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LIMERICK GENERATING STATION

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September 19, 1992

Docket No. 50-353

License No. NPF-85

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

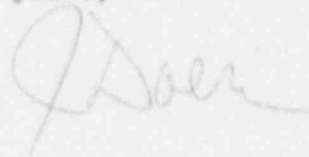
SUBJECT: Licensee Event Report
Limerick Generating Station - Unit 2

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. 50-353
Report Number:	2-92-010
Revision Number:	00
Discovery Date:	August 28, 1992
Reportability Date:	August 28, 1992
Report Date:	September 19, 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,



DMS:cah

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

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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Limerick Generating Station, Unit 2										DOCKET NUMBER (2) 0 5 0 0 0 3 5 3				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 repetitive 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 8	2 8	9 2	9 2	0 1 0	0	0 9	1 9	9 2					0 5 0 0 0			
OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5. (Check one or more of the following) (11)														
1		20.402(b)				20.406(c)				50.73(a)(2)(iv)				73.71(b)		
POWER LEVEL (10)		20.405(a)(1)(ii)				50.36(a)(1)				50.73(a)(2)(v)				73.71(c)		
1 0 0		20.405(a)(1)(iii)				50.36(a)(2)				X 50.73(a)(2)(vi)				OTHER (Specify in Abstract below and in Text, NRC Form 386A)		
		20.405(a)(1)(iv)				50.73(a)(2)(ii)				50.73(a)(2)(vii)(A)						
		20.405(a)(1)(v)				50.73(a)(2)(iii)				50.73(a)(2)(viii)(B)						
		20.405(a)(1)(vi)				50.73(a)(2)(iii)				50.73(a)(2)(ix)						
LICENSEE CONTACT FOR THIS LER (12)																
NAME G. J. Madsen, Regulatory Engineer, Limerick Generating Station										TELEPHONE NUMBER						
										AREA CODE 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 NRC		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC						
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 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 six SRVs lifted within the Technical Specifications (TS) required limit of $\pm 1\%$ of the nameplate setpoint as specified in TS Section 3.4.2. The average percent drift of the eight SRVs which failed to lift within the TS required limit was 2.4%. The root cause for the setpoint drift of the eight 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 fourth Unit 1 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).

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Limerick Generating Station, Unit 2	015000013153	912	010	00	03	OF	08

TEXT (If more space is required, use additional NRC Form 388A's) (17)

Unit Conditions Prior to the Event:

Unit 2 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:

Following the first Unit 2 refueling outage, 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. This review identified that only six of the fourteen reactor Main Steam System (EISS:SB) Target Rock Corp., Model 7567 F, pilot operated two-stage safety relief valves (SRVs) (EISS: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 first Unit 2 operating cycle which end on March 22, 1991, and refurbished using stellite pilot discs, pressure tested, and recertified prior to being reinstalled during the fourth Unit 1 refueling 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
502 (ADS)	1130	1177	4.2
503 (ADS)	1140	1132	-0.70
507	1150	1168	1.6
524	1150	1143	-0.61
530 (ADS)	1140	1133	-0.61
531	1130	1137	0.62
532	1130	1126	-0.35
1209 (ADS)	1140	1153	1.1
1210 (ADS)	1140	1158	1.6
1211	1150	1184	2.9
1212	1150	1188	3.3
1213	1150	1182	2.9
1214	1130	1144	1.2
1215	1140	1140	0.0

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

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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 28, 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 $\pm 1\%$ 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 since no overpressure transients occurred, which would have caused the SRVs to open based on set pressure, during the first cycle of Unit 2 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 as documented in Report No. NEDO-22210. This study determined that sufficient

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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 2 first operating cycle was 1.23%. 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, and the results were issued as GE Report No. MDE-85-0786. The results of this evaluation are also applicable to LGS Unit 2. 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 7375 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 following the first Unit 2 operating cycle exhibited setpoints well within 10% above nameplate values, no adverse safety consequences would have resulted had the limiting pressure transient event occurred during the first operating cycle.

Cause of the Event:

The root cause for the setpoint drift of eight 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:

The fourteen SRVs installed on Unit 2 for its first operating cycle were removed for setpoint testing during the first refuel outage that began on March 22, 1991. However, this setpoint testing in addition to industry in-plant experience showed that the modified pilot disc material (i.e., PH13-8Mo) 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 a program to resolve

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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 1-92-017 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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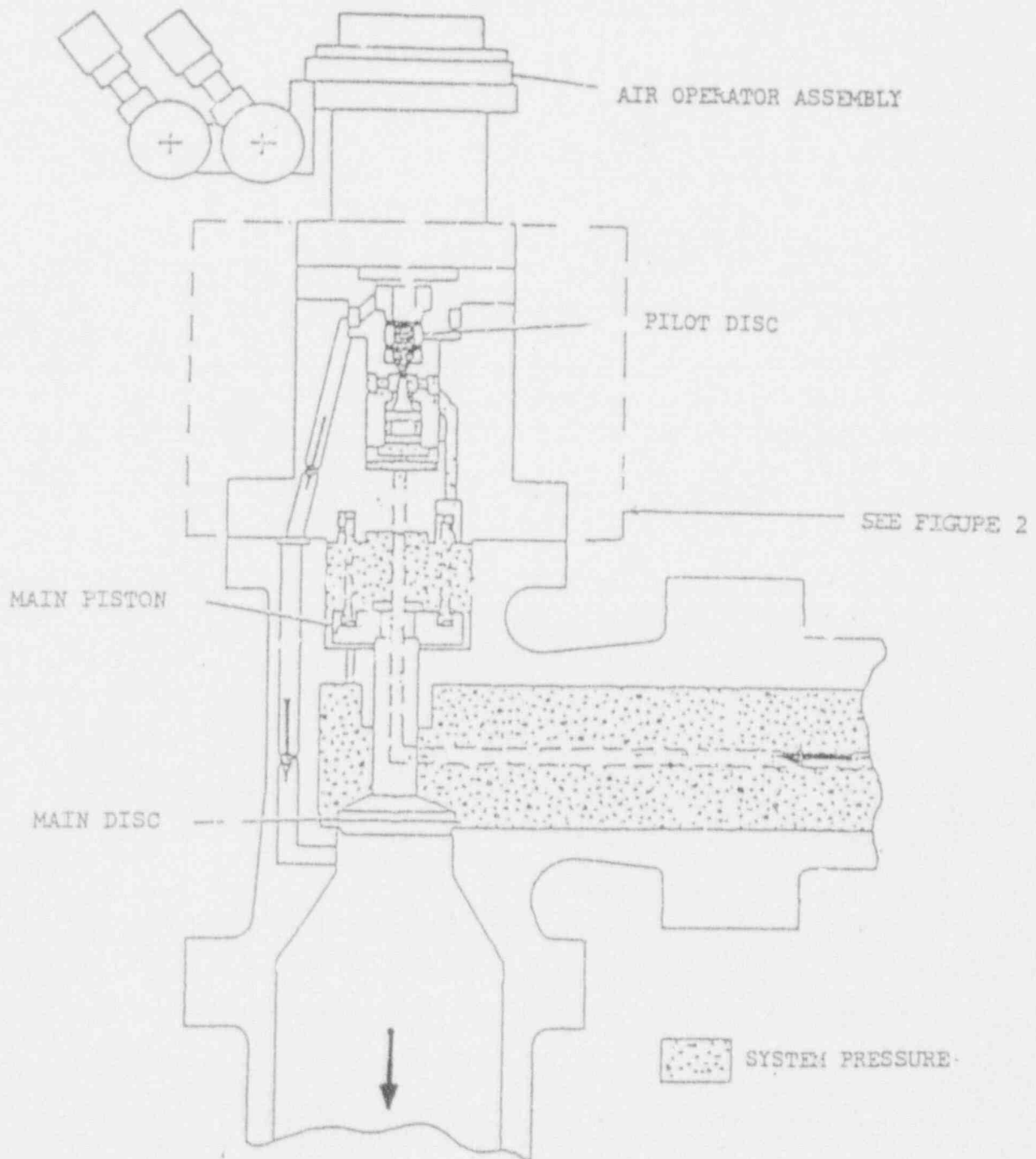
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FIGURE 1



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FIGURE 2

