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

November 18, 1994

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

Licensee Event Report #94-008-00, Docket #050-374 is being submitted to your office in accordance with 10CFR50.73(a)(2)(iv).

D. J. Ray
Station Manager
LaSalle County Station

DJR/SCK/lja
Enclosure
cc: NRC Region III Administrator
NRC Senior Resident Inspector
INPO - Records Center
IDNS Resident Inspector
IDNS Senior Reactor Analyst
Nuclear Licensing Administrator

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1/1

LICENSEE EVENT REPORT (LER)

Form Rev 3.0

Facility Name (1)	Docket Number (2)	Page (3)
LaSalle County Station Unit 2	0 5 0 0 0 3 7 4	1 of 0 8

Title (4)

Reactor Scram Due to Electro-hydraulic Control Line Break

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)
1 0	1 9	9 4	9 4	--- 0 0 8 ---	0 0	1 1	1 8	9 4		

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (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)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)
		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)

73.71(b)
73.71(c)
Other (Specify in Abstract below and in Text)

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER
	AREA CODE
Steve Kleinhardt, EHC System Engineer, Extension 2245	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	J J			Y					

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 October 19, 1994, Unit 2 was in Operational Condition 1 (Run). Turbine Control Valve oscillations were first observed at 1015 hours. Plant personnel were unable to immediately determine the cause of the oscillations and as a result, Turbine Electro-hydraulic Control (EHC,EH)[TG] piping, which was subjected to the same oscillations hydraulically, began to leak. The piping eventually broke due to low cycle fatigue. A Main Turbine trip occurred due to the loss of EHC fluid pressure, and the reactor scrammed on high neutron flux when the turbine control valves closed.

Reactor Core Isolation Cooling (RCIC, RI)[BN] and Anticipated Transient Without Scram-Recirc Pump Trip/Alternate Rod Insertion (ATWS-RPT/ARI) initiated on spurious low reactor water level (Level 2, -50") signals. Six Safety Relief Valves (SRVs) opened sequentially following the turbine trip. The required notifications were made, and troubleshooting and investigation were initiated.

An investigation was performed, and the root cause for the EHC control valve oscillations was due to a faulty connector on the EHC permanent magnet generator (PMG) 30 Vdc power supply. The faulty connector allowed the 30 Vdc power to cycle between the house supplied power supply and the PMG power supply.

EHC piping and the electrical connector were repaired and an extensive inspection of EHC piping was completed prior to starting up Unit 2.

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): 2 Event Date: 10/19/94 Event Time: 1057 Hours

Reactor Mode(s): 1 Modes(s) Name: Run Power Level(s): 100%

B. DESCRIPTION OF EVENT

At 1015 hours on 10/19/94 Unit 2 was in Operational Mode 1 (Run) at 100% power when the Unit 2 Nuclear Station Operator (NSO)(RO) observed Turbine Electro-hydraulic Control (EHC,EH)[TG] pressure set bias swings and Turbine Control Valve oscillations of approximately 2-3%. Reactor power and pressure remained stable.

Control Room Personnel notified System Engineering and Instrument Maintenance (IMD) to assist with troubleshooting. Operators were dispatched to the Auxiliary Electric Equipment Room where they noted small voltage swings on the 30 Vdc power supply. IMD later confirmed that the 30 Vdc power supply appeared to be switching back and forth between house power and the permanent magnet generator (PMG). Power was reduced 15 MWe in an effort to change the precise point at which EHC was controlling but the oscillations continued.

The acting Unit 2 Operating Engineer (OE) had been requested to report to the Control Room, and as he travelled from the Service Building to the Control Room, he looked in the EHC Pump Room. The OE witnessed and reported that EHC piping near the pump skid had substantial movement and that the oscillations needed to be stopped quickly.

At approximately 1050, IMD Personnel were about to adjust the power supplies to stop the electrical oscillations when a licensed Shift Supervisor, who had been dispatched to inspect the EHC Pump Room, reported a large EHC fluid leak on the pump skid. The Shift Supervisor could not enter the room to determine the exact location of the leak due to the mist of EHC fluid (Fyrquel) creating a hazardous environment. In an attempt to stop the leak, the EHC pumps were swapped. Immediately after the pump swap, the EHC system pressure decreased rapidly, and the Main Turbine tripped on low EHC fluid pressure. It was subsequently determined that the EHC line failed as a result of low cycle fatigue and breakage at a piping size reducer at the EHC pump skid.

B. DESCRIPTION OF EVENT (continued)

The loss of EHC fluid caused the Turbine Control Valves to go shut, which resulted in the reactor scramming on high flux. The closing of the Turbine Control Valves caused reactor pressure to increase which collapsed voids within the reactor driving power upward to the scram setpoint. A direct scram signal from the Turbine Stop Valve position occurred milliseconds after the high flux scram. This sequence is appropriate for these circumstances. The maximum reactor power reached during the transient was 123% before the control rod insertion from the scram turned power downward.

Main Turbine Bypass Valves opened as well as six Safety Relief Valves (SRV,MS)[SB] to limit reactor pressure rise to approximately 1090 psig. The Bypass Valves stayed open for 23 seconds on the accumulator pressure until reactor pressure dropped below the EHC pressure setpoint (approximately 1000 psig). After the Bypass Valves closed, the running EHC pump was secured, thereby stopping the EHC fluid leak in the EHC pump room.

When the Main Turbine tripped, a hydraulic shock wave was created from the rapid closure of the Turbine Stop Valves which affected the reactor vessel level instrument lines and caused Reactor Core Isolation Cooling (RCIC,RI)[BN] and ATWS-RPT/ARI initiations. Reactor Recirculation (RR)[AD] pumps tripped to zero speed from the ATWS-RPT initiation and RCIC injected into the reactor vessel. This occurred due to the sensitivity of the Rosemount level transmitters and the shock wave having generated a false (-50") low reactor level signal. The minimum recorded reactor vessel level was approximately 0" on a wide range level recorder.

Following the Main Turbine trip, a generator trip was received on reverse power which in turn initiated a fast transfer of 4.16Kv and 6.9Kv electrical busses that are fed from the Unit Auxiliary Transformer (UAT) to the System Auxiliary Transformer (SAT). During the fast transfer, 2C Circulating Water (CW)[KE] Pump, 2C Primary Containment (VP)[KM] Chiller and the Process Computer room north air conditioning unit tripped. 2A CW pump tripped 36 seconds after the fast transfer. It is believed that these trips occurred due to control power degradation in the control circuits of the equipment. The cause is still being investigated.

The 2A Turbine Driven Reactor Feed Pump (TDRFP,FW)[SJ] failed to trip on the first attempt to manually trip the pump, because the operator did not depress the trip pushbutton for the 750 milliseconds required for the logic to pickup. The NSO immediately identified this problem and depressed the trip pushbutton long enough to actuate the trip logic, and the pump tripped. Subsequent investigation revealed that the pump would have tripped if the manual trip push button had been depressed slightly longer to enable the hydraulic actuation of the trip to occur.

B. DESCRIPTION OF EVENT (continued)

An observation was made during the investigation of the scram and related events that RCIC Injection Testable Check Valves (2E51-F065 and 2E51-F066) had position indication concerns. When Control Room Operators decided to switch RCIC modes from reactor vessel injection to reactor pressure control, the injection testable check valves did not indicate full closed. Operating switched modes to re-inject into the reactor vessel, and then decreased RCIC system pressure to less than reactor pressure in order to seat the testable check valves prior to closing the RCIC Injection Valve (2E51-F013). The testable check valves require some differential pressure to cause the valves to fully seat. Any condition of reverse flow through the testable check valves would have seated them and provided proper full closed indication. The appropriate Operating Procedures will be changed to reflect how to properly shutdown or re-align the system.

Nine control rods failed to indicate "full in" for approximately 2 minutes following the scram.

C. APPARENT CAUSE OF EVENT

The root cause of this event was a faulty connector on the 30 Vdc power supply which created electrical oscillations affecting Turbine Control Valve positions. The electrical pulses were converted to hydraulic pulses going to the Turbine Control Valves which established vibration that caused a piping support in the EHC Pump Room to separate from the piping. Without proper restraint, the EHC piping was subjected to substantial swings which resulted in low cycle fatigue and breakage at a piping size reducer at the EHC pump skid.

The cause of the turbine trip and reactor scram was the EHC pipe breaking and bleeding pressure off the Turbine Control Valves allowing them to drift closed.

RCIC and ATWS-RPT/ARI Systems automatically initiated due to a spurious Level 2 (-50 inches) initiation signals. The spurious "Reactor Level 2 Low" signals were due to a pressure oscillation (ringing) which resulted from the closure of the Turbine Stop Valves. The individual spikes decayed to nearly a zero amplitude in approximately 3-4 seconds. This phenomenon was previously documented following Main Turbine trips on both units. The duration of the spikes have not been sufficient nor have they been in phase such that all isolation or actuation instrumentation are able to sense the trip signals simultaneously.

C. APPARENT CAUSE OF EVENT (continued)

The probable cause for equipment trips during fast bus transfers of non-safety related busses is that certain 120 Vac control circuits, powered from a transferring bus, do not survive the suppressed voltage experienced during the transfer. LaSalle has done extensive research on the safety related power system and has made design changes due to calculated degraded voltage problems.

Lack of Control Rod "full in" indication situation has occurred following previous scrams and is due to the control rods being inserted slightly beyond the full in reed switch position and the Rod Worth Minimizer (RWM,RW)[AA] does not acknowledge the rods as being "full in" until they settle back to the "full in" reed switch position. Both units have corrected this condition via a design change that has been installed on the "A" RWM computer for each respective unit. Testing has been successfully completed on one Unit 1 "A" RWM computer and will be complete on the Unit 2 "A" RWM computer during the next opportunity of having all control rods fully inserted.

D. SAFETY ANALYSIS OF EVENT

A complete shutdown of the reactor was successful as a result of the automatic scram signal. Reactor pressure vessel (RPV) level was never below the 0 inch level (more than 13 feet above the top of active fuel). Six SRVs opened, limiting maximum reactor pressure to 1090 psig. All five Turbine Bypass Valves opened and remained open until reactor pressure had dropped below the EHC pressure setpoint. RCIC automatically initiated and injected into the reactor vessel and ATWS-RPT automatically initiated a trip to zero speed of the RR pumps. These items create no direct safety concern but do provide a distraction to the Control Room Operators.

Based on the above, the overall safety significance to this event is minimal.

E. CORRECTIVE ACTIONS

EHC Electrical

1. Investigate availability of improved connectors for the EHC System.

2. Disseminate information to workers regarding this problem to re-emphasize the importance of proper inspection, handling, and installation of connectors.

E. CORRECTIVE ACTIONS (continued)

EHC Electrical (continued)

3. Inspect all the installed connectors of this type in the EHC system, on both units, for similar problems.

EHC Mechanical-Piping Failure

1. Implement TIL 1123-3, Hydraulic Leaks in BWR Steam Turbine Electro-hydraulic Control Systems, recommendations.

2. Evaluate benefits of design change to eliminate pipe size reduction at the manifold.

3. Modify appropriate Operating procedures to minimize EHC piping oscillations during turbine chest and shell warming.

4. A Temporary Procedure Change was written for LOP-TG-01, "Turbine Trip Resetting, Shell Warming, and Chest Warming". Appropriate Operating personnel were trained on this change prior to taking the shift following this event.

5. Provide additional training to Operators on methods to avoid EHC oscillations and corrective measures to take if oscillations occur.

6. Engineering is monitoring the EHC piping through physical walkdowns and TV cameras monitoring of EHC piping in the Heater Bay.

EHC Mechanical-Hangers/Supports

1. Re-evaluate present EHC piping support design, particularly the pump room clamp, and locations in heater bay that allow significant pipe movement.

2. Consider addition of locking mechanisms for support bolting, such as lock nuts, lock wire, tabs, locking compound, etc.

3. Expand pipe support inspection plan to include more non-Safety Related piping supports, especially high vibration systems, such as EHC.

Reactor Level 2 Low Signal

The ringing experienced by the level instrumentation is a natural phenomena which has been seen at LaSalle following scrams due to turbine trips. The effect of this ringing has been reviewed and installation of a modification which would provide a time delay of the signal has been installed on Unit 1 RCIC.

E. CORRECTIVE ACTIONS (continued)

Reactor Level 2 Low Signal (continued)

AIR 374-200-90-06501; "Review Need to Filter RWL Trips" remains open. This AIR has tracked the RWL ringing issue and the subsequent modification. (RCIC TD Relay Installation)

1. Install RCIC time delay relay design change on Unit 2 as soon as practicable.

Bus Transfer

Verify cause(s), then develop and implement a viable action plan to fix the problem.

TDRFP Trip Operation

1. For Unit 2, install design change provide seal-in feature on manual trip pushbutton actuation.

2. Remove unneeded Unit 1 main control board operator aid - manual TDRFP trip labels. This action has been completed.

3. Consider changing the guidance for Operators on manual TDRFP trip. (For example; depress trip pushbutton until valve positions indicate a trip has been completed.)

Control Rod Position Indication

Install the rod capture modification on all Rod Worth Minimizer (RWM) computers, bot units.

RCIC Testable Check Valves

The appropriate Operating procedures will be revised to give specific guidance to enable all NSO's to accomplish the same results.

F. PREVIOUS EVENTS

LER Number	*Title*
374/91-010	Manual Scram Due to Electro-hydraulic Control Leak
373/92-003	Reactor Scram Due to Loss of Condenser Vacuum

F. <u>PREVIOUS EVENTS</u> (continued)

LER Number	*Title*
374/92-012	Reactor Scram Due to a Main Turbine Trip Caused by a Thrust Bearing Wear Detector Signal

G. <u>COMPONENT FAILURE DATA</u>

None.