NRC Form 366 (9-83)

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Limerick Generating Station - Unit 1	0 5 0 0 0 3 5 2	1 OF 0 5

TITLE (4) Reactor Scram Resulting From a Main Turbine Trip Due to Low Electro-Hydraulic Control Pressure

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 9	1 9	8 7	8 7	0 4 8	0 1	0 1	0 5	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 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)	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)	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
Charles A. Mengers, Senior Engineer, Licensing Section	AREA CODE 2 1 5 8 4 1 - 5 1 8 4

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

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)

Abstract: 87-048 Rev. 1

On September 19, 1987 at 0910 hours, the Reactor Protection System initiated a reactor scram from 90% power and a recirculation pump trip, following a Main Turbine trip on low Electro-Hydraulic Control (EHC) system oil pressure. Following the turbine trip, operation of the turbine bypass valves was maintained until their EHC accumulator pressure bled down. Reactor pressure reached a peak value of 1093 psig and reactor vessel water level reached a minimum level of minus 2 inches during the event. There were no adverse consequences and there was no release of radioactive material resulting from this event. The root cause of the event was the failure of a tubing socket weld in the EHC fluid actuating supply (FAS) line to the #3 Main Turbine Control Valve (MTCV). The section of EHC pipe containing the failed weld was removed for inspection and a new section of pipe was welded into the line. Analysis of the measured piping movement and vibration levels indicates that a properly bonded weld would not have failed as a result of the vibrations present. As such, this event is considered an isolated incident. Adjustments were made to the Steam Line Resonance Compensator (SLRC) and a second SLRC was installed in series with the first to reduce the control signal oscillations. Full power operation was achieved on November 21, 1987.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Unit Conditions Prior to the Event:

Operating Mode 1 (Power Operation)

Reactor Power - 90%

Description of the Event:

On September 19, 1987 at 0910 hours, the Reactor Protection System (RPS) initiated an unplanned Reactor scram from 90% power and a recirculation pump trip, as a result of a turbine stop valve fast closure signal following a Main Turbine Trip. The Main Turbine tripped due to low Electro-Hydraulic Control (EHC) system oil pressure following a failure of a tubing socket weld which joins the EHC fluid actuating supply (FAS) line to the #3 Main Turbine Control Valve (MTCV). Following the turbine trip, the turbine bypass valves opened as designed and were held open until their accumulator pressure bled down. An unusual event was declared at 0915 hours. Reactor pressure reached a peak value of 1093 psig and reactor vessel water level reached a minimum level of minus 2 inches following the scram. Nuclear Steam Supply Shutoff System (NSSSS) Groups IIA (Shutdown Cooling) and IIB (Residual Heat Removal Sample and Drain Lines) isolation signals were received on Reactor low level 3 (12.5 inches) signal; however, the affected valves did not reposition since a Group IIA isolation signal had already been received on high reactor pressure (75 psig), and the valves affected by the Group IIB signal are normally closed. At the time of the event, the operators were reducing reactor power in accordance with General Plant Procedure, GP-3 (Normal Plant Shutdown), following the identification of an EHC oil leak on the FAS line; however, they were unable to reduce reactor power to the capacity of the turbine bypass valves before the weld failed.

Following the scram, the operators started a second steam jet air ejector, opened the main steam line drains, and manually initiated the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Coolant (RCIC) systems in the full flow test mode to control reactor pressure and level, in accordance with Trip Procedure, T-101. At 0959 hours, the scram was reset in accordance with General Plant Procedure GP-11 (Post Scram Control Rod Position Determination and Recording). HPCI and RCIC systems were secured at 0958 hours and 1102 hours, respectively, and the notice of an unusual event was terminated at 1443 hours. On

September 20, 1987 at 1220 hours, the reactor was critical again and at 2330 hours the generator was resynchronized to the grid.

The EIIS codes for the effected systems are TG, BJ, BN for the EHC, HPCI and RCIC systems respectively.

Consequences of the Event:

There were no adverse consequences resulting from this event. There was no release of radioactive material as a result of this event. The Main Turbine tripped on low EHC system oil pressure and the reactor scrammed as designed in response to the turbine stop valve fast closure signal.

Reactor pressure was maintained below the Safety Relief Valve (SRV) setpoints; however, if reactor pressure had increased above 1130 psig, the SRVs would have operated as designed to maintain reactor pressure below the reactor coolant pressure boundary transient pressure limit (1375 psig). This event is evaluated in the Final Safety Analysis Report Section 15.2 as a Turbine Trip with Failure of the Bypass system. The analysis indicates that a scram from 100% power with the bypass system inoperable would raise reactor pressure to a peak value of 1223 psig.

If reactor vessel level had decreased below minus 38 inches, HPCI and RCIC would have automatically initiated and injected to the reactor.

Cause of the Event:

The root cause of the event was failure of a tubing socket weld in the EHC FAS line to the #3 Main Turbine Control Valve. Control System operation of the Main Turbine Control Valves produced oscillations along the EHC tubing which induced failure of a defective weld. Loss of EHC oil pressure initiated a Main Turbine trip, closing all control valves, stop valves and combined intermediate valves on the Main Steam Line. Reactor Protection System initiated a full reactor scram when the turbine stop valves were less than 95% full open. At the time of the event, the operators were reducing reactor power in accordance with General Plant Procedure GP-3 (Normal Plant Shutdown), following the identification of an EHC oil leak on the FAS line;

however, they were unable to reduce reactor power to the capacity of the turbine bypass valves before the weld failed.

Corrective Actions:

Following the scram and closure of the turbine bypass valves, the operators placed a second steam jet air ejector in service, opened the main steam line drains, and manually initiated HPCI and RCIC systems in full flow test mode to control reactor pressure and reactor vessel water level, in accordance Trip Procedure (T101). At 0959 hours, the scram was reset in accordance with GP 11 (Post Scram Control Rod Position Determination and Recording) HPCI and RCIC systems were secured at 0958 hours and 1102 hours, respectively. The notice of unusual event was terminated at 1443 hours.

The section of EHC tubing at the failed weld site was removed for inspection and replaced with a section of new tubing. The EHC reservoir was filled, the system vented, and the generator was returned to the grid at 2330 hours on September 20, 1987.

Visual and dye penetrant inspections of other welds in the EHC lines revealed no evidence of cracking. As such, the failure of the weld on the EHC supply line to the #3 Main Turbine Control Valve is considered an isolated event resulting from a poor weld. It has been determined that a properly bonded weld would not have failed as a result of the vibrations present.

Actions Taken to Prevent Recurrence:

Subsequent investigative testing, revealed that the EHC control system was responding to a steam line resonance signal. The oscillating signal which has been present since initial startup is now being amplified by higher control valve gains (above approximately 85% power). The high gains were introduced with the conversion of the turbine control valves from full arc admission to partial arc admission during the recent refuel outage. Even though evaluations indicate that the pipe vibration is within acceptable limits for good quality welds, actions which will reduce or filter out the steam line resonance excitation, such as adjusting the steam line resonance compensator, are being evaluated. Special procedures and tests have been developed for the adjustment of these devices and verification of the effectiveness of the change. Power operation was restricted to 83% until the adjustments were complete.

On October 15, 1987 adjustments were made to the Steam Line Resonance Compensator in accordance with special test procedure SP-HF-008, "EHC Pressure Control System Steam Line Resonance Compensator Adjustment". Subsequent testing at low power indicated that the adjustment was successful at reduced power; however, on October 19, 1987 during power ascension, it was determined that the adjustments made did not sufficiently limit the control signal oscillations. On November 20, 1987 a temporary circuit alteration (TCA) was performed to install a second Steam Line Resonance Compensator in series with the first. This reduced the control signal oscillations experienced at increased power levels. This TCA will be made into a permanent modification during the next outage of sufficient length. Full power operation was achieved on November 21, 1987.

Previous Similar Occurrences:

Cause Code: B (construction/Installation)

There have been no similar events involving the EHC system to date.

PHILADELPHIA ELECTRIC COMPANY
2301 MARKET STREET
P.O. BOX 8699
PHILADELPHIA, PA. 19101
(215) 841-4000

January 5, 1988

Docket No. 50-352

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

SUBJECT: Licensee Event Report - Revised
Limerick Generating Station - Unit 1

This LER concerns a Reactor Scram resulting from a Main Turbine trip due to low Electro-Hydraulic Control pressure.

Reference:	Docket No. 50-352
Report Number:	87-048
Revision Number:	01
Event Date:	September 19, 1987
Report Date:	January 5, 1988
Facility:	Limerick Generating Station P.O. Box A, Sanatoga, PA 19464

This revised LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(iv) and to update the corrective actions which have been taken to prevent future similar occurrences. The changes made to this revision are indicated by bars in the right hand margin.

Very truly yours,

R. H. Logue
Assistant to the Manager
Nuclear Support Department

cc: W. T. Russell, Administrator, Region I, USNRC
E. M. Kelly, Senior Resident Site Inspector

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