CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9703060173 DOC.DATE: 97/02/28 NOTARIZED: NO DOCKET #
FACIL:50-387 Susquehanna Steam Electric Station, Unit 1, Pennsylva 05000387

AUTH.NAME	AUTHOR AFFILIATION
ELLIS,S.J.	Pennsylvania Power & Light Co.
KUCZYNSKI,G.J.	Pennsylvania Power & Light Co.
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 97-004-00:on 970201,reactor core thermal power exceeded 102%.Caused by fuse failure on circuit card in pump's generator speed control loop.Core thermal power reduced to within design limits,circuit card replaced.W/970228 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
	PD1-2 PD	1	1	POSLUSNY,C	1	1
INTERNAL:	ACRS	1	1	AEOD/SPD/RAB	2	2
	AEOD/SPD/RRAB	1	1	FILE CENTER	1	1
	NRR/DE/ECGB	1	1	NRR/DE/EELB	1	1
	NRR/DE/EMEB	1	1	NRR/DRCH/HHFB	1	1
	NRR/DRCH/HICB	1	1	NRR/DRCH/HOLB	1	1
	NRR/DRCH/HQMB	1	1	NRR/DRPM/PECB	1	1
	NRR/DSSA/SPLB	1	1	NRR/DSSA/SRXB	1	1
	RES/DET/EIB	1	1	RGN1 FILE 01	1	1
EXTERNAL:	L ST LOBBY WARD	1	1	LITCO BRYCE,J H	1	1
	NOAC POORE,W.	1	1	NOAC QUEENER,DS	1	1
	NRC PDR	1	1	NUDOCS FULL TXT	1	1
NOTES:		1	1			

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM OWFN 5D-5(EXT. 415-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 26 ENCL 26

CATEGORY 1 DOCUMENT

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

FEB 28 1997

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

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 50-387/97-004-00
PLAS - 696 FILE R41-2

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

Attached is Licensee Event Report 50-387/97-004-00. This report is being made pursuant to NRC Document SSINS #0200, "Discussion of Licensed Power Level", in that the Reactor Core Thermal Power Level exceeded the guidance set forth in that document, due to a Reactor Recirculation Pump Speed Control Loop Component Failure. Power was promptly reduced to less than 100% rated.

G. J. Kuczynski
Plant Manager - Susquehanna SES

Attachment

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

IE22/1

Mr. Kenneth M. Jenison
Sr. Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 35
Berwick, PA 18603-0035

NRC FORM 366 (6-89) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED OMB NO. 3150-0104 EXPIRES: 4/30/92

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)
Reactor Core Thermal Power Exceeded 102%

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 2	0 1	9 7	9 7	— 0 0 4	— 0 0	0 2	2 8	9 7		0 5 0 0 0
										0 5 0 0 0

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

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

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 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)	[X] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	SSINS #0200
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(1)(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)
Stephen J. Ellis - Nuclear Licensing Engineer	717 542-3537

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	AD	SIK	B040						

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)

On February 1, 1997 at 2334 hours, with Unit 1 at 100% Power in Condition 1, the control room received the APRM UPSCALE and the ROD OUT BLOCK alarms. The 'A' reactor recirculation pump had run up to its high speed limit increasing core power. As a result, the core thermal power went to maximum of 103.5% (3560.8 MWth) of rated power (3440 MWth). Speed of the 'B' recirculation pump was reduced to control power. Power was returned to below 3440 MWth in approximately 65 seconds. This constitutes a condition outside of the operating guidance set forth in NRC memorandum SSINS #0200. The root cause of the event is attributed to a failed fuse on a circuit card in that pump's generator speed control loop. The reactor recirculation pump controller failure high is an analyzed accident which completely bounds this event. No licensing safety limits were approached and as such, there were no safety consequences to the plant or compromises to public health and safety. Corrective actions completed include: Reducing core thermal power to within design limits, and replacing the circuit card. In addition, a failure analysis of the removed circuit card will be performed.

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	9 7	— 0 0 4	— 0 0	2 OF 4

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

EVENT DESCRIPTION

On February 1, 1997, with Unit 1 at 100% in Condition 1 (Power Operation), the control room received the APRM UPSCALE and ROD OUT BLOCK alarms. The Operator noticed a mismatch between the 'A' recirculation pump's demand and speed, and observed core power increasing. He immediately attempted to reduce speed of the 'A' recirculation pump, but was unsuccessful. He took manual control of the 'B' recirculation pump and reduced its speed which resulted in lowering Reactor Power. The maximum percent core thermal power reached was 103.5% (3560 MWth). The power excursion above rated (3440 MWth) lasted for approximately 65 seconds.

CAUSE OF EVENT

The controller failure has been traced to a blown 1/8 amp fuse in the circuit card in the speed loop of the recirculation pump. The function of the card is to convert the voltage from the reactor recirculation MG tach generator to a voltage level signal which is compatible with the recirculation demand control signal. Unfortunately, the fuse was discarded before failure analysis could be performed on this component. The circuit card was removed from service. A suitable replacement was obtained and bench calibrated prior to being installed in the control loop. The old circuit card was tagged and will be subjected to further analysis to determine if any other component(s) on the card failed (besides the fuse).

The industry data suggests that no trend exists for fuse failures with any specific group of Bailey cards. For the seventeen events reviewed, a fuse failure is accompanied by physical damage to the card or replacement of the fuse and/or a recalibration check of the card. The historical data supports the conclusion that a fuse failure is caused because of an obvious failure on the card, initiated through human actions or a normal aging of the fuse resulting in the element melting. No physical damage to other components on the card was observed.

Based on the preponderance of data which supports the fuse failing as the initiator of the event, an action to replace similar fuses was considered. Input from both Instrument & Controls and Nuclear System Engineering was considered. This event is a one time occurrence for Susquehanna Steam Electric Station and the fourth similar event reported for the entire industry for use of the Bailey 7000 series control equipment (reactor recirculation, feedwater, HPCI, etc.). Replacing the fuses would then introduce a failure rate component because of "infant mortality". The expectation would be a higher failure rate than what has been experienced in the past. Since no technical basis could be supported based on this single fuse failure, no action to replace all the fuses in the Bailey control cards was taken.

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	9 7	— 0 0 4	— 0 0	3 OF 4

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

REPORTABILITY/ANALYSIS

The described event was determined reportable pursuant to NRC Document SSINS #0200, "Discussion of Licensed Power Level" (AITS.F14580HZ), dated August 22, 1980. This document states, in part, ...In no case should 102% power be exceeded... Contrary to this, due to a failure of the speed control of the 'A' reactor recirculation pump, power reached a maximum value during the transient of 103.5%.

No safety consequences resulted from the equipment failure and the subsequent transient. A Reactor Recirculation Speed Controller-failure high is an analyzed event, being re-analyzed for each new core as part of the re-load analysis. The analyzed event for the specific fuel cycle produces a much more significant transient allowing power to increase to the high neutron flux SCRAM setpoint. The event described in this LER is completely bounded by that analysis. The maximum values for critical fuel parameters recorded during the event were used to run a prediction computer program (PREDICT) to evaluate the effects on the fuel thermal limits. All fuel thermal limits were well within acceptable values. Since 100% power was exceeded, the preconditioning limits were exceeded, as would be expected, but were also well within the flow control envelope. The PREDICT case was run using the conservative assumption of the maximum recorded power for the full duration of the event. No fuel damage occurred as indicated from lack of an increasing trend from either the Offgas System or the Unit 1 Vent Stack Monitoring System (SPING). Based on the above discussion, the health and welfare of the public was not compromised.

In accordance with guidance provided by NUREG 1022, Supplement 1, Item 14.1, the required submission date for this report was determined to be March 3, 1997.

CORRECTIVE ACTIONS

The following immediate corrective actions were taken:

- Power was reduced to less than 100%.
- The circuit card was replaced.

The following actions to prevent recurrence will be completed:

- Analysis will be performed on the removed circuit card to determine if additional causal factors contributed to the failure.

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

- Training will be performed to emphasize the importance of properly tagging and controlling failed equipment removed from the plant.

ADDITIONAL INFORMATION

Failed Component Identification:	Speed Control Circuit Card for the 'A' Reactor Recirculation Pump Bailey 7000 Series - Millivolt Conversion Card
Previous Similar Events:	LER 50-387/94-017-00