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

DCD

February 05, 1992

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

Dear Sir:

Licensee Event Report #92-001-00, Docket #050-373 is being submitted to your office in accordance with 10CFR50.73(a)(2)(v).

WRD [signature]
for G. J. Diederich
Station Manager
LaSalle County Station

GJD/GM/mkl

Enclosure

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

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 5

Title (4)

Average Power Range Monitors (APRM) Set Nonconservatively Due To Communication Error

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 1	0 7	9 2	9 2	0 0 1	0 0	0 2	0 5	9 2		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 8

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	__ 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	X 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	
Guy McCallum, Technical Staff Engineer, Extension 2250	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
A	I G			No					

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 January 7 1992, during LaSalle Instrument Surveillance LIS-NR-109 "Unit 1 Average Power Range Monitor (APRM) Gain Adjustment", a miscommunication event occurred. At 0740, a Control System Technician (CST) commenced to perform the gain adjustment for the APRM Neutron Monitoring System (NR) [IG]. He contacted a Qualified Nuclear Engineer (QNE) for the value to which the gains should be set. The QNE assumed the CST was referring to LIS-NR-107 "Unit 1 APRM/Rod Block Monitor Flow Converter To Total Core Flow Adjustment". The CST provided the QNE with the drive flows from the Core Monitoring Code's Core Power to Flow Log, and the QNE instructed him to set the gains to 93 percent while the reactor power was actually 98 percent.

The event resulted in all six APRMs exceeding their allowable Technical Specification tolerance, and the intended function of the Reactor Protection System (RPS, RP) [JC] was compromised. The total time elapsed from the time at which the first APRM was set nonconservatively to the time when all six APRMs were set correctly was 52 minutes. The individuals were counselled on the importance of communication and having a questioning attitude. Procedure revisions will be implemented providing additional guidance to the CSTs while setting the APRM gains, and requiring operations review of desired APRM settings.

This event is reportable to the Nuclear Regulatory Commission as a Licensee Event Report in accordance with 10CFR50.73(a)(2)(v) due to an event which could prevent the fulfillment of a safety function.

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: 01/07/92 Event Time: 0740 Hours

Reactor Mode(s): 1 Mode(s) Name: Run Power Level(s): 98%

B. DESCRIPTION OF EVENT

On January 7, 1992, at 0740 hours, with Unit 1 in Operational Condition 1 (Run), a Control System Technician (CST) was requested to perform LaSalle Instrument Surveillance, LIS-NR-109 "Unit 1 APRM Gain Adjustment". Prior to commencing, the CST obtained and reviewed the procedure, and, in accordance with Step F.1., identified that the desired power level was to be obtained from either a Nuclear Engineer or the on duty Shift Engineer.

The CST contacted a Qualified Nuclear Engineer (QNE), informed him that he was performing "a gain adjustment", and requested a value for the gains. The CSTs rarely contact a Nuclear Engineer for the Average Power Range Monitor (APRM) (NR) [IG] Gain Adjustments. Consequently, the QNE assumed the CST was performing LIS-NR-107 "Unit 1 APRM/Rod Block Monitor Flow Converter to Total Core Flow Adjustment", a surveillance for which he is frequently called upon to provide the desired recirculation flow. The QNE inquired if the CST had obtained the "numbers" from the Core Monitoring Code's (CMC) Core Power and Flow Log (OD-3). The CST became confused over which "numbers" the QNE sought, and told the QNE he would obtain the "numbers" and call him back. The CST proceeded back to the Instrument Maintenance Department for clarification from his Supervisor. Since the Supervisor was unavailable, the CST discussed the QNE's request with another, more experienced CST. The more experienced CST was also confused with which "numbers" the QNE requested, but believed the QNE wanted the summed Reactor Recirculation Pump Drive Flow. The CST returned to the control room to begin LIS-NR-109.

At approximately 0800, the CST called the QNE and provided him with the "numbers" (Drive Flow) requested. Based on the numbers provided, the QNE instructed the CST to set gains to 93 percent. During the conversation the CST recalled stating he was performing APRM gain adjustments, however, the QNE does not recall the CST comments. The CST obtained permission to perform LIS-NR-109 from both the Station Control Room Engineer (SCRE) and the Unit 1 Nuclear Station Operator (NSO). The SCRE and NSO did not question the CST regarding the APRM power or gain settings because the gain settings are normally adjusted to 1.0. This was not a procedural requirement. At 0810, APRM A was set nonconservatively and at 0840 the sixth and final APRM was set nonconservatively. The correct reactor power was 98 percent of rated core thermal power (RCTP), yet the APRMs indicated approximately 93 percent of RCTP. The unit was currently on a ramp to full power at a rate of approximately one percent RCTP per hour. The procedure did not require that power be held constant.

B. DESCRIPTION OF EVENT (CONTINUED)

Upon completion of the APRM calibrations, the CST demanded an OD-3 and identified that all six APRM's had Gain Adjustment Factors (AGAF) of 1.04 to 1.06, which exceeded the Technical Specification tolerance (Table 4.3.1.1.-1). He proceeded to notify the SCRE, of the problem. Concurrently, a QNE was informed by a member of station management that the APRM readings and indicated thermal power on the individual's desktop computer did not appear to be matching. The QNE immediately contacted the SCRE and asked to talk to the CST. When the CST told the QNE what he was doing, the QNE instructed the CST to reset the APRM gains to 1.0. At 0902, all six APRMs were indicating properly, possessing AGAFs within their Technical Specification tolerance (0.98 to 1.02).

This event is reportable to the Nuclear Regulatory Commission as a Licensee Event Report in accordance with 10CFR50.73(a)(2)(v) due to an event which could prevent the fulfillment of a safety function.

C. APPARENT CAUSE OF EVENT

The root cause of this event is a failure in communication between the CST and the QNE. The CST and QNE should have used formal communication skills. The use of the repeat back conversation technique may have eliminated this event. Also, had the CST expressed his confusion about providing the Recirculation Drive Flow values, the QNE might have questioned him further and identified that he had misunderstood which surveillance was being performed. Additionally, during the surveillance, the NSO observed that the APRM readings were decreasing. He knew it was normal for APRM readings to change during this surveillance, so no action was taken at the time.

Additionally the procedure should require the channel being adjusted to be verified within approved tolerance prior to proceeding with the next channel. Also, the lack of second verification of the desired settings presented a situation where a single error resulted in all APRMs being set incorrectly.

D. SAFETY ANALYSIS OF EVENT

The Technical Specifications, "Reactor Protection System Instrumentation Surveillance Requirements", Table 4.3.1.1-1, requires a weekly channel calibration for the APRM Flow Biased Simulated Thermal Power-Upscale and Fixed Neutron Flux-High. The purpose of this calibration is to confirm the APRM channel is consistent with the power level calculated by a heat balance. Furthermore, Note (d) from Table 4.3.1.1-1, states "Within two hours, adjust any APRM channel with a GAF > 1.02", however, the APRM's still remain operable. The Reactor Protection System (RPS, RP) [JC] automatically initiates a reactor scram, and provides limiting conditions for operation necessary to preserve the ability to perform its intended function. The Technical Specifications require a minimum of two operable APRM channels per Trip System (Table 3.3.1.1-1). The trip of one APRM in a channel will result in a half scram. With less than the minimum operable channels per Trip System, place the inoperable Trip System in the tripped condition within one hour (T.S. 3.3.1). In this event, the APRM gains exceeded the allowable tolerance, and from the time the first gain was out of tolerance to all gains within the allowable tolerances, 52 minutes elapsed.

D. SAFETY ANALYSIS OF EVENT (CONTINUED)

The safety significance from this event is minimal. The APRMs were set approximately 5 percent nonconservative. This means a reactor scram resulting from either the Fixed Neutron Flux (120 percent of RCTP) or Flow Biased Thermal Power (115.5 percent of RCTP) may not have initiated until up to 5 percent of RCTP higher than expected. The most limiting Anticipated Operational Occurrence (AOO) which is mitigated by an APRM initiated scram is the Loss of Feedwater Heaters (LOFWH). Actually, the LaSalle Updated Final Safety Analysis Report takes credit for the APRM Flow Biased Thermal Power Scram during the LOFWH event. However, current licensing basis for this event does not take credit for any scram. The Supplemental Reload Licensing Submittal for Unit 1 Cycle 5 demonstrates that the LOFWH AOO Minimum Critical Power Ratio (MCPR) is bounded by the Rod Withdrawal Error (RWE), the most limiting AOO for LaSalle Unit 1 Cycle 5. Given the fact that LaSalle Unit 1 had approximately 14 percent margin to its MCPR Limiting Condition for Operation during this event and the additional margin between the LOFWH and the RWE AOOs, it is not expected that the MCPR Safety Limit would have been violated if a LOFWH AOO had occurred.

The RWE AOO is the most limiting event for LaSalle Unit 1 Cycle 5 and is mitigated by the Rod Block Monitor (RBM, NR) [IG] System. If the postulated RWE AOO had occurred during this event, a rod block may not have initiated until 5 percent higher than expected. Given the fact that the RBM settings are typically 3 to 5 percent conservative to the analyzed setpoint at full power and that Unit 1 had approximately 14 percent margin to its MCPR Limiting Condition for Operation during this event, it is not expected that the MCPR Safety Limit would have been violated if the postulated RWE had occurred. Additionally, control rod movements are not performed during this surveillance.

The consequences of the Design Basis Rod Drop Accident (RDA) is terminated with a scram from the APRM 120 percent Upscale Neutron Trip and the inherent Doppler shutdown mechanism. Analysis has shown that above 10 percent RCTP, no RDA can occur with a peak fuel enthalpy greater than the RDA design limit of 280 cal/gm, due to the negative reactivity from increased voiding in the core. Therefore, the fact that the APRMs were set nonconservatively is not expected to have a major impact on the resultant peak fuel enthalpy for a RDA. Additionally, LaSalle had performed weekly exercising of all withdrawn control rods on the previous shift per Technical Specification 4.1.3.1.2.a. This included a coupling check of all fully withdrawn control rods per Technical Specification 4.1.3.6.b. Since the results of this surveillance were satisfactory, a RDA would not have been expected to occur during this event.

E. CORRECTIVE ACTIONS

A Human Performance Enhancement System (HPES) investigation, HPES report 92-03, was performed shortly after the event. Station Personnel will be trained on this event. Emphasis will be placed on the importance of the use of formal communications and repeat backs when transmitting information, (LAP-100-37). General Information Notice (GIN #92-003) was generated to track completion of training personnel on this event.

The LaSalle Instrument Maintenance Surveillance LIS-NR-109 "Unit 1 APRM Gain Adjustment" has been enhanced as a result of this event. The procedural enhancements included the following: (1) Operating personnel acknowledgment of the desired gain adjustment setting, (2) CST verification of the APRM being within approved tolerance prior to starting work on the next channel, (3) detailed explanations for allowable gain tolerance emphasizing gain settings in excess of 1.02 are unacceptable and (4) a precaution that reactor power changes not be made during the calibration. LIS-NR-209 "Unit 2 APRM Gain Adjustment" will be revised prior to Unit 2 Cycle 5 startup. This procedure revision is being tracked by Action Item Record (AIR) 373-180-92-00201.

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 1	-	0 0	0 5	OF	0 5

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

E. CORRECTIVE ACTIONS (CONTINUED)

The Instrument Maintenance Department will review additional procedures to verify components are within acceptable tolerance prior to proceeding. This action is being tracked by AIR 373-180-92-00202.

F. PREVIOUS EVENTS

There have been several Licensee Event Reports at LaSalle which were attributed to poor communications. A list of recent Licensee Event Reports attributed to poor communication practices has been included.

LER Number	Title
374/85-020-00	Missed Service Water Sample, Radiation-Chemistry Supervisor Told Radiation-Chemistry Technician To Sample The Wrong Point
373/86-003-00	Missed Sample On Reactor Water Ph - Personnel Error Missed Communications
373/86-016-00	Fuel Bundle Loaded Without Proper SRM Instrumentation - Poor Communication Techniques
373/87-019-00	Incomplete Surveillance On CRD Cycling - Inattention To Detail And Poor Communication
373/87-030-00	SCRAM Signal During LIS-RP-107 - Miscommunication Between NSO and Electrician.
373/89-028-00	Outboard Isolation Valve Closure Due To Miscommunication Error During Instrument Surveillance
373/90-005-00	Fire Detection Zone Out Of Service Greater Than 14 Days and Special Report Submitted Due To Poor Communications
373/90-008-00	Missed Technical Specification Hourly Fire Watch Due To Miscommunication

G. COMPONENT FAILURE DATA

There was no component failure.