PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION

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J. DOERING, JR.
PLANT MANAGER
LIMERICK GENERATING STATION

September 18, 1992
Docket No. 50-352
License No. NPF-39

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Licensee Event Report
Limerick Generating Station - Unit 1

This LER concerns pressure setpoint drift of the Main Steam System safety relief valves due to corrosion induced bonding within the valves such that a single cause resulted in more than two independent trains becoming inoperable in a single safety system.

Reference:	Docket No. 50-352
Report Number:	1-92-016
Revision Number:	00
Discovery Date:	August 28, 1992
Reportability Date:	August 28, 1992
Report Date:	September 18, 1992
Facility:	Limerick Generating Station P.O. Box 2300, Sanatoga, PA 19464

This LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(vii).

Very truly yours,

J Doer

DMS:cah

cc: T. T. Martin, Administrator, Region I, USNRC
T. J. Kenny, USNRC Senior Resident Inspector, LGS

300012

JE22 /

NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Limerick Generating Station, Unit 1

DOCKET NUMBER (2): 0 5 0 0 0 3 5 2

PAGE (3): 1 OF 0 8

TITLE (4): This LER reports pressure setpoint drift of the Main Steam Safety Relief Valves due to corrosion induced bonding within the valves which has been a repetative problem.

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 NAMES	DOCKET NUMBER(S)
										0 5 0 0 0
0 8	2 8	9 2	9 2	– 0 1 6 –	0 0	0 9	1 8	9 2		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)	X	50.73(a)(2)(vii)		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)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(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	
G. J. Madsen, Regulatory Engineer, Limerick Generating Station	2 1 5	3 2 7 - 1 2 0 0

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	S B	R V	T 0 2 0	YES		B	S B	R V	T 0 2 0	YES
B	S B	R V	T 0 2 0	YES		B	S B	R V	T 0 2 0	YES

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 fifteen single-space typewritten lines) (16)

On August 28, 1992, Limerick Generating Station personnel identified that pressure setpoint testing of the fourteen reactor Main Steam System Target Rock Corp., Model 7567 F, pilot operated two-stage safety relief valves (SRVs) revealed that only two SRVs lifted within the Technical Specifications (TS) required limit of ± 1% of the nameplate setpoint as specified in TS Section 3.4.2. The average percent drift of the twelve SRVs which failed to lift within the TS required limit was 4.7%. The root cause for the setpoint drift of the twelve SRVs was primarily corrosion induced bonding between the pilot disc made of either stellite or stainless steel and the stellite seat. The fourteen SRVs will be refurbished using stellite pilot discs, pressure tested, and recertified prior to being reinstalled during the second Unit 2 refuel outage. There were no actual adverse consequences or release of radioactive material as a result of this condition. This condition was determined as reportable on August 28, 1992, since this condition resulted in more than two independent trains becoming inoperable in a single safety system due to a single cause. Therefore, this report is being submitted in accordance with the requirements of 10CFR 50.73(a)(2)(vii).

NRC Form 366
(9-83)

NRC Form 366B (9-83)

LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Limerick Generating Station, Unit 1	0 5 0 0 0 3 5 2	9 2	0 1 6	0 0	0 2 OF 0 8

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (12)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	
B	SV	RV	T020	YES							
B	SV	RV	T020	YES							
B	SV	RV	T020	YES							
B	SV	RV	T020	YES							
B	SV	RV	T020	YES							
B	SV	RV	T020	YES							
B	SV	RV	T020	YES							
B	SV	RV	T020	YES							

NRC FORM 366B (9-83)

Unit Conditions Prior to the Event:

Unit 1 Operational Condition was 1 (Power Operation) at 100% power level.

There were no other structures, systems, or components out of service which contributed to this event.

Description of the Event:

On August 28, 1992, Limerick Generating Station (LGS) plant personnel reviewed completed procedure ST-4-041-210-2," Main Steam Relief Valves Test," which documents compliance with ASME Code Testing and Technical Specifications (TS) Section 3.4.2. During this review plant personnel identified that only two of the fourteen reactor Main Steam System (EIIS:SB) Target Rock Corp., Model 7567 F, pilot operated two-stage safety relief valves (SRVs) (EIIS:RV) lifted within the TS required limit of ± 1% of the nameplate setpoint as specified in TS section 3.4.2. The fourteen SRVs had been removed and tested following the fourth operating cycle which ended on March 20, 1992, and will be refurbished using stellite pilot discs, pressure tested, and recertified prior to being reinstalled during the second Unit 2 refuel outage. Table 1 below provides the test results.

Table 1: Pressure Setpoint Test Results

Note: Automatic Depressurization System (ADS)

SRV S/N		Nameplate Setpoint (PSIG)	As Found Setpoint (PSIG)	%Drift
508	(ADS)	1140	1154	1.2
513		1150	1227	6.6
515	(ADS)	1140	1126	-1.2
516		1130	1247	10.3
517		1150	1227	6.6
518		1130	1208	6.9
519		1150	1174	2.0
520		1150	1180	2.6
521		1130	1208	6.9
522	(ADS)	1140	1119	-1.8
523	(ADS)	1130	1105	-2.2
525		1150	1156	0.53
526		1140	1232	8.0
528	(ADS)	1140	1146	0.53

Reactor overpressure protection for the LGS Nuclear Steam Supply System (NSSS) is provided by the nuclear pressure relief system which includes fourteen pilot-operated SRVs manufactured by Target Rock Corp. and supplied by General Electric (GE). Nominal set pressures for the SRVs are as follows: four at 1130 psig,

five at 1140 psig, and five at 1150 psig. The safety function of the SRVs is to prevent steam pressure excursions from causing the reactor coolant system pressure to exceed the ASME Section III Level B Service (i.e., Upset) limit. This limit is defined as 110% of the design pressure rating for the protected vessel which is, for LGS, 1375 psig (i.e., 1.10 X 1250 psig).

On August 26, 1992, we completed our reportability evaluation and determined that this condition resulted in more than two independent trains becoming inoperable in a single safety system and that this condition also may have resulted in a violation of TS Section 3.4.2. LGS TS Section 3.4.2 requires that, in order to be considered operable, the safety valve function of at least eleven of the fourteen SRVs be within ± 1% of their nameplate set pressure. If the safety pressure relief function of one or more of the eleven required SRVs is considered inoperable, the unit is required to be placed in cold shutdown within twenty-four hours. However, there is no recommended method of verifying functional operability of an installed SRV during plant operation. Therefore, end-of-cycle testing is performed to determine whether the SRVs are in compliance with TS Section 3.4.2. However, this testing does not provide indication as to when the SRV(s) may have failed during the operating cycle to satisfy the TS SRV setpoint drift limits.

Therefore, since this condition did result in more than two independent trains becoming inoperable in a single safety system due to a single cause, this report is being submitted in accordance with the requirements of 10CFR50.73(a)(2)(vii).

Analysis of the Event:

There were no actual adverse consequences associated with this event because there was no release of radioactive material to the environment. SRV setpoint drift would have had no impact on either the Automatic Depressurization System (ADS) function or the manual actuation mode of the SRVs as based on studies performed by GE under guidance of the Boiling Water Reactor Owners' Group (BWROG) SRV Setpoint Drift Fix Program. In both cases, the valve is opened by actuation of the air operator which lifts the pilot rod above the pilot disc and allows main steam pressure to lift the pilot disc and open the valve (see Figures 1 and 2). In the case of an overpressure situation, plant procedures instruct the Reactor Operator to reduce reactor pressure below 1020 psig by reducing reactor power and/or recirculation flow rate. If reactor pressure increases above 1020 psig, a reactor high pressure alarm sounds. A scram is automatically initiated if reactor pressure increases above 1037 psig. In the event that reactor pressure continues to increase, the Reactor Operator has manual control of the SRVs.

As part of the BWROG Setpoint Drift Fix Program, a sensitivity study of BWR plants using the Target Rock Corp. two-stage SRV was performed by GE as documented in Report No. NEDO-22210. This study determined that sufficient overpressure protection margin existed at all plants to tolerate an upward setpoint drift of 10% on each SRV during the limiting pressurization transient.

The actual average setpoint drift for all 14 SRVs in use during the Unit 1 fourth operating cycle was 3.35%. The most severe pressurization transient event was conservatively assumed to be the simultaneous closure of all Main Steam Isolation Valves (MSIVs) with a coincident failure of the MSIV position scram signal. In this case, a reactor scram subsequently occurs on a high neutron flux signal.

A plant specific evaluation was performed for LGS Unit 1 by GE during the 1986 Surveillance Test outage. The results were issued as GE Report No. MDE-85-0786. A 10% setpoint drift above nameplate set pressures was assumed for all fourteen SRVs coincident with the most severe pressurization transient defined above. Under these conditions, peak reactor vessel pressure would have been approximately 1350 psig below the ASME Code Limit of 1375 psig. Additionally, the impact on fuel thermal margin, Loss of Coolant Accident/Emergency Core Cooling System (LOCA/ECCS) performance, High Pressure Coolant Injection/Reactor Core Isolation Cooling (HPCI/RCIC) systems operability, and drywell pressure and temperature response was assessed. The evaluation showed that a 10% setpoint drift of all fourteen SRVs would not have caused any of the plant safety limits to be exceeded or impact safe plant operation. Since only one SRV tested following the fourth Unit 1 operating cycle, exhibited a setpoint 10% above nameplate value, no adverse safety consequences would have resulted had the limiting pressure transient event occurred during the fourth operating cycle since the remaining 13 SRVs would have provided adequate pressure relieving capacity.

Cause of the Event:

The root cause for the setpoint drift of twelve SRVs was primarily corrosion induced bonding between the pilot disc made of either stellite or a high alloy stainless steel (SS) (i.e., PH13-8Mo) and the stellite seat. The corrosion stems from oxidation product build-up due to the presence of moisture and the heated environment. Figures 1 and 2 are illustrations of the Target Rock Corp. SRV model 7567 F SRVs Assembly 77R-000.

Corrective Actions:

As committed to in LER 1-89-036, the fourteen SRVs installed on Unit 1 for its fourth operating cycle were removed for setpoint testing. However, this setpoint testing in addition to industry in-plant experience showed that the modified pilot disc material (i.e., PH13-8Mo) installed prior to the last Unit 1 refuel outage, experienced SRV setpoint drift equal to or greater than the originally installed stellite pilot disc material. Therefore, the BWROG Setpoint Drift Fix committee has reconsidered the use of high alloy SS material as a solution to the SRV drift problem.

The BWROG SRV Setpoint Drift Fix Committee, in which Philadelphia Electric Company (PECo) is participating, is currently engaged in an program to resolve concerns with SRV setpoint drift. Two parallel paths are being taken. One

involves the design, testing, and implementation of a hydrogen-oxygen recombiner (i.e., catalyst) to alter the environment surrounding the pilot valve and thus mitigate pilot disc bonding. The other path involves implementation of pressure switches as an alternate means to actuate the valves. The BWROG plans to pursue implementation of the pressure switch option if the catalyst option fails. The BWROG and PECo expect to make a final decision as to which option will be implemented to resolve SRV Setpoint Drift pending conclusion of ongoing testing.

Previous Similar Occurrences:

LERs 1-87-034, 1-89-036, 1-91-015, and 2-92-010 report Main Steam system SRV setpoint drift.

The cause of each of these events is the same and the issue of resolving the SRV setpoint drift problem is being addressed by the BWROG SRV Setpoint Drift Fix Committee.

Tracking Codes: B - Design, Manufacturing Deficiency

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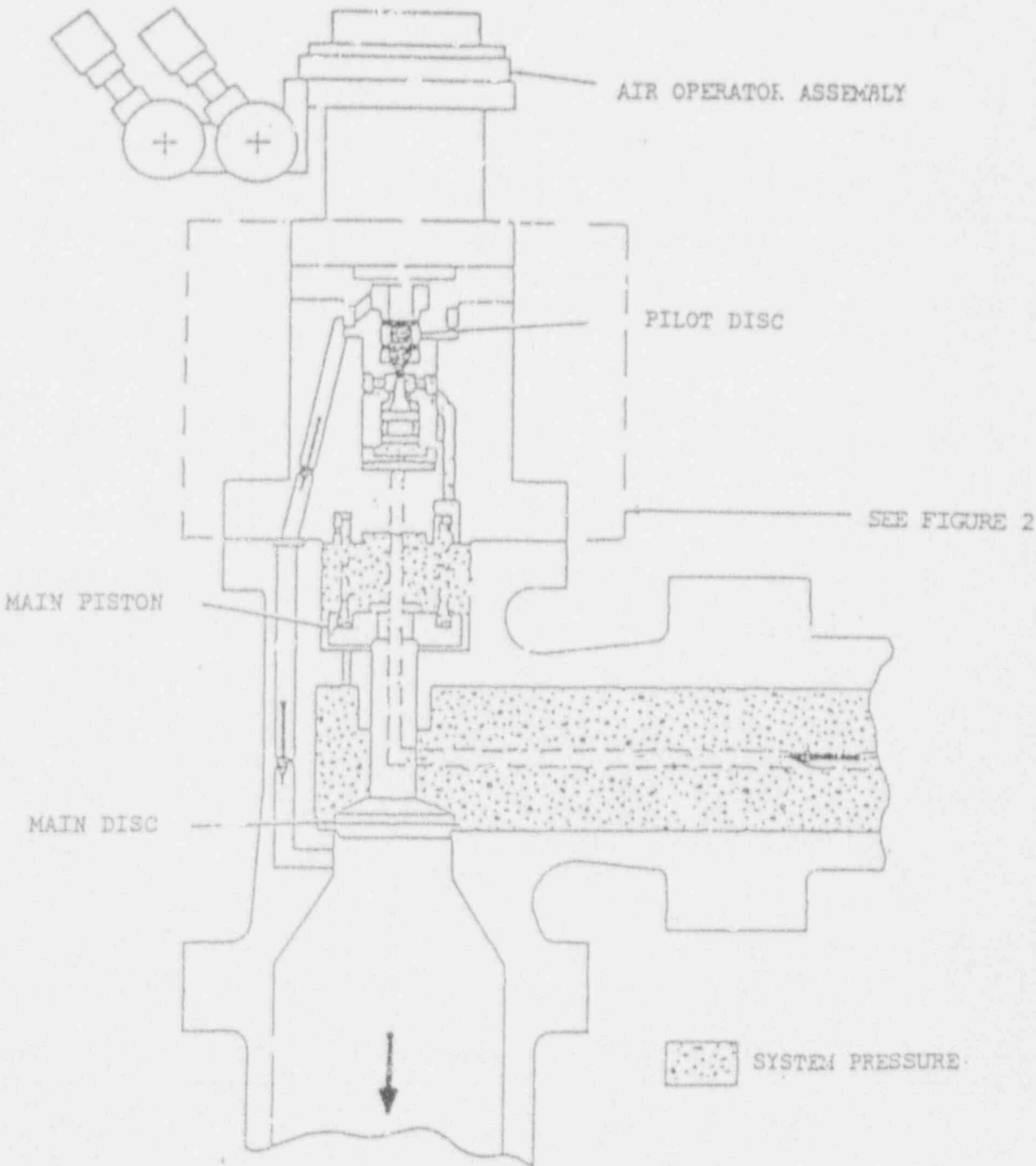
Limerick Generating Station, Unit 1	0 5 0 0 0 3 5 2	9 2	— 0 1 6 —	0 0	0 7 OF 0 8

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

FIGURE 1



FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Limerick Generating Station, Unit 1	0 5 0 0 0 3 5 2	9 2	— 0 1 6 —	0 0	0 8 OF 0 8

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

FIGURE 2