10 CFR 50.73

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION

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May 13, 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 and Special Report concerns a Unit 1 reactor SCRAM (a Reactor Protection System actuation), resulting from a closure of the Main Turbine Stop Valves, an actuation of the Primary Containment and Reactor Vessel Isolation Control System (an Engineered Safety Feature), and a manual initiation of the Reactor Core Isolation Cooling System. The cause of this event was due to a loose copper link in an electrical switch cabinet.

Reference:	Docket No. 50-352
Report Number:	1-91-009
Revision Number:	00
Event Date:	April 12, 1991
Report Date:	May 13, 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)(iv). This LER is also being submitted as a Special Report pursuant to Technical Specifications Reporting Requirement 6.9.2, as required by Technical Specifications Action 3.7.3.b.

Very truly yours,

J Doering

DMS:cah

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

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

LICENSEE EVENT REPORT (LER)

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TITLE (4) Reactor SCRAM resulting from a Spurious Loss of the DC Electrical Power Supply to the Main Turbine Electrohydralic Control System.

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)
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0 4	1 2	9 1	9 1	0 0 9	0 0	0 5	1 3	9 1		0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	1 0 0	[ ] 20.402(b)	[ ] 20.405(c)	[X] 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)	Special Report
		[ ] 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

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 April 12, 1991, a Unit 1 reactor SCRAM and a partial Group VIC Primary Containment and Reactor Vessel Isolation Control System actuation occurred following a Main Turbine trip. The Main Turbine trip resulted from a spurious loss of the 125 volt DC electrical power supply to the Electrohydraulic Control (EHC) system. Following the SCRAM, all control rods fully inserted, reactor pressure increased to 1103 psig, and level decreased to approximately -20 inches instrument level. The Reactor Core Isolation Cooling (RCIC) system was manually operated to maintain reactor level when a loss of normal feedwater injection occurred. All systems operated as designed except for the normal feedwater system in which operator interaction occurred. Operations personnel successfully controlled the plant shutdown using the appropriate plant procedures. The cause of the loss of the 125 volt DC electrical power supply to the EHC system was due to a loose copper link inside a switch cabinet. The cause of the loss of normal feedwater injection was due to a personnel error resulting from a misinterpretation of the indications for the Condensate Filter Demineralizer system. The copper links were replaced with hard wire connectors, and an evaluation of other similar switch cabinets with copper links will be performed. Operator training will be implemented to address this specific loss of normal feedwater injection incident.

Unit Conditions Prior to the Event:

Unit 1 Reactor was in Operational Condition 1 (Power Operation) operating at 100% Power Level.

The 20 Startup Bus, one of two offsite power sources, was deenergized and blocked out of service in accordance with the operating procedures for the 13.2 KV system to support work related to the Unit 2 Refueling Outage. The (Unit 1) 12 Station Auxiliary Power Bus feeder breaker from the 20 Startup Bus (20-12 breaker) was blocked out of service as part of the 20 Startup Bus being deenergized, and the 13.2 KV fast transfer select switch was in the 20-12 position. In the event of a Unit 1 Main Turbine trip, the 12 Station Auxiliary Power Bus would be transferred to the deenergized 20 Startup Bus where it could then be isolated, transferred to the 10 Startup Bus (the other offsite power source), and then reloaded. The (Unit 1) 11 Station Auxiliary Power Bus and feeder breaker from the 10 Startup Bus was unaffected by the 20 Startup Bus being deenergized.

Description of the Event:

On April 12, 1991, at 1343 hours, a Unit 1 reactor SCRAM occurred as a result of a Reactor Protection System (RPS) (EIIS:JD) actuation due to a Main Turbine Stop Valve (TSV) (EIIS:PCV) closure, following a Main Turbine (EIIS:TRB) trip. The Main Turbine trip resulted from a spurious loss of the 125 volt DC electrical power supply to the Main Turbine Electrohydraulic Control (EHC) System.

All control rods (EIIS:AA) fully inserted as designed. Following the Main Turbine trip and reactor SCRAM, Reactor Pressure Vessel (RPV) (EIIS:RPV) pressure increased to 1103 psig and RPV level decreased to approximately minus twenty (-20) inches instrument level (147 inches above the top of active fuel) as an expected result of the TSV fast closure. All nine Main Turbine Bypass Valves opened, and three Main Steam Relief Valves (MSRV), H, K, and N, lifted to automatically reduce reactor pressure. The MSRVs were open for a duration of eleven seconds while the nine bypass valves remained open for twenty-five seconds. MCR personnel initiated Transient Response Implementation Plan (TRIP) Procedure, T-101, "RPV Control," to control reactor pressure and level.

During the event, the End-Of-Cycle (EOC)-Recirculation Pump (EIIS:P) Trip (RPT) breakers (EIIS:BKR) tripped, as designed, due to the turbine trip from greater than 30 percent reactor power. This resulted in a trip of both of the Unit 1 recirculation pumps. When the RPV exceeded 1093 psig, redundant control rod insertion and RPT breaker trip signals were also received, as designed, from the Alternate Rod Insertion (ARI), and Anticipated Transient Without SCRAM (ATWS) - RPT logics of the Redundant Reactivity Control System (RRCS).

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

Additionally, the High Pressure Coolant Injection (HPCI) (EIIS:BJ) system, the Reactor Core Isolation Cooling (RCIC) (EIIS:BN) system, and the Group VIC (Primary Containment Sampling/Recombiner) Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) (EIIS:JM) valves received spurious partial initiation signals due to momentary spiking of reactor vessel level instrumentation to below the initiation setpoint of -38 inches instrument level (low-low level). The duration of the spikes was approximately 100 milliseconds and was the direct result of the closure of the main turbine stop valves from full power. Both the HPCI and RCIC systems received initiation signals, however, the spurious signal did not exist long enough to cause either system to fully initiate and automatically inject into the reactor vessel. As a result of the spurious initiation signal, a partial Group VIC PCRVICS actuation occurred, an Engineered Safety Feature (ESF), causing the following Solenoid Valves (SV) for the Primary Containment Hydrogen/Oxygen (H2/O2) Combustible Gas Analyzers (CGA) and a Primary Containment Leak Detection System (PCLDS) to close:

SV-57-132, SV-57-133, SV-57-134, SV-57-150, SV-57-181, SV-57-183, SV-57-190, SV-26-190B, and SV-26-190D.

Immediately following the Main Turbine trip, the 11 and 12 Station Auxiliary Buses automatically transferred to their selected Startup Buses, although all loads fed from the 12 Aux Bus were lost since the 20-12 feeder breaker was blocked out of service. This resulted in a loss of electrical power to various Non-Safeguard loads including the Condensate Filter Demineralizer (CFD) Bypass Valve (HV-16-105), all of the CFD Holding Pumps, and the local CFD control panel. The 'B' and 'C' Reactor Feedwater Pumps (RFP) were manually tripped by MCR personnel following the SCRAM to prevent over filling of the reactor vessel; however, three (3) minutes following the reactor SCRAM, the 'A' RFP tripped on low suction pressure. With the loss of normal feedwater flow, MCR personnel aligned the RCIC system to inject into the reactor vessel for level control until normal feedwater injection could be re-established. The low suction pressure trip occurred after a Radwaste (RW) operator manually ran back the position controllers for the CFD outlet valves to the zero percent open position. This action resulted in the isolation of all eight (8) CFDs, and hence isolation of the suction flowpath for the RFPs, resulting in a loss of normal feedwater injection. The RW operator concluded that all eight (8) CFDs were isolated as evidenced by flow indication on the local CFD control panel. The low flow indication was due to the low feedwater flow demand at the time. After power was restored to the 12 Station Auxiliary Bus, the individual load center breakers were closed, which restored power to the CFD Bypass Valve. The bypass valve automatically opened on high CFD differential pressure and provided suction pressure to the 'A' RFP. After sufficient suction line pressure was obtained, the 'A' RFP Turbine was restarted and normal feedwater injection to the reactor vessel was re-established. At 1408 hours on April 12, 1991, the RCIC injection was terminated, and the RCIC system was realigned for automatic operation.

Operations personnel had manually operated the RCIC system to maintain RPV level and remove residual heat for 17 minutes. The reactor parameters prior to the injection were as follows:

Reactor Thermal Power:	0 MWth
Reactor Vessel Dome Pressure:	900 PSIG
Core Moderator Temperatures:	534 degrees F
Core Flow:	15 Mlbm/hr
Feedwater Flow	0 Mlbm/hr
Feedwater Temperature:	400 degrees F

This injection represents the nineteenth Unit 1 RCIC system actuation cycle to date.

At 1443 hours, the Group VIC PCRVICS isolations were reset in accordance with General Plant (GP) Procedure, GP-8, "Primary and Secondary Containment Isolation Verification and Reset." This allowed the Primary Containment Sampling Lines, the CGAs, and the PCLDS to be returned to service. At 1528 hours, operations personnel reset the Unit 1 SCRAM in accordance with procedure GP-11, "SCRAM Reset," and then initiated procedure GP-3, "Normal Plant Shutdown," to begin normal depressurization and cooldown of the reactor.

After recovery from the SCRAM and implementation of the necessary corrective actions to prevent recurrence, the MCR operators restarted Unit 1 at 1919 hours on April 16, 1991. The plant was shutdown for approximately four days as a result of this transient.

A four hour notification was made to the NRC at 1712 hours on April 12, 1991, in accordance with the requirements of 10CFR50.72(b)(2)(ii) since this event resulted in automatic actuations of the RPS and an ESF. This LER is being submitted in accordance with the requirements of 10CFR50.73(a)(2)(iv).

Additionally, Technical Specifications (TS) 3/4.7.3, "Reactor Core Isolation Cooling System," Limiting Condition for Operation (LCO) Action 3.7.3.b states, "In the event the RCIC system is actuated and injects water in to the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date." TS Section 6.9.2, "Special Reports," states, "Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report." Therefore, this LER is also being submitted as a Special Report pursuant to TS Section 6.9.2 as required by TS LCO Action 3.7.3.b.

Analysis of the Event

The TSV closure caused a reactor SCRAM as designed and all control rods fully inserted. There was no release of radioactive materials to the environment as a result of this event.

A Main Turbine trip from high power is the most severe transient that the plant is anticipated to undergo, from an instrument response viewpoint. The level oscillations and resulting instrumentation spiking experienced during this trip were consistent with previous testing on both Units 1 and 2, and all instrumentation responded as designed. All systems operated as designed except for the normal feedwater system in which operator interaction occurred. MCR Operations personnel successfully controlled the plant shutdown using the appropriate plant procedures. The spurious initiation signals received by the HPCI and RCIC systems were a result of the level instrumentation spikes. Both the HPCI and RCIC systems remained operable throughout the event. Had a valid initiation signal occurred at any time during this event, the initiation signal would have existed long enough for either system to perform its intended safety function.

In response to the loss of feedwater injection capability, the MCR operators followed TRIP procedure T-101 and manually initiated the RCIC system to maintain normal reactor water level. The HPCI system was also operable in the event that additional high pressure injection was necessary. Additionally, the Automatic Depressurization System and all of the low pressure Emergency Core Cooling Systems were operable to provide adequate core cooling in the event that high pressure injection to the reactor could not be re-established. The MCR operators would have implemented the necessary TRIP procedures to ensure safe shutdown of the plant without normal feedwater injection.

The maximum reactor pressure reached during this event was 1103 psig, well below the TS Safety Limit of 1250 psig. All fourteen MSRVs were operable and would have lifted as necessary to maintain reactor pressure below the Safety Limit. An evaluation of the three MSRVs that lifted at pressures slightly below their setpoints concluded the this was consistent with previous testing on both Units 1 and 2. Due to the dynamic condition of the transient (i.e., induced vibration and MSRV placement), the three MSRVs lifted prior to reaching their set pressure, which is conservative.

The following automatic RPS actuation signals also occurred that would have actuated a SCRAM if the TSV closure signal had failed.

- o Reactor Water Level < +12.5 inches
- o Reactor pressure > 1037 psig
- o Turbine Control Valve fast closure

The RRCS ARI response provided a redundant method of control rod insertion and would have successfully shutdown the reactor if required. The RRCS ATWS-RPT logic provided a redundant trip signal if the EOC-RPT logic failed to trip the recirculation pump breakers.

As a result of the partial Group VIC PCRVICS actuation, both Primary Containment H2/O2 CGAs were isolated and recirculating their gas flows for approximately one hour during this event. No abnormal H2/O2 concentrations were identified prior to or following this event. Alternate sample pathways were available for manual alignment in the event the Primary Containment H2/O2 CGAs were required to monitor drywell and suppression pool H2/O2 concentrations. Also, the actions to bypass the isolation signal and reopen the isolation valves are directed when required by TRIP procedure T-102, "Primary Containment Control," and therefore, system operation could have been manually restored by the operators if the isolation signal could not have been reset. Additionally, as a result of the partial Group VIC PCRVICS actuation, one of the four Primary Containment Leak Detection Systems isolated for approximately one hour during this event. The three redundant systems were in operation during this event and no abnormal leakage was identified prior to, during, or following this event.

Cause of the Event:

The cause of the loss of the 125 volt DC electrical power supply to the EHC system, which initiated the Main Turbine trip, was due to a loosely fitting copper link found in the General Electric (GE) Company supplied DC power supply cabinet (See Diagram 1). The copper links were installed during original construction of the Unit 1 EHC system. The EHC system 125 volt DC electrical power supply lines have the capability to be double fused. There was no need or requirement to double fuse these DC power lines, and hence, copper links were installed in place of the secondary fuses. Troubleshooting by the EHC System Engineer (SE) following the Unit 1 SCRAM actuation revealed that one of the copper links was loose, and showed signs of pitting, which is indicative of arcing in the circuit. Additionally, loss of the DC power supply to the EHC system through this loose copper link was repeatable during troubleshooting activities when vibration was initiated in the area of the switch cabinet. Loss of 125 volt DC power to the EHC system will initiate a Main Turbine trip signal if the undervoltage condition is present for 0.2 seconds or longer per design. During troubleshooting, the EHC SE determined that the actual undervoltage condition existed for 0.5 seconds. Therefore, the Main Turbine trip initiated as designed on the loss of 125 volt DC power to the EHC system. This event is the first occurrence of a loose copper link or fuse and it has been concluded to be an isolated occurrence.

The cause of the spurious partial initiation signals to the HPCI, RCIC, and the Group VIC PCRVICS systems was the momentary spiking of the RPV water level instrumentation. This spiking of the level instrumentation was the result of the level and pressure perturbations that occurred immediately following the Main Turbine trip and reactor SCRAM actuation. The level spikes were too short in duration to fully initiate the HPCI and RCIC systems; however, partial Group VIC PCRVICS actuations did occur.

The cause of the loss of suction flow to the RFPs was due to a personnel error resulting from a misinterpretation of the indications for the CFD system. The RW operator manually ran back the position controllers for the CFD outlet valves because the flow indication on the local CFD control panel led him to conclude that the CFD outlet valves had closed when power was lost to the CFD control panel. The low flow indication was due to the low feedwater flow demand at the time. Contrary to this indication, the CFD outlet valves fail "as is", which were in the open position. Additionally, there are no procedures or operator training which address the specific actions that are to be implemented on a loss of power/indications to the CFD control panel.

Corrective Actions:

1. The four (4) copper links installed in the 125 volt DC electrical power supply switch cabinets for the Unit 1 and Unit 2 EHC systems were removed and replaced by lugged hardwire connectors to ensure reliability of both EHC systems.

2. An evaluation of known copper links installed in similar electrical switch cabinets for other Unit 1 and Unit 2 systems as listed in the Limerick Generating Station (LGS) fuse index will be performed. This evaluation is expected to be completed by December 1, 1991, and appropriate actions will be implemented as necessary.

3. A Shift Training Bulletin discussing this event and the loss of the normal feedwater injection incident has been issued to all plant operators to avoid inappropriate actions on a loss of power/indication to the CFD control panel.

4. The Non-Licensed and Licensed Operator Continuing Training Programs will address this specific incident concerning the loss of suction flow to the RFPs, and will address the actions operators are to take on a loss of power/indications to the CFD control panel. This specific incident and the operator actions are expected to be addressed to all Non-Licensed Operators by September 17, 1991, and to all Licensed Operators by December 16, 1991.

Previous Similar Occurrences:

No previous SCRAMs or Main Turbine trips have occurred on either Unit at LGS due to the cause of this event.

Tracking Codes: B17 Deficient Equipment

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

DIAGRAM 1