NRC Form 366
(9-83)

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
LaSalle County Station Unit 1	0 5 0 0 0 3 7 3	1 OF 0 6

TITLE (4)

Manual Scram Due to Loss of Circulating Water

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 5	3 1	8 5	8 5	– 0 4 5	– 0 0	0 6	2 6	8 5	Unit 2	0 5 0 0 0 3 7 4
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 6 4

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 (AREA CODE)
Paul S. Watford, extension 323	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
B	K E	I S V	P 3 4 0	N	X	K E	E X J	P 3 4 0	N
B	K E	M O	L 2 0 0	N					

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 31, 1985, at 1822 hours, the 1B Circulating Water (CW) pump tripped. At 1829 hours it was discovered that water was rapidly flowing from a rubber expansion joint between the 1B CW pump discharge and its discharge valve. A load drop was initiated at 1835 hours. At 1945 hours, the unit was manually scrammed with Unit 1 at 64% power in anticipation of loss of the CW system. Heat loads in Unit 1 and 2 were reduced. Cold Shutdown of the reactor was initiated with a combination of RCIC and the SRV's. At 2145 hours, Reactor Water Cleanup isolated on high differential flow. At 2200 hours, LGA-3, Containment Control, was entered due to high temperatures in the drywell and Suppression Pool.

At 2225 hours, an Unusual Event was declared and notifications were made per appropriate procedures. This was terminated at 1027 the next day when Unit 1 reached Cold Shutdown. Water level in the Lake Screen House reached lake level at 0115 hours on June 1, 1985, for a total of 675,000 gallons of water.

The primary cause of the event was fatigue failure of the 1B CW pump discharge valve gear operator mounting bolts.

Station procedures and plant design are being reviewed for long-term corrective action.

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

I. EVENT DESCRIPTION

On May 31, 1985, at 1945 hours, with Unit 1 at 64% power, the reactor was manually scrammed (JC) due to a failed 1B Circulating Water Pump (KE) Discharge Valve, 1CW006B.

The following sequence of events occurred prior to the manual scram. Unit 1 was at 85% power with the 1A and 1B Circulating Water pumps in operation and the 1C pump in Standby. Unit 2 was in Cold Shutdown for a scheduled surveillance outage.

At 1822 hours the 1B Circulating Water pump tripped. Unsuccessful start attempts were made on the Standby 1C pump and the pump that tripped (1B). This difficulty in starting CW pumps is not unusual at LaSalle Station. The pumps have several start permissives such as: a) discharge valve full closed, b) gland water pressure available, and c) exciter interlocks which contribute to this phenomenon. On the second attempt at 1825 hours, the 1C pump started.

At 1825 hours, the Unit 1 Shift Foreman (SRO) and a Non-Licensed Operator went to the Lake Screen House (MD) (where the Circulating Water pumps are located) to determine the cause of the pump trip.

At 1829 hours, a Security Officer on a routine inspection of the Lake Screen House reported a major water leak in the basement. Moments later, the Shift Foreman and the Operator arrived. It was discovered that water was rapidly flowing from a rubber expansion joint between the 1B pump discharge and its discharge valve. This has been conservatively estimated at 2000 gallons per minute. Based on the Shift Foreman's evaluation, a load drop of 200 MWe/hr was initiated on Unit 1 at 1835 hours and the 1B Service Water (KG) pump was shutdown.

At 1945 hours, the unit was manually scrammed in anticipation of a loss of Circulating Water. The location of Circulating Water pump exciter panels required that the Circulating Water pumps be shut down prior to the panels being flooded. The Unit 2 Circulating Water pumps were shut down. Unit 1 and Unit 2 heat loads were reduced to maintain cooling to the most important plant systems, primarily the air systems (LD).

At 1946 hours, the 1C Circulating Water pump was shut down and at 1954 hours, the 1A Circulating Water pump was shutdown.

From 2030 to 2045 hours, the 2A and Common Service Water pumps tripped on neutral overcurrent and the 1A pump was shut down. Equipment in the Lake Screen House was de-energized to prevent water damage.

Service water provides cooling to non-safe shutdown equipment including the following: Turbine Lube Oil Cooling (TD), Primary Containment Chill Water (KM), RBCCW (CC), TBCCW (KB), and the process computer (ID).

I. EVENT DESCRIPTION (Continued)

Cold Shutdown of the reactor vessel was initiated with a combination of the Reactor Core Isolation Cooling System (RCIC, BN) and the cycling of the main steam Safety Relief Valves (SRV, SB).

At 2057 hours, the process computer (ID) was shut off due to a loss of air conditioning. SPDS became not functional when the computer was shut down.

At 2145 hours, Reactor Water Cleanup (CE) isolated on high differential flow created by the changing reactor pressure caused by the cycling of main steam safety valves at 2130 and 2135 hours.

At 2200 hours emergency procedure LGA-03, Containment Control, was entered due to high temperatures in the drywell and Suppression Pool (NH). This was a result of manually depressurizing the reactor with the SRV's to reach Cold Shutdown. The Steam Condensing Mode of RHR (BO) was not utilized because plant instrument air was at risk when Service Water was lost.

At 2225 hours, an Unusual Event was declared under Emergency Action Level (EAL) #21 and notifications were made per appropriate procedures.

Water level in the Lake Screen House basement reached lake level of 698' at 0115 hours on June 1, 1985. This was approximately 675,000 gallons of water.

GSEP was terminated at 1027 hours on June 1, 1985, after Unit 1 reached Cold Shutdown.

II. CAUSE

The primary cause of the event was fatigue failure of the 1B Circulating Pump Discharge Valve gear operator mounting bolts.

After the valve bolt failure, the valve disc rapidly rotated towards the closed position, rotating in the reverse-to-normal direction. As determined by CECo Engineering Department SNED - R&DE, a closure rate of 5 - 10° rotation in 20-30 milliseconds created an extremely rapid transient hydraulic pressure spike, peaking in excess of 50 psig. This was sufficient to blow the expansion joint out of its retaining bars. This occurred while the 1B Circulating Water pump was running. The rapid valve closure caused the 1B CW pump to become deadheaded. The motor load suddenly increased and under the influence of excess shaft torque, the motor speed decreased, causing the rotor and stator fields to become out of synchronism and slip which resulted in a "slip guard relay" trip.

Two root causes precipitated this event. The first was a valve installation error. The fractured gear operator mounting bolts and corresponding bolting from other Circulating Water discharge valves revealed that the applied assembly torque was predominantly less than the valve manufacturer's specification.

II. CAUSE (Continued)

The inadequate bolt tensioning undoubtedly led to loosening of the bolts in service through normal flow induced oscillations of the valve disc in the open position. The loosening permitted relative motion between the gear operator and the valve yoke and led to application of a relatively high amplitude, low frequency, cyclical shear load on the bolts. This cyclical load was clearly involved with the bolt fatigue fracture.

The second root cause was probably a design deficiency. First, there appears to be an error in the assumptions used to structurally size the gear operator and its associated attachment bolting. The operator structural sizing was based on a symmetrical flow velocity distribution model whereas the LaSalle application involves an asymmetrical model. An asymmetrical distribution consistently represents a significant increase in butterfly valve operating torque and consequently, an increase in the gear operator structural loading and susceptibility to vibration and flutter.

It is also believed that there may have been insufficient conservative design margin assigned to the gear operator-valve yoke mounting bolts. Bolt tensioning is expected to prevent both axial and rotational movement of the operator relative to the valve yoke. During valve-operator reassembly following this event, it was discovered and verified that despite thorough operator-yoke mating surface preparation and proper torquing of new mounting bolts, manual valve actuation still produced significant operator-yoke relative motion.

The discharge isolation valve (1CW006B) is a 108 inch butterfly valve manufactured by the Henry Pratt Company. The expansion joint with restraining bars, located two feet upstream of the discharge valve is also manufactured by the Henry Pratt Company. The manual operator, model SMB1-40 was manufactured by Limitorque Company as specified by Henry Pratt Company.

III. PROBABLE CONSEQUENCES OF THE OCCURRENCE

The loss of Service Water caused loss of cooling to the Station air compressors and the primary containment air coolers among other loads. One of two Station air compressors was running hot and subsequently shut down. The Fire Protection system Diesel Fire Pumps (KP) were crosstied to the Service Water system using hoses, to supply sufficient cooling to keep the second compressor operating in order to provide air for the plant Instrument Air system.

Loss of the Turbine Lube Oil Cooling system could result in costly turbine-generator bearing damage due to lack of cooling to the bearings before the turbine-generator could be put on its timing gear. The turbine lube oil coolers were cooled by crosstying the Fire Protection system to the TBCCW system.

Loss of Drywell Pneumatics (LE) could jeopardize long-term operation of the SRV relief function which could inhibit one pathway to Cold Shutdown. Fire Protection Water was also crosstied to RBCCW which cools the Drywell Pneumatics compressor and the operability of the SRV's was maintained.

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III. PROBABLE CONSEQUENCES OF THE OCCURRENCE (Continued)

Loss of cooling to the Primary Containment Cooling system caused drywell temperature to rise to 167°F. This is 32° above the Technical Specification temperature limit of 135°F (Technical Specification 3.6.1.7). The drywell pressure was kept below the isolation pressure of 1.69 psig by venting the containment. A containment air sample had been taken just prior to the water leak occurring which proved that containment venting could be performed. This action prevented isolation of water to the containment coolers. The drywell temperature exceeded Technical Specification limits for approximately eight hours. An analysis by Sargent & Lundy has been performed and it was determined that the temperatures experienced in the drywell had insignificant impact on all the equipment. The Station Heat Recovery system (LV), which uses glycol to air heat exchanger, was tied into the chillers for the Primary Containment Cooling Water system. This was used until Service Water was returned to service.

The RWCU system pumps were expeditiously shutdown following the reactor scram to help reduce the RBCCW heat load. It was minimized to allow RR pump's operation which provided accurate moderator temperature indication. The normal RWCU shutdown procedure closes the inboard and outboard valves. Because the valves were not closed, the pressure transient from the SRV closure was sensed by the RWCU inlet flow sensor and a high differential flow isolation occurred. The sensor taps off the RWCU suction line. The isolation occurred under unique conditions and was of minor concern.

Prior to the event, Unit 1 was in "RUN" at approximately 85% rated power. No work was in progress that had an impact on the Circulating Water system. Unit 2 was in "Cold Shutdown" for a scheduled surveillance outage.

RHR Service Water (BI) was not lost nor effected by this event and was capable of (and did) removing heat from the primary system.

IV. CORRECTIVE ACTION

Stop logs were placed in the intake structure for the 1B Circulating Water pump. Portable pumps were used to pump the water out of the basement from 2025 hours on the day of the event and continuing to 1900 hours on June 1, 1985, when the basement was pumped nearly dry.

All of the Service Water pump motors were sent offsite to be dryed and inspected. The first Service Water pump was returned to service on June 4, 1985, and the last on June 10, 1985, at 1310 hours.

A replacement expansion boot was made from a 120 inch boot obtained from Quad Cities Nuclear Station. This was installed on June 8, 1985.

In the Lake Screen House all circulating water piping expansion joint boots and their retaining rings were inspected for integrity. The same inspection was performed on the inlet waterboxes.

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IV. CORRECTIVE ACTION (Continued)

Inspection of the bolts connecting the operator gear to the valve yoke was performed on the remaining Unit 1 valves and all the Unit 2 discharge valves.

The process computer was returned to service at 1447 hours on June 4, 1985.

A special test has been written to verify that the torque developed by potentially asymmetric Circulating Water flow on the discharge valve does not exceed the allowable torque specified by the vendor. The test will also collect vibration data on the gear operator and valve body during valve operation.

Data sets are to be obtained for two operating conditions on the Unit 2 discharge valve, 2CW006B, in order to envelope the highest anticipated valve operating torques and gear operator loads.

1. Start, run and shutdown of the 2B Circulating Water (CW) pump, and

2. Start and run of another CW pump associated and the 2B CW pump associated with the instrument valve followed by the tripping of the 2B pump, followed by rapid start and run of the remaining CW pump.

The test results will determine what valve design changes, if any, are necessary, and to generate quantitative acceptance standards for such changes. The test results will also serve to further characterize the probabilistic risk of similar valve failure initiating event reoccurences in the future.

An investigation is being conducted to consider moving the CW pump exciters to a location above lake level. In addition, there is a study being performed to investigate possible long-term corrective actions (AIR 373-200-85-00108) which can be taken to prevent loss of all Circulating Water and Service Water systems due to flooding.

AIR 373-200-85-00107 has been written to review and revise as appropriate, all CW and WS shutdown and emergency procedures to insure they are adequate to address affects of station flooding on plant safety systems and systems necessary for safe shutdown of the plant.

V. PREVIOUS OCCURRENCES

None.

VI. NAME AND TELEPHONE NUMBER OF PREPARER

Paul S. Watford, 815/357-6761, extension 323.

Commonwealth Edison
LaSalle County Nuclear Station
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Marseilles, Illinois 61341
Telephone 815/357-6761

June 26, 1985

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

Dear Sir:

Reportable Occurrence Report #85-045-00, Docket #050-373 is being submitted to your office in accordance with 10CFR 50.73.

C E Sargent
for G. J. Diederich
Station Manager
LaSalle County Station

GJD/DRR/kg

Enclosure

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

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