Commonwealth Edison
LaSalle County Nuclear Station
2601 N. 21st. Rd.
Marseilles, Illinois 61341
Telephone 815/357-6761

March 30, 1992

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Dear Sir:

Licensee Event Report #92-003-00, Docket #050-373 is being submitted to your office in accordance with 10CFR50.73(a)(2)(iv).

WRoHuff
G. J. Diederich
for Station Manager
LaSalle County Station

GJD/HTV/mkl

Enclosure

xc: Nuclear Licensing Administrator
NRC Resident Inspector
NRC Region III Administrator
INPO - Records Center
IDNS Resident Inspector

660005

IE22

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

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)

Unit 1 Scram Due To Loss Of Condenser Vacuum

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|5|0|0|0| |
0|3	0|1	9|2	9|2	0|0|3	0|0	0|3	3|0	9|2		0|5|0|0|0| |

OPERATING MODE (9): 1

POWER LEVEL (10): 0\|8\|9

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR
(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)
___ 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	
Harold T. Vinyard, Technical Staff Engineer, Extension 2499	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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	T|C	L|T|	F|1|3|0						
X	T|C	P|I|C	F|1|3|0						

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 March 1, 1992 at approximately 1635 hours, a Unit 1 steam seal header low pressure alarm was received in the control room. At LaSalle, sealing steam for the Main Turbine is produced from a Steam Seal Evaporator (SSE). Upon investigation of the alarm, the Nuclear Station Operator (NSO, licensed RO) observed that steam seal header pressure was downscale but the SSE shell water level was normal.

Prior to establishing an alternate steam seal source, condenser low vacuum alarms annunciated in the control room and seconds later the turbine tripped on low vacuum. The turbine trip caused a Unit 1 reactor scram due to Turbine Stop Valve closure. Recovery actions from the scram were normal and reactor parameters were stabilized.

The cause of this event was the loss of main turbine sealing steam. This resulted in a loss of condenser vacuum, turbine trip, and reactor scram. The root cause of this event appears to be loss of SSE condensate level control due to a partially plugged level instrument sensing line.

The turbine trip in this event initiated the reactor scram as required. Following the scram, reactor pressure was controlled by the Safety/Relief Valves (SRV's) and Turbine Bypass Valves. These events are consistent with the Updated Final Safety Analysis Report (UFSAR) analysis of this event.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

A. CONDITION PRIOR TO EVENT

Unit(s): 1 Event Date: 3/01/92 Event Time: 1640 Hours

Reactor Mode(s): 1 Mode(s) Name: Run Power Level(s): 89%

B. DESCRIPTION OF EVENT

On March 1, 1992, LaSalle Station Unit 1 was in Operational Condition 1 (Run) at a power level of 89%. At approximately 1635 hours, a Unit 1 steam seal header low pressure alarm was received in the Control Room. At LaSalle, sealing steam for the Main Turbine (TG) [TA] is produced from a Steam Seal Evaporator (SSE, GS) [TC]. The SSE is a shell and U-Tube type heat exchanger of conventional design. Heating steam, which can be Extraction Steam (ES) [SE] from the turbine (normal source) or Main Steam (MS) [SB] from the equalizing header, is passed through the tubes which boils the shell side condensate (CD) [SD] to produce sealing steam. Upon investigation of the alarm, the Nuclear Station Operator (NSO, licensed RO) observed that steam seal header pressure was downscale but the SSE shell water level was normal. Further investigation revealed that the Extraction Steam Admission Non-return Check Valves were closed. These indications led the NSO to believe that the low steam seal header pressure was due to a loss of heating steam.

At this time, Off Gas (OG) [WF] flow was increasing and condenser vacuum began decreasing. The Unit NSO called for assistance from the auxiliary RO's present in the Control Room. One RO began reducing power while another began performing LaSalle Operating Procedure LOP-GS-03, "Transfer to the Backup Gland Seal Steam Supply". When attempting to establish MS as the steam seal source, SSE bypass valve 1GS-002 failed to open as required per LOP-GS-03. An operator was subsequently dispatched to locally open the valve. Prior to opening the valve, Condenser Low Vacuum alarms annunciated in the Control Room and approximately 30 seconds later the turbine tripped on low vacuum. Since reactor power was greater than 30% (75% and decreasing), the turbine trip caused a Unit 1 reactor scram due to Turbine Stop Valve (EH) [TG] closure. Recovery actions from the scram were normal and reactor parameters were stabilized.

Review of the plant performance after the trip led to investigations of several components. No significant failures occurred.

After the scram, a Group 1 Safety Relief Valve (SRV, NB) [AD] lifted. Group 1 SRV's have the lowest relief pressure settings and would be expected to lift first during a pressure transient. Review of the vessel pressure transient indicates the SRV operated consistent with its design setpoint.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION							Form Rev 2.0
FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					Page (3)
		Year	////	Sequential Number	////	Revision Number	
LaSalle County Station Unit 1	0 | 5 | 0 | 0 | 0 | 3 | 7 | 3	9 | 2	-	0 | 0 | 3	-	0 | 0	0 | 3 OF 0 | 6

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

B. DESCRIPTION OF EVENT (CONTINUED)

Reactor Water Level spikes were observed on Startrec (a high speed transient data recorder) for approximately 1.5 seconds after the scram. At the low level point of the first spike, alarms for LEVEL 2 (-50") and LEVEL 1 (-129") were annunciated although actual reactor water level never fell below +5". These level spikes were of short duration and shorter than the response time of most Engineered Safety Feature (ESF) actuation relays. However the following ESF actuations did seal-in:

1. Reactor Core Isolation Cooling System (RCIC, RI) (BN) initiated and injected into the reactor vessel,

2. Division 1 Anticipated Transient Without Scram (ATWS) logic sealed in resulting in the trip of both Reactor Recirculation (RR) [AD] Pumps, and

3. Division 1 Alternate Rod Insertion (ARI) (RD) [AA] initiated.

The level spikes result from pressure waves traveling from the Turbine Control and Stop Valves up the steam lines into the Reactor Pressure Vessel dome. General Electric, the Nuclear Steam Supply System (NSSS) supplier, has indicated that these spikes are expected and were present on the initial cycle startup tests.

This event is reportable pursuant to the requirement of 10CFR50.73(a)(2)(iv) due to the automatic actuation of the Reactor Protection System.

C. APPARENT CAUSE OF EVENT

The cause of this event was the loss of main turbine sealing steam. This resulted in a loss of condenser vacuum, turbine trip, and reactor scram.

The root cause of this event appears to be loss of SSE condensate level control due to a partially plugged level instrument sensing line. Upon investigation, the variable leg of the level sensing instrument used to control condensate feed flow into the SSE was found to be partially plugged. This fact severely inhibited the feedflow control system to respond to level changes in the SSE.

Although Control Room indication of SSE water level was normal during the event, speculation is that actual level was decreasing and eventually reduced to zero (dry). As shell side pressure decreased with level, the heating steam demand signal increased. This was due to the fact that the SSE steam control system erroneously assumed insufficient heating steam was the cause for the low steam seal header pressure. Since extraction steam was already full open to the SSE, the control system began opening the Main Steam Admission Valve for additional heating steam. After the Main Steam Valve opened, the Extraction Steam Non-return Check Valves closed by design due to the pressure differential between main steam and extraction steam. The operator noticed these check valves closed and, coupled with normal SSE water level indication, incorrectly diagnosed the problem to be loss of heating steam. Once it was apparent that normal operation of the SSE could not be achieved, attempts were made to bypass the SSE and send main steam directly to the seals. It was at this time valve 1GS-002 failed to open. While local efforts were in progress to open the valve, the turbine tripped on low vacuum and the reactor scrammed.

D. SAFETY ANALYSIS OF EVENT

Turbine trip with bypass is classified as a transient of moderate frequency in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). The turbine trip in this event initiated the reactor scram as required. Following the scram, reactor pressure was controlled by the SRV's and Turbine Bypass Valves. These events are consistent with the UFSAR analysis of this event.

Safety features were initiated as designed and expected, with the exceptions of the following:

1. RCIC initiation and vessel injection,

2. ATWS initiation with Recirculation Pump Trip (RPT), and

3. ARI initiation.

These actuations were due to pressure spikes sensed in the variable leg of various level instruments.

E. CORRECTIVE ACTIONS

Work Request L13871 was written to inspect the SSE control system. Work Request L13870 was written to investigate valve 1GS-002.

The Instrument Maintenance Department (IMD) performed calibration checks under work request L13871 on the six instrumentation loops existing on the SSE. The results of these checks were as follows:

1. Remote condensate level indication/alarms -- no problems were found with instruments although it was noticed that control room level indication severely lagged behind actual level changes.

2. Level control loop -- the level transmitter had drifted out of tolerance low. This would have resulted in level control 1.5-2" below the desired setpoint. This would have had minimal impact on the event.

3. Tubeside heating steam drain tank normal level control loop -- the transmitter for the normal drain valve was found to be inoperable. The failure of this transmitter rendered the control loop inoperable and caused the normal drain valve to fail closed. Since the emergency level control loop was operational (see below), this failure had no impact on the event.

4. Tubeside heating steam drain tank emergency level control loop -- this loop was found fully operational.

5. Tubeside/shellside pressure control loop -- this loop consists of two pressure controllers (tubeside and shellside), a high select relay, and a pnuematic valve positioner for the tubeside Main Steam Admission Valve. Both pressure controllers send their output to the high select relay which passes the higher pressure signal to position the main steam valve. A higher pressure signal calls for reduced opening of the main steam valve. During the inspection, the tubeside controller was found to be inoperable. Aside from providing excess main steam to the tube side, this failure had minimal impact on the event.

E. CORRECTIVE ACTIONS (CONTINUED)

6. Control room tubeside/shellside pressure indication -- this loop was found fully operational.

As a result of calibration checks on the level instruments, LaSalle Limited Procedure LLP-92-060 was written to further investigate the level sensing instrumentation. During the performance of this procedure it was discovered that the level instrument variable leg piping (common between instrument loops #1 and #2 above) was partially plugged. This would explain the failure of the level control system to maintain proper level and why the Control Room level meter indicated normal level when the shell had actually boiled dry (no evidence of shell damage). The instrument piping was flushed out several times to remove the debris.

All control loop discrepancies noted above were corrected. Following repeated instrument piping flushes, the level instrument loops were verified to respond normally to SSE level changes.

Troubleshooting of valve 1GS-002 under Work Request L13870 did not reveal any obvious problems. With the unit shutdown and no differential pressure across the valve, 1GS-002 cycled satisfactorily several times. Further investigation, however, revealed that the closed limit switch (LS/C) used for bypassing the open torque switch (TSC) was probably set too low (set at 5%). It is probable that, with main steam differential pressure across the closed valve, the opening torque is still greater than the TSO setpoint when the valve reaches 5% open. With both the closed limit switch and torque switch open, valve travel in the open direction is prevented. This hypothesis is consistent with the observations of the unit NSO, i.e. power to the valve did not trip; the valve simply quit moving.

The aforementioned limit switch was reset to 25% and valve 1GS-002 successfully cycled at full power against rated differential pressure.

Due to the generic nature of this Motor Operated Valve (MOV) problem, the Electrical Maintenance Department (EMD) is looking at other balance of plant valves with similar settings in high differential pressure applications. This action is being tracked by Problem Analysis Data Sheet (PADS) E-0144. No safety related valves are affected by this issue since their open torque switch bypasses have already been set to 25% in accordance with Commonwealth Edison Company (CECo) Nuclear Operations Directive (NOD) MA-1.

The steam dome pressure spiking from Turbine Stop Valve closure is a phenomenon affecting the reference legs of various level instrumentation. Wide range level instrumentation using Rosemount 1153 and 1154 series transmitters appear to be most affected by these pressure spikes. A long term solution to this problem is being tracked by Action Item Record 374-200-90-06501.

Unit operation was resumed on March 5, 1992. On March 16, 1992 during a scheduled load drop, SSE level control experienced another upset. The remote level indicator was found again reading high. The electronic zero of the indicating loop was shifted to correct the discrepancy. The operating department has started a special log (shiftly) to verify control room level indication agrees with actual SSE level.

F. PREVIOUS EVENTS

There are no previous occurrences of low vacuum turbine trips from SSE malfunctions.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

G. COMPONENT FAILURE DATA

Manufacturer	Nomenclature	Model Number	MFG Part Number
Fischer Controls	Level Xmitter	2500T-259B	N/A
Fischer Controls	Press Controller	4181	N/A

NRC FORM 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit 3	0 5 0 0 0 2 8 7	9 2	— 0 0 2	— 0 0	0 8 OF 0 8

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

ATTACHMENT 1