CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9909130092 DOC.DATE: 99/09/07 NOTARIZED: NO DOCKET #
FACIL:50-387 Susquehanna Steam Electric Station, Unit 1, Pennsylva 05000387
AUTH.NAME AUTHOR AFFILIATION
METER,J.J. Pennsylvania Power & Light Co.
SAUNDERS,R.F. Pennsylvania Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 99-004-00:on 990808,reactor core isolation cooling manually isolated due to failure of steam leak detection temp switch module.Module replaced & RCIC sys returned to service on 990808.With 990907 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 5
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES: 05000387

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	LPD1-1 PD	1	1	NERSES,V	1	1
INTERNAL:	ACRS	1	1	FILE CENTER	1	1
	NRR/DIPM/IOLB	1	1	~~NRR/DRIP/REXB~~	1	1
	NRR/DSSA/SPLB	1	1	RES/DET/ERAB	1	1
	RES/DRAA/OERAB	1	1	RGN1 FILE 01	1	1
EXTERNAL:	L ST LOBBY WARD	1	1	LMITCO MARSHALL	1	1
	NOAC POORE,W.	1	1	NOAC QUEENER,DS	1	1
	NRC PDR	1	1	NUDOCS FULL TXT	1	1
NOTES:		1	1			

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LISTS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTROL DESK (DCD) ON EXTENSION 415-2083

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 17 ENCL 17

A04

Robert F. Saunders
Vice President - Nuclear Site Operations

Susquehanna Steam Electric Station.
P.O. Box 467, Berwick, PA 18603
Tel. 717.542.3256 Fax 717.542.1504
rfsaunders@papl.com

ppl

September 7, 1999

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

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 50-387/99-004-00
PLA - 5107 FILE R41-2

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

Attached is Licensee Event Report 50-387/99-004-00. This report is being made pursuant to 10CFR50.73(a)(2)(v) and 10CFR50.73(a)(2)(vi), in that the Susquehanna Unit 1 Reactor Core Isolation Cooling (RCIC) was manually isolated in anticipation of closure of its inboard steam supply isolation valve due to a failure of the "B" Steam Leak Detection temperature switch module.

Robert F Saunders

Robert F. Saunders
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

13¹¹ 17

IE22

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)
Susquehanna Steam Electric Station - Unit 1	05000387	1 OF 4

TITLE (4)
Reactor Core Isolation Cooling Manually Isolated Due To Failure Of Steam Leak Detection Temperature Switch Module

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
08	08	99	99	-- 004 --	00	09	07	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)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		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)
Joseph J. Meter - Senior Engineer, Nuclear Licensing	570 / 542-1873

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
A	IJ	CAP	R278	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 August 8, 1999, at 0340 hours, with Unit 1 operating in mode 1 (Power Operation) at 100% power, the Unit 1 Control Room received the Reactor Core Isolation Cooling (RCIC) Steam Leak Detection Logic "B" High Temperature alarm. Control Room Operators verified that the 15-minute logic timer for the "B" RCIC Pipe Routing Area High Temperature Steam Leak Detection was energized and that corresponding temperature indicator was pegged up-scale. The circuit provides an isolation signal to the RCIC Turbine Steam Supply Inboard Isolation valve. The temperature indicator for the "A" Steam Leak Detection circuit was checked and found to be reading normal. The RCIC Pipe Routing Area was verified to have no steam leak present. Control Room Operators manually isolated the Unit 1 RCIC system at 0346 hours to preclude an automatic isolation of the RCIC system. The cause of the erroneous Steam Leak Detection signal was due to a failure of a temperature switch module. The module was replaced and the Unit 1 RCIC system was returned to service on August 8, 1999 at 0735 hours. Analysis of the failed temperature switch module revealed that "infant" failure of a power supply capacitor was the cause for the failure. Inadequate processing (human error) by PP&L of a manufacturer advisory for upgrading aging module capacitors led to installation of the subject module. All spare like-in-kind modules were inspected to ensure they did not contain non-upgraded capacitors. All like-in-kind modules that are in service will be inspected for non-upgraded capacitors also. Similar-type modules will be evaluated to ensure they have an adequate shelf life restriction to protect the capacitors from "infant" failure.

EVENT DESCRIPTION

At 0340 hours on August 8, 1999, with Unit 1 operating in mode 1 (Power Operation) at 100% power, the Unit 1 Control Room received the Reactor Core Isolation Cooling (RCIC) (EIIS Code: BN) Steam Leak Detection Logic "B" High Temperature alarm. Control Room Operators verified that the 15 minute logic timer for the "B" RCIC Pipe Routing Area High Temperature Steam Leak Detection (SLD) (EIIS Code: IJ) was energized and the corresponding temperature indicator was pegged up-scale. After the 15 minute logic timer times-out, the "B" RCIC Pipe Routing Area High Temperature SLD circuit provides an isolation signal to the RCIC Turbine Steam Supply Inboard Isolation valve which is a primary containment isolation valve (EIIS Code: JM). The temperature indicator for the "A" RCIC Pipe Routing Area High Temperature SLD was checked and found to be reading at a normal temperature of 111° F. The isolation setpoint for both the "A" and "B" RCIC Pipe Routing Area High Temperature SLD circuits is 167° F. A Nuclear Plant Operator was dispatched to the RCIC Pipe Routing Area and verified no steam leak was present. Control Room Operators manually isolated the Unit 1 RCIC system at 0346 hours to preclude an automatic isolation of the RCIC Turbine Steam Supply Inboard Isolation valve. No structures, systems or components were inoperable prior to this event, that would have contributed to this event. All equipment operated per design during isolation of the system. There were no Emergency Core Cooling System initiations and no Diesel Generator (EIIS Code: EK) starts. There were no challenges to Containment (EIIS Code: NH). The cause of the erroneous "B" RCIC Pipe Routing Area High Temperature SLD signal was investigated by Instrument and Control (I&C) personnel and was determined to be due to a failure of a temperature switch module. The module was replaced and the Unit 1 RCIC system was returned to service at 0735 hours on August 8, 1999. The High Pressure Coolant Injection (HPCI) (EIIS Code: BJ) system remained operable during the time that the RCIC system was isolated.

CAUSE OF EVENT

Analysis of the failed temperature switch module by I&C personnel revealed that failure of a power supply capacitor was the cause for the failure. This capacitor failure was attributed to an "infant" failure of an electrolytic capacitor in a degraded condition. This module had been in-service for three days prior to its failure. The previous "B" RCIC Pipe Routing Area High Temperature SLD circuit temperature switch module had been replaced because routine surveillance testing had indicated that the module exhibited relay chattering near its setpoint. Storage of the modules in a de-energized state increases the degradation process of the electrolytic capacitors, and therefore increases the chance for "infant" failure. The subject module had been available as a spare since 1984.

The subject module was received on site in 1984. It was assigned a seven-year shelf life in accordance with vendor recommendations. In 1986 a vendor notice was received that stated the seven-year shelf life of the modules could only be extended if upgraded capacitors were installed. This vendor information was processed in accordance with the station's Industry Events Review Program (IERP). The resolution of the IERP stated that modules of this type receive the vendor recommended capacitor upgrade or are disposed of. Less than adequate completion of the IERP recommendations by Plant Staff personnel (utility; non-licensed) led to the subject module not receiving a capacitor upgrade nor disposed of.

REPORTABILITY/ANALYSIS

Per the guidance in NUREG-1022, Rev.1, page 71, this report is being made as a loss of safety system pursuant to 10CFR50.73(a)(2)(v) and 10CFR50.73(a)(2)(vi), in that the Susquehanna Unit 1 (RCIC) was manually isolated in anticipation of closure of its Inboard Steam Supply Isolation valve due to a failure of the "B" SLD temperature switch module. Per NUREG-1022, Rev. 1, page 59, this event was not reported as an Engineered Safety Feature (ESF) actuation since the safety function (i.e. primary containment isolation) had already been completed prior to the RCIC Turbine Steam Supply Inboard Isolation valve receiving the invalid ESF actuation signal.

The safety function of the "B" RCIC Pipe Routing Area High Temperature SLD circuit is to close the RCIC Turbine Steam Supply Inboard Isolation valve in the event of a steam leak in the RCIC pipe routing area. The failure of the temperature switch module would not have prevented this valve from performing this safety function in the event of an actual steam leak.

The RCIC system is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation, accompanied by a loss of coolant flow from the feedwater system, to provide adequate core cooling and control of RPV water level.

The RCIC system was isolated for 3 hours 49 minutes during the event. The out-of-service time allowed for RCIC by the plant's Technical Specifications is 14 days, provided the High Pressure Coolant Injection (HPCI) System is operable. The HPCI system was operable during the time the RCIC system was isolated. This event did not affect the overall plant capability to provide makeup inventory at high reactor pressure since the HPCI system is the only high-pressure injection system assumed to function during a loss of coolant accident.

In accordance with the guidance provided in NUREG 1022, Rev. 1, Item 5.1.1, the required submission date for this report was determined to be 09/07/99.

CORRECTIVE ACTION

After investigating the RCIC SLD Logic "B" High Temperature alarm, Control Room Operators manually isolated the Unit 1 RCIC system at 0346 hours to preclude an automatic isolation of the RCIC Turbine Steam Supply Inboard Isolation valve. The cause of the erroneous "B" RCIC Pipe Routing Area High Temperature SLD signal was investigated by I&C personnel, and was determined to be due to a failure of the power supply capacitor on the temperature switch module.

The following corrective actions have been completed:

- The failed RCIC module was replaced with one containing properly upgraded capacitors and the Unit 1 RCIC system was returned to service on August 8, 1999 at 0735 hours.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Susquehanna Steam Electric Station - Unit 1	05000 387	99 --	004 --	00	4 OF 4

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

- All spare like-in-kind modules were inspected to ensure they did not contain non-upgraded capacitors, and none were found.
- The potential for failure was evaluated for like-in-kind modules that are in-service. "Infant" failures are not expected since all modules have been in-service for at least one year.
- The Industry Events Review Program was directly linked with the stations corrective action program (Condition Reports) in 1998 to ensure proper completion of IERP recommendations.

The following corrective actions are to be completed:

- All like-in-kind modules that are in service will be inspected to determine if they contain properly upgraded capacitors. All modules containing the old style capacitors will be properly upgraded.
- Similar-type modules will be evaluated to ensure they have an adequate shelf life restriction to protect the capacitors from "infant" failures.

ADDITIONAL INFORMATION

Failed Component Identification: Component – Steam Leak Detection Temperature Switch Module
Model – 163C1940
Vendor – General Electric

Previous Events with similar results but with dissimilar causes: None

Previous Events with dissimilar results but with similar causes:

Docket No. 50-387 LER 99-001-01 – Loss of Both Loops of Residual Heat Removal – Low Pressure Coolant Injection

Docket No. 50-388 LER 99-003-00 – Unplanned ESF Actuation – Generator Load Reject and Reactor Scram