the Unit 1 RHR F017B valve stem/disk assembly with new like-in-kind material.
- Inspection of the equivalent Unit 1 RHR F017A valve. No indication of stem cracks was observed.
- Inspection of both equivalent RHR F017 valves in Unit 2. No indication of stem cracks was observed in the Unit 2 F017B valve. The Unit 2 F017A valve did show indications of stem cracks.
- Completion of the metallurgical analysis by an independent testing laboratory on the failed valve stems.
- Redesign of the entire stem/plug assembly to eliminate interferences which lead to stress concentrations
- Issuance of modification packages to replace existing stem/plug assemblies with redesigned assemblies that are less susceptible to environmentally assisted stress corrosion cracking.

- Replacement of both Unit 2 RHR F017 valve stem/plug assemblies with redesigned assemblies that are less susceptible to environmentally assisted stress corrosion cracking.
- Review and revision of valve specifications to limit the hardness of type 410 stainless steel materials.
- Review of appropriate architect engineer (AE) specifications prepared for PP&L to ensure appropriate design requirements were considered/incorporated into the design.
- Lessons learned training performed to update design personnel regarding the design-related issues surrounding this event.

Corrective actions that are to be completed include:

- Replace the Unit 1 F017A and F017B valves stem/plug assemblies with redesigned assemblies that are less susceptible to environmentally assisted stress corrosion cracking.
- Evaluate improvement in the process for troubleshooting and investigations of plant problems using lessons learned from this event.
- Review appropriate AE generated specifications to ensure that those converted to PP&L specifications met the requirements to incorporate design requirements in the specification or any other applicable PP&L procedure requirements in force at the time.

ADDITIONAL INFORMATION

Past Similar Events: Docket No. 50-387 LER 83-091-00

Failed Component: Broken stem/plug assembly on CCI Drag valve (HV151F017B)

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

(FIGURE 1 DRAWING ATTACHED HERE)

FIGURE 1