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

January 25, 1991

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

Dear Sir:

Licensee Event Report #90-001-01, Docket #050-374 is being submitted to your office in accordance with 10CFR50.73(a)(2)(iv) and supersedes previous submitted report.

for G. J. Diederich
Station Manager
LaSalle County Station

GJD/AJM/mkl

Enclosure

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

IE22
1/1

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

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

Title (4)

Reactor Scram during Instrument Surveillance Testing Caused by Spurious Spike on Average Power Range Monitor

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 0 0 0
0 2	0 6	9 0	9 0	0 0 1	0 1	0 1	2 5	9 1		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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	
Alan J. McLaughlin, Technical Staff Engineer, Extension 2705	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	L G			N					
X	A A		G 0 8 0	Y					

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 February 6, 1990 at 0926 hours, while Unit 2 was in Operational Condition 1 (Run) at 99.8% power, during the performance of LaSalle Instrument Surveillance LIS-NR-403, "Unit 2 Average Power Range Monitor (APRM) Rod Block and Scram Functional Test," a full reactor scram occurred. Normally the surveillance only causes 991f-scrams. At the time of the occurrence, F APRM was tripped, per the procedure, which tripped Reactor Protection System (RPS) Channel A. While the RPS Channel "A" half scram condition was in effect, E APRM spiked spuriously, causing RPS Channel B to trip and a full reactor scram occurred.

Additionally, it was also determined that all other expected automatic actions occurred as expected including Primary Containment Isolation signals when reactor water level reached 12.5 inches decreasing.

Initiation of the event was not due to an actual transient on a parameter which is monitored to protect the reactor core but due to spurious spike of APRM E. Troubleshooting E APRM was performed under Work Request L96857 in an attempt to determine the cause of the spurious spikes. The cause of the spikes could not be repeated or determined.

Unit 1 was not affected by this event.

This event is reportable pursuant to 10CFR50.73(a)(2)(iv) due to an actuation of an Engineered Safety Feature.

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: 02/06/90 Event Time: 0926 Hours

Reactor Mode(s): 1 Mode(s) Name: Run Power Level(s): 99.8%

B. DESCRIPTION OF EVENT

On February 6, 1990 at 0926 hours, while Unit 2 was in Operational Condition 1 (Run) at 99.8% power, during the performance of LaSalle Instrument Surveillance LIS-NR-403, "Unit 2 Average Power Range Monitor (APRM) Rod Block and Scram Functional Test," a full reactor scram occurred. Normally the surveillance only causes half-scrams. At the time of the occurrence, F APRM [IG] was tripped, per the procedure, which tripped Reactor Protection System (RPS, C71) [JC] Channel A. While the RPS Channel "A" half scram condition was in effect, E APRM spiked spuriously, causing RPS channel B to trip and a full reactor scram occurred. No other Neutron Monitor trips were present prior to the time of the reactor scram.

After the automatic scram, the Control Room Operator (NSO) noticed that Control Rod Drive (CRD) (RD, C11) [AA] 26-47 was latched at the "02" position. The rod was subsequently manually inserted to the required "00" position. Subsequent testing revealed that control rod 26-47 initially went to the Full-In position. Upon resetting the automatic scram, the rod drifted and latched at the "02" position.

Additionally, it was determined that all other expected automatic actions operated correctly including Primary Containment Isolation signals (PCIS, PC) [JM] when reactor water level reached 12.5 inches decreasing.

Unit 1 was not affected by this event.

This event is reportable pursuant to 10CFR50.73(a)(2)(iv) due to an actuation of an Engineered Safety Feature.

C. APPARENT CAUSE OF EVENT

The cause of the reactor scram was due to an unexpected actuation of the E APRM trip circuitry. A half scram condition existed due to the performance of LIS-NR-403. When the E APRM trip circuitry actuated, a full reactor scram occurred. The cause of the spike is unknown.

C. APPARENT CAUSE OF EVENT (Continued)

Special recorders had been installed in the RPS logic strings to attempt to help pinpoint the source and collect data on the character of very short trip signals entered into the RPS logic (LER 374/89-011-01). The recorder on the A2 logic string recorded this scram event. This recorder trace shows that the RPS logic string was opened for slightly less than 1 AC cycle, or approximately 12 - 15 milliseconds. The RPS contactors completely de-energized as expected. A recorder trace was also obtained from the B1 logic string recorder. It was triggered at the time of the intentional 1/2 scram from APRM F. This recorder trace does not extend in time to the APRM E trip. (Due to the scan rate, a recorder trace only shows about 400 milliseconds of data - the E APRM trip was 1.3 seconds after APRM F). However, it does show that the electrical conditions in the B RPS channel were completely stable within 2 AC cycles after the F APRM trip, well before the APRM E trip occurred.

The RPS recorders indicate that the APRM E trip was internal to the APRM drawer, and not induced by the actuation of other RPS circuitry. There appears to be no direct relationship between the surveillance in APRM F and the APRM E trip. This was verified by the testing sequence following the scram, where the functional tests for APRM F, then E, then F again were conducted, with no trip indications appearing in APRM E. (This test sequence is the same as occurred just before the reactor trip.)

Troubleshooting CRD 26-47 indicated that it was running "hotter" than most other CRD's. This is consistent with data collected over the past cycle. This data, collected previously per LaSalle Special Test, LST-89-151, "CRD Thermocouple Direct Measurement at 1(2)C11-R018 Recorder," showed CRD 26-47 running consistently at 350 - 400 degrees F. At the start of this cycle, both the insert and withdraw stall flows were still in the normal range. This CRD was installed in Unit 2 during its first refuel outage in the spring of 1987.

Due to the station's awareness of this CRD's high temperature and degrading stall flow trends several corrective measures were attempted during the August, 1989 Unit 2 shutdown. This included flushing the CRD in accordance with LaSalle Operating Procedure, LOP-RD-19, "CRD Flushing," which involves a high pressure extended withdraw stall flow to flush collet area. Also, the CRD cooling water orifice was back flushed through the insert line vent valve in an attempt to increase cooling water flow. Neither of the above actions proved successful for this CRD.

CRD 26-47 (S/N A994) was replaced during LaSalle's Unit 2 third refueling outage (L2R03) per Work Request L94782 and rebuilt on September 4, 1990 per Work Request L96482 in accordance with LaSalle Maintenance Procedure LMP-RD-01, "Control Rod Drive Inspection and Maintenance." This inspection revealed the following major items:

Stop piston seal rings number 2 and 3 had broken springs.
Inner filter was lined with dirt.
Outer filter was clean.
Cooling orifice appeared unblocked.
A nickel based thread lubricant (N 5000), which is used to lubricate the CRD Flange cap screws, was found in the area of the drive piston and drive piston seals.

C. APPARENT CAUSE OF EVENT (Continued)

From these inspection results it is believed that an excessive amount of thread lubricant was used on the Flange cap screws causing the excess lubricant to be squeezed into the CRD internals when the flange was torqued to the CRD housing. This lubricant might then have restricted the flow of cooling water through the ball retainer orifices, cooling orifice, or past the CRD seals. This could explain why this CRD was running excessively hot (approximately 350-400 F) during the previous (third) operating cycle. This known hot operating temperature is what caused the two stop piston seal springs to fail due to thermal and mechanical stresses experienced when the rod was exercised or scrammed.

This stop piston seal damage is what caused the rapidly increasing withdrawal stall flow trend as the seals degraded over the third cycle. Excessive stop piston seal leakage is known to result in a CRD operating anomaly of "02" scrams in which the CRD is prematurely stopped just short of Full-In. This condition (as described in General Electric Company Service Information letter (SIL) 052 S1) occurs when reactor vessel back flow through the stop piston seals is increased sufficiently to stop the index tube motion prior to Full-In. Once the scram is reset the CRD then settles to 02. However, the pressure above the index tube will not build up until the CRD begins to slow down in the buffer hole region past 02 and hence is not expected to settle any further withdrawn than 02.

D. SAFETY ANALYSIS OF EVENT

Initiation of the event was not due to an actual transient on a parameter which is monitored by RPS. With the exception of Control Rod Drive 26-47, all systems required to operate functioned as designed.

A shutdown margin calculation was performed by General Electric to determine the consequences of CRD 26-47 settling at position "02" instead of "00." The results indicate that the shutdown margin was adequate with this rod at position "02" and the next strongest rod fully withdrawn.

From the discussion in Section C of this LER relating to GE SIL 052, it is believed to be highly unlikely that the CRD would settle any further withdrawn than 02.

Based on the above, the safety consequences of this event are considered minimal.

E. CORRECTIVE ACTIONS

An investigation wa[illegible]formed following the event and the following items were noted:

N[illegible] LPRM HI or HI-HI alarms annunciator before or during the reactor scram.

[illegible]ransmissions were noticed just prior to or during the scram. This addressed any radio noise concerns which may have caused interference with the RPS or APRM drawer's trip

[illegible]sonnel were present inside the Control Room panels which could have inadvertently bumped a

E. CORRECTIVE ACTIONS (Continued)

An interview of all personnel working in the Control Room area was performed and it was determined that no work was in progress that could have resulted in a trip of APRM Channel E. It was also determined that no one was present in the Cable Spreading Room which could have used a radio or bumped a cable.

It was determined that no welding was in progress at the time of the event that could have resulted in the trip of APRM E.

The entire calibration procedure for APRM E was conducted to ensure that no abnormalities exist in the circuitry, not seemingly related to the neutron flux trip circuit. This calibration procedure completely checks the drawer, power supply ripple and regulation, and all trip functions and indications. No out of calibration conditions were found.

Troubleshooting E APRM continued under Work Request L96857 in an attempt to determine the cause of the spurious spikes. The cause of the spike was not determined and the problem could not be repeated. As a precautionary measure, the 2C51E-K18 relay was replaced. This relay, when de-energized, places a 15% power scram setpoint on the APRM when the mode switch is in the "Startup" or "Refuel" positions. By placing the mode switch in "RUN", the relay becomes energized and the scram setpoint is then flow biased.

As control rod 26-47 would likely again drift from position "00" to position "02" after a scram reset, the shutdown margin analysis was revised to assume that control rod 26-47 is at "02" along with the highest worth rod being fully withdrawn, prior to the startup after the scram. A caution card was placed on the Control Room bench board (scram reset switch) to immediately select and insert CRD 26-47 when resetting a full scram. This caution card has been removed since the drive was replaced under Work Request L94782.

CRD 26-47 (S/N A994) was inspected and rebuilt following LaSalle's Unit 2 third refuel outage (L2R03) under Work Request L96482. The results of this inspection are described in Section C of this LER. As the 02 scram is believed to have been attributed to by an excess use of thread lubricant on the CRD Flange cap screws, revisions will be made to LMP-RD-05, "CRD installation using GE supplied Winch System" and LMP-RD-19, "CRD Changeout using NES Handling System" adding a caution to use the thread lubricant sparingly and wipe off excess. Action Item Record (AIR) 374-200-90-00302 will track the completion of these procedure revisions.

F. PREVIOUS EVENTS

LER Number	Title
374/89-011-00	Spurious Reactor Protection System Actuation Due to Unknown Causes

G. COMPONENT FAILURE DATA

N/A