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)
LASALLE COUNTY STATION	0 5 0 0 0 3 7 3	1 OF 0 4

TITLE (4)
RX SCRAM/LOSS OF MAIN CONDENSER

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 1	1 6	8 4	8 4	0 0 5	0 0	0 2	1 5	8 4	NA	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 8 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)	XX	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
Baron S. Westphal, ext. 247	AREA CODE 8 1 5 — 3 5 7 - 6 7 6 1

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
X	S G	S E A L	Z 9 9 9	N					
X	E S	B L L	T 1 1 7	N					

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	XX NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On 1-16-84 at 1522 the Reactor Protection System initiated a reactor scram when the turbine generator tripped due to low vacuum. The reactor was manually isolated from the condenser because vacuum deteriorated very rapidly. Pressure increased to about 1050psig at which point a safety relief valve (SRV) lifted in the pressure relief mode. Two subsequent manual actuations of an SRV were required before RCIC was capable of controlling pressure. Reactor water level decreased to about +2" and was initially regained by feedwater to +54.5" which is the hi level trip for the feed pump. Water inventory was subsequently maintained by RCIC. No ECCS or PCIS initiations were required and no maintenance or testing was in progress that made the transient more severe.

The loss of vacuum occurred because the rubber boot seal which forms an expansion joint boundary between the "C" low pressure turbine and the "C" condenser hood ruptured. The boot seal was overheated and subsequently failed when an extraction steam expansion joint located inside the condenser failed thereby causing steam to impinge on the metal shield protecting the boot seal. The boot seal was replaced and the extraction steam line was capped until a replacement expansion joint could be obtained.

IE22

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(9-83)

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)

I. EVENT DESCRIPTION:

At 1522 on 1-16-84 with Unit One reactor at 80% power, the Turbine Generator (TA) tripped on low vacuum. The Turbine Stop Valves (TA, TSV) close in response to a turbine trip which in turn caused a reactor scram upon receipt of the Reactor Protection System (JC, RPS) signal indicating the Turbine Stop Valves were less than or equal to 95% open with reactor power greater than 30%. Condenser vacuum deteriorated rapidly so the Main Steam Isolation Valves, (SB, MSIV) were closed to isolate the reactor from the condenser. A loss of circulating water (NN, CW) also occurred at this time when the CW pumps tripped during the automatic transfer of their power supply from the normal to the alternate source.

Reactor pressure was controlled via Safety Relief Valve (SB, SRV) operation and the Reactor Core Isolation Cooling system (BN, RCIC). SRV E automatically lifted in the pressure relief mode at about 1050 psig and SRV F was manually operated two more times at which point RCIC was operating and able to control reactor pressure.

Reactor water level was controlled with the Feedwater System (SJ, FW) and RCIC. The 1A and 1B Turbine Driven Reactor Feed Pumps (TDRFP) lost their steam supply when the MSIV's were closed so the Motor Driven Reactor Feed Pump (MDRFP) was manually started to provide makeup. The minimum reactor water level was about +2" and increased to +54.5" at which point the MDRFP tripped on hi water level.

No Primary Containment Isolation System (JM, PCIS) group isolations or Emergency Core Cooling System (JE, ECCS) actuations were required.

II. CAUSE:

A. LOSS OF VACUUM

An inspection of the condenser determined the loss of vacuum occurred because the rubber boot seal on the "C" hood was overheated and then failed when a 16" Extraction Steam (SE, ES) expansion joint, ES15M, ruptured and allowed extraction steam to heat the boot seal. The boot seal acts as an expansion joint between the Low Pressure Turbine and the condenser and is a boundary for condenser vacuum. Pressure indication for the 14C Low Pressure Heater (SM) fed by this ES line was much lower than that of the 14A and 14B Low Pressure Heaters with the same pressure extraction steam from the A and B Low Pressure Turbines. The low pressure was thought to be caused by a hole in one of three expansion joints in this line but couldn't be verified because they were located inside the condenser. This possible situation was evaluated and it was determined that turbine operation could continue. After the condenser inspection above, it was postulated that a turbine trip from 80% reactor power on 1-13-84 may have jarred this extraction line enough to totally rupture the leaking expansion joint. This allowed steam to impinge on the metal shield protecting the rubber seal and subsequently produce an over temperature condition that caused the seal to fail. The 16" expansion joint is manufactured by Temp Flex.

B. LOSS OF CIRCULATING WATER TO CONDENSER

During plant operations two transformers provide electrical power for Unit 1 (U-1). These are the Unit Auxiliary Transformer (EL, UAT) which is fed from the U-1 Generator and the System Auxilliary Transformer (SAT) which is fed from the Edison electrical grid (off site power). Bus 141X (EA) provides the electrical feeds for 1A and 1C CW pumps and its normal feed is the UAT and its alternate feed is the SAT. When the U-1 Generator trips a low voltage condition to Bus 141X will initiate a bus transfer sequence which trips the UAT feed to 141X and then closes the SAT feed. The CW pumps are synchronous motors and have a relay protection called the "slip guard relay" to prevent large motor currents when the magnetic field is out of phase with respect to the motor shaft. This condition can occur on a bus transfer and is the reason 1A and 1C CW pumps tripped causing a loss of circulating water to the condenser. Attempts to immediately restart the CW pumps were unsuccessful.

U-1 has 3 CW pumps and 2 of these are required for normal operations. Usually either 1A or 1C and 1B CW pumps are running. This lineup helps prevent a loss of circulating water on a bus transfer because the normal feed for 1B CW pump is ultimately the SAT. Prior to this event, the 1B CW pump was inoperable and consequently 1A and 1C CW pumps were running.

III. PROBABALE CONSEQUENCES OF THE OCCURRENCE:

There was no maintenance or testing in progress that inhibited proper performance of systems required during the event. SRV actuation and RCIC operation removed decay heat from the reactor vessel to the suppression pool after the reactor was isolated from the condenser. The HPCS system (BG) was also available as were all the low pressure ECCS systems. RHR (BO) was placed in operation to remove heat from the suppression pool.

The large rupture in the boot seal caused a very rapid loss of vacuum and the loss of circ. water probably did little to increase this rate. Attempts to reestablish a vacuum after CW was regained failed to achieve even greater than 7" of Hg backpressure which would have permitted use of the condenser as a heat sink for the reactor.

IV. CORRECTIVE ACTION:

A. The "C" condenser hood boot seal was replaced per Work Request L31992.

B. Expansion joint ES15M was not replaced because a replacement was not available. Modification M-1-1-84-009 capped this extraction steam line and provided drainage for moisture collected in the line. This modification will be removed and a new expansion joint installed during an outage of sufficient length following receipt of a replacement. The remaining expansion joints in this line and those in the similar extraction steam lines from the A and B L.P. Turbines were inspected for damage - none was found. (AIR 1-84-67018).

C. Troubleshooting the 1A and 1C CW pumps discovered the timing in the field exciter circuit was permitting the "slip guard relay" to trip the pump before the motor had a chance to achieve the proper speed. The timing was reset for proper sequencing.

V. PREVIOUS EVENTS:

None.

VI. NAME AND TELEPHONE NUMBER OF PREPARER:

Baron S. Westphal - 815/357-6761, extension 247.

Commonwealth Edison
LaSalle County Nuclear Station
Rural Route #1, Box 220
Marseilles, Illinois 61341
Telephone 815/357-6761

February 15, 1984

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

Dear Sir:

Reportable Occurrence Report #84-005-00, Docket #050-373 is being submitted to your office in accordance with 10 CFR 50.73(d).

G. J. Diederich
G. J. Diederich
Superintendent
LaSalle County Station

GJD/GW/rg

Enclosure

cc: NRC, Regional Director
INPO-Records Center
File/NRC

IE22
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