PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION
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J. DOERING, JR.
PLANT MANAGER
LIMERICK GENERATING STATION

July 5, 1991
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:	1-91-015
Revision Number:	00
Discovery Date:	May 28, 1991
Reportability Date	June 6, 1991
Report Date:	July 5, 1991
Facility:	Limerick Generating Station P.O. Box A, Sanatoga, PA 19464

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

Very truly yours,

LA Hopkins for J Doering

WGS:cah

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

JE22 1/1

NRC Form 366 (9-83)

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

LICENSEE EVENT REPORT (LER)

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TITLE (4) This LER Reports Pressure Setpoint Drift of the Main Steam Safety Reflief 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 5	2 8	9 1	9 1	— 0 1 5 —	0 0	0 7	0 5	9 1		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)

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 May 28, 1991, 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 three SRVs lifted within the Technical Specifications (TS) required limit of ± 1% of the nameplate setpoint as specified in TS Section 3.4.2. The root cause for the setpoint drift of the eleven SRVs was primarily corrosion induced bonding between the pilot disc made of either stellite or stainless steel (SS) and the stellite seat. The fourteen SRVs were all refurbished using stellite pilot discs, pressure tested, and recertified prior to being reinstalled during the first 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 June 6, 1991, 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).

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 May 28, 1991, while Limerick Generating Station (LGS) Operations personnel reviewed 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, it was identified that only three 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, tested, and refurbished using stellite pilot discs during the third Unit 1 refueling outage. Ten SRVs lifted within the range of 1% to 4.09% above the nameplate setpoint and one SRV lifted at 1.32% below its nameplate setpoint. Table 1 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
504	(ADS)	1130	1176	4.07
505	(ADS)	1140	1160	1.75
506		1150	1178	2.43
508(B), 527(P)	(ADS)	1140	1159	1.67
509		1140	1173	2.89
510		1150	1175	2.17
511		1150	1180	2.61
514	(ADS)	1140	1164	2.10
521(B), 501(P)		1130	1145	1.33
528(B), 535(P)		1150	1135	-1.32
529	(ADS)	1140	1145	0.45
532(B),512(P)		1130	1138	0.71
533		1130	1165	3.10
534		1150	1153	0.26

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 distributed 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 (Upset) limit. This limit is defined as 110% of the design pressure rating for the protected vessel which is, for LGS, 1375 psig (1.10 X 1250 psig).

On June 6, 1991, 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 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 since no overpressure transients occurred, which would have caused the SRVs to open based on set pressure, during the third cycle of Unit 1 reactor operation. There was no release of radioactive material as a result of this event.

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 (NEDO-22210). This study determined that sufficient overpressure protection margin

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

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 third operating cycle was 1.73%. 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 (note that in this case 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 vessel pressure would have been approximately 1350 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 all fourteen SRVs tested during the third Unit 1 refueling outage exhibited setpoints well within 10% above nameplate values, no adverse safety consequences would have resulted had the limiting pressure transient event occurred during the third operating cycle.

Cause of the Event:

The root cause for the setpoint drift of eleven SRVs was primarily corrosion induced bonding between the pilot disc made of either stellite or PH13-8Mo (i.e., SS) 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 third refuel cycle were removed for setpoint testing. However, this setpoint testing in addition to industry in-plant experience showed that the modified pilot disc material (PH13-8Mo) installed during 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

(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 implement 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 and 1-89-036 reported Main Steam system SRV setpoint drift.

The cause of each of these events are 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

NRC Form 366A
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

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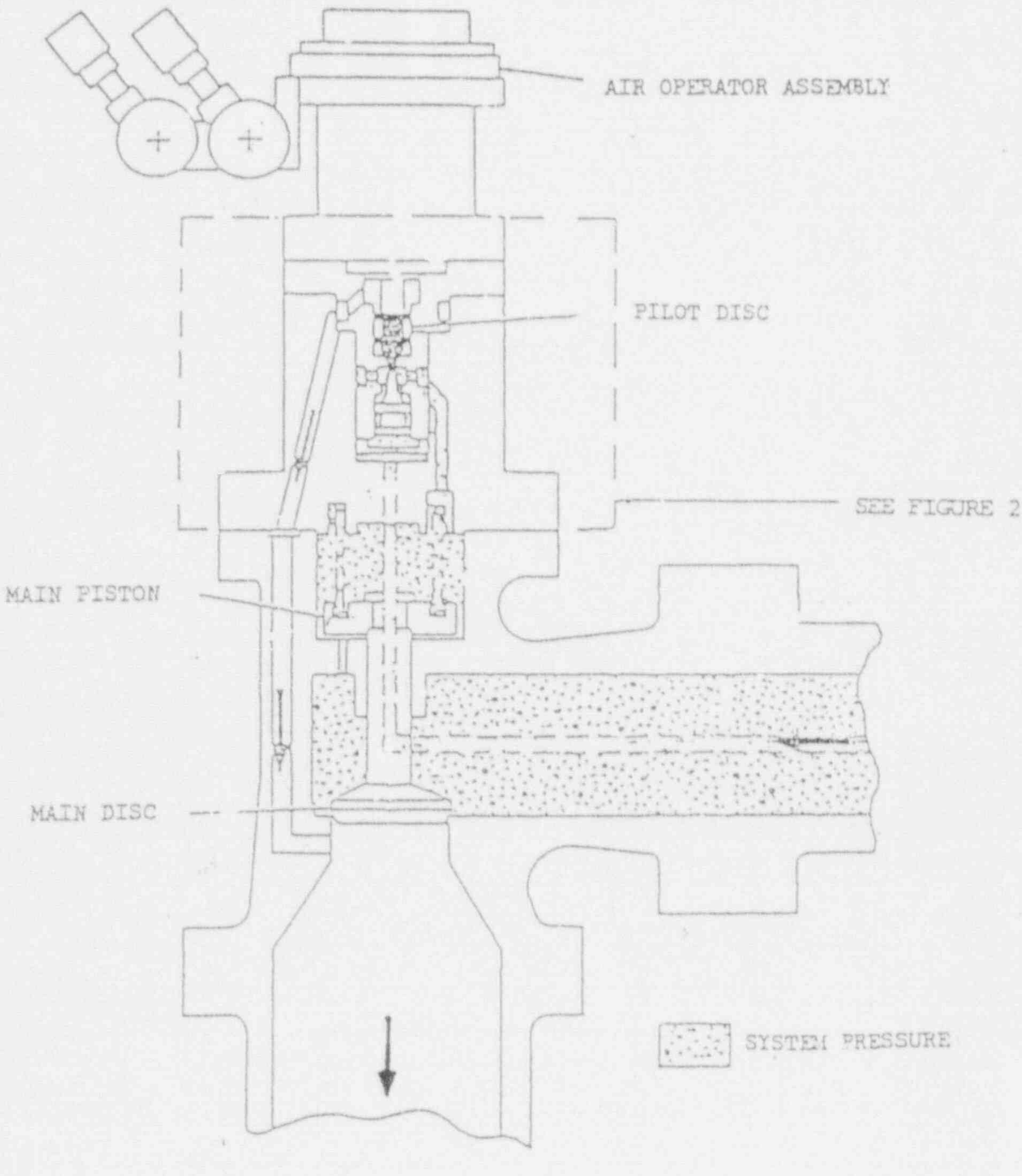
TEXT (If more space is required, use additional NRC Form 366A's) (17)

Figure 1



Target Rock Safety/Relief Valve Model 7567F

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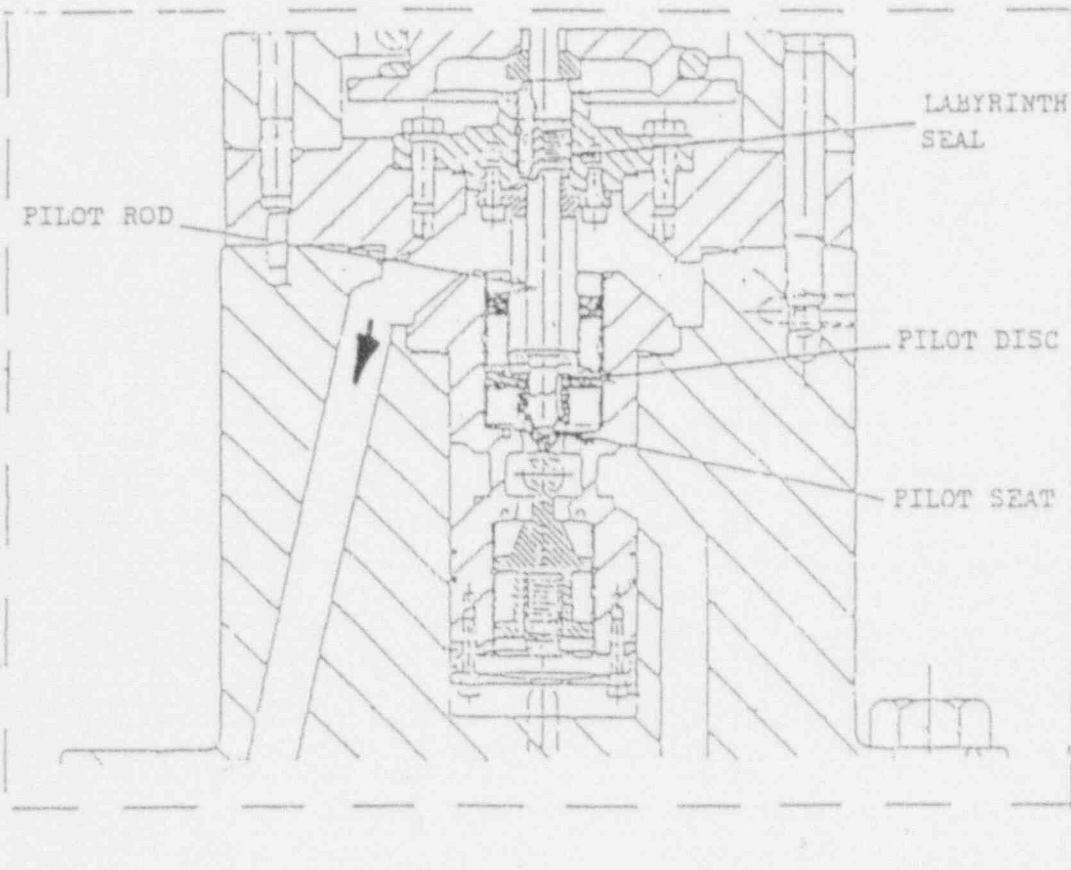
TEXT (If more space is required, use additional NRC Form 366A's) (17)

Figure 2



Target Rock Safety/Relief Valve Model 7567F Internal Schematic

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)
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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	
B	S B	R V	T 0 2 0	YES							
B	S B	R V	T 0 2 0	Y							
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							
B	S B	R V	T 0 2 0	YES							

NRC FORM 366B (9-83)