Commonwealth Edison
Quad Cities Nuclear Power Station
22710 206 Avenue North
Cordova, Illinois 61242-9740
Telephone 309/654-2241

RLB-90-281

November 28, 1990

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

Reference: Quad Cities Nuclear Power Station
Docket Number 50-265, DPR-30, Unit Two

Enclosed is Licensee Event Report (LER) 90-011, Revision 00, for Quad Cities Nuclear Power Station.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(iv): The licensee shall report any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

An extension of the 30 day reporting requirement was granted on November 26, 1990 by Mr. Lerch of the NRC, Region III. Report due out by November 28, 1990.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD CITIES NUCLEAR POWER STATION

R. L. Bax

R. L. Bax
Station Manager

RLB/MJB/jlg

Enclosure

cc: R. Stols
T. Taylor
INPO Records Center
NRC Region III

3156H

IE22
1/1

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Quad Cities Unit Two	0 5 0 0 0 2 6 5	1 of 0 9

Title (4)

Unit Scram From IRM-13 and 16 High-High Due To Personnel Inattention.

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
1 0	2 7	9 0	9 0	--- 0 1 1	--- 0 0	1 1	2 8	9 0		0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL (10): 0 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	
M. Brown, Regulatory Assurance, Ext. 3102	3 0 9	6 5 4 - 2 2 4 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

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 27, 1990 at 1559 hours, Unit Two reactor scrammed from Intermediate Range Monitors (IRM) 13 and 16 high-high signals. The station was in the process of returning to normal operation following the discontinuation of a turbine torsional test. While reducing reactor pressure to return the turbine electro-hydraulic control (EHC) system to normal, the Nuclear Station Operation (NSO) did not realize the reactor had gone subcritical. As reactor pressure decreased below 800 psig, the NSO began withdrawing control rods to increase reactor pressure. The rod withdrawals resulted in a short period and the IRM scram.

The primary cause of the event was personnel error. Contributing causes were ineffective communications and management oversight, insufficient training, and the on-site review process.

Corrective actions completed included: an in-depth discussion of the event, additional management oversight, remedial training, and an independent in-depth investigation of the event. Further corrective actions will include: training on this event during license requalification, procedure enhancement, personnel counseling, assessment of reactivity management training, communications enhancement, proceduralized turn-over checklists, and a committee to address procedure adherence.

This report is being submitted in accordance with 10CFR 50.73(a)(2)(iv). An extension of the 30 day reporting requirement was granted by Region III NRC on November 26, 1990. Report due November 28, 1990.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2511 MWt rated core thermal power.

EVENT IDENTIFICATION: Unit Scram From IRM-13 and 16 High-High Due To Personnel Inattention.

A. CONDITIONS PRIOR TO EVENT:

Unit: Two Event Date: October 27, 1990 Event Time: 1559
Reactor Mode: 3 Mode Name: STARTUP/HOT STANDBY Power Level: 00%

This report was initiated by Deviation Report D-4-2-90-061

HOT STANDBY Mode (3) - In this position, the reactor protection scram trips, initiated by condenser low vacuum and main steamline isolation valve closure are bypassed, the low pressure main steamline isolation valve closure trip is bypassed and the reactor protection system is energized, with IRM and APRM neutron monitoring system trips and control rod withdrawal interlocks in service.

B. DESCRIPTION OF EVENT:

On October 27, 1990, at 1559 hours, Unit Two was in the STARTUP/HOT STANDBY mode at less than one percent of rated core thermal power. At this time, a reactor scram occurred as a result of Intermediate Range Monitors (IRM) [IG] 13 and 16 High-High signal.

The following is a discussion of the events leading up to and following the scram.

At 1900 hours, on October 26, 1990, load reduction began in accordance with Temporary Procedure 6303 as required for preparations for a Turbine [TRB] [TA] Torsional test. Temporary Procedure 6303 was a minor change to QGP 2-4, Shutdown from Power Operation to a Standby Hot Pressurized Condition, which allowed initial power reduction using recirculation flow and/or control rod insertion. The original QGP 2-4 only allowed initial power reduction using recirculation flow control. The purpose of the Turbine Torsional test was to verify the location of the torsional resonance vibrations on the main turbine. The test was performed with the reactor [RCT] at power but the turbine generator (TG) [TG] off the system grid [EL]. To accomplish the test, temporary alterations to the turbine Electro-hydraulic Control (EHC) [HCU] system [TG] were needed. In order to install the temporary alterations, reactor power and pressure were reduced to close the turbine bypass valves [V] which then allowed the EHC oil pumps [P] to be secured. The turbine test team consisted of Technical Staff, Operational Analysis Department and General Electric (GE) Company representatives. Operating Engineers were assigned to perform the duties of Test Director on each shift.

The turbine torsional test was being performed in accordance with Special Test Procedure 2-95 Partial B which received On-Site Review (OSR) and approval on October 26, 1990.

At 2345 hours, on October 26, 1990, the Unit Two drywell [NH] was deinerted and a drywell entry was made to manually insert IRM 16 because of drive problems.

At 0 2 hours, on October 27, 1990 the main turbine was taken off line and the Shift 1 Nuclear Station Operator (NSO) continued inserting rods to come to hot standby with all bypass valves closed and reactor pressure at 850-900 psig in accordance with Temporary Procedure 6303 and the current approved Control Rod Sequence. At 0240 hours, the mode switch was placed in the STARTUP/HOT STANDBY position. Control rod insertion continued until the turbine bypass valves were closed. At this point 4 control rods were inserted from position 12 to position 08. As power decreased, the IRM's were ranged from Range 8 to Range 6. At this time, the NSO withdrew one control rod from position 08 to position 10 and the IRM's began increasing and were ranged from Range 6 to Range 8 in order to maintain on scale readings per procedure. The NSO, SCRE, Test Director and control room extra NSO (CE) were monitoring reactor parameters closely during this evolution. They did recognize that the notch worth of the control rod withdrawn was significant but none of them considered the notch worth unusual for the current reactor conditions. At 0420 hours, with reactor power and pressure stable, the EHC pumps were secured. At 0900 hours, the Nuclear Engineer arrived on site to oversee calibration of the Average Power Range Monitors (APRM) [IG] prior to power increase for the performance of the turbine torsional test. At 1010 hours, following required temporary alterations to the EHC system, the EHC system was restarted and power increase was begun to establish the necessary reactor conditions for performing the test.

At 1226 hours, the Unit 2 turbine acceleration began at the medium startup rate. At 1323 hours, with turbine speed at 571 rpm and an acceleration rate of 3-4 rpm per minute, the main turbine was tripped because of difficulties in attaining the needed turbine acceleration required to perform the test. At 1445 hours, a conference call was held between the Shift Engineer (SE), Test Director, and Production Superintendent on turbine test difficulties.

At 1500 hours, the turbine torsional test was aborted due to the difficulties. The Nuclear Engineer left the site after the test was aborted. He discussed with the SCRE his plans to return later during power ascension. Operating shift 3 had assumed control room duties. Plant status was as follows: MODE SWITCH was in the STARTUP/HOT STANDBY position, reactor pressure was being maintained at approximately 920 psig by the turbine bypass valves (one and three quarters bypass valves were open), the IRM's were on Range 9 which is approximately 7 percent of rated core thermal power, IRM 12 was bypassed due to spiking, IRM 17 was bypassed due to a depleted detector, and Temporary Procedure 6303 was in effect.

At 1510 hours, the SE directed the SCRE, who instructed the NSO, to decrease and maintain reactor pressure at 800 psig. This order was given to accommodate the removal of the test instrumentation from the main turbine EHC system due to the Turbine Torsional test being aborted. The SCRE focused his attention on removing turbine test instrumentation, returning equipment to service, and re-inerting the drywell (The drywell inerting normally takes about 10 hours and was required by Technical Specifications to be completed by 0200 hours on October 28, 1990).

The NSO began inserting rods per the current approved Control Rod Sequence to reduce reactor power thereby reducing reactor pressure. His focus was on reducing reactor pressure. By 1540 hours, the NSO had completed 14 steps of the control rod sequence which involved 84 incremental control rod movements. All turbine bypass valves were closed by 1553 hours and reactor pressure was approximately 805 psig. The EHC pumps were then secured. The NSO and SCRE observed that reactor pressure was continuing to decrease and the SCRE directed the NSO to withdraw control rods to bring pressure to 800 psig.

At 1556 hours, reactor power was at IRM Range 1. A rod block signal was present because the IRMs were on Range 1 and the Source Range Monitors were not fully inserted (SRM) [IG] and their count rate was less than 100 counts per second (CPS). The SRMs were inserted and the rod block cleared at 1557 hours. The NSO began to withdraw rods at 1558 hours with a reactor pressure of 776 psig and decreasing. Four control rods were withdrawn one notch, from position 04 to 06. The SCRE approached the 902-5 panel [PL] just prior to the NSO withdrawing control rod G-7 from position 06 to 08. Control rod G-9 was then selected but not withdrawn. The reactor scrammed at 1559 hours due to the high neutron flux sensed by IRMs 13 and 16. The mode switch was subsequently placed in shutdown and procedure QGP 2-3, REACTOR SCRAM, was entered.

An Emergency Notification System (ENS) phone notification was completed at 1643 hours in accordance with 10CFR50.72(b)(2)(ii).

No other systems were known to be inoperable or degraded at the start of this event that could have contributed to the event.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with 10CFR50.73(a)(2)(iv): The licensee shall report any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

The primary cause of this event was personnel error. The licensed NSO at the time of the event focused on controlling reactor pressure and did not realize that the reactor was subcritical. Additionally, the NSO did not utilize the reactor physics knowledge expected of a licensed operator and did not follow all required procedural steps.

A contributing cause was insufficient management oversight of hot standby operation. The SCRE directed the NSO to insert control rods in order to reduce reactor pressure; however, did not directly supervise the NSOs control rod movements. The SCRE was involved with preparing to remove test instrumentation to recover from the Special Test and re-inerting the drywell to prevent a violation of Technical Specifications. The Test Director and Shift Engineer did not provide sufficient oversight during the transition from the test conditions to hot standby operation recovery.

Another contributing cause was ineffective communications. If the information regarding the earlier shift's experience with the control rod withdrawal and the resulting pressure increase had been communicated to the Shift 3 crew through shift turnover or the unit logs, they may have been more cautious during rod withdrawals. Due to a teleconference to discuss the discontinuation of the test and inerting the drywell, the Shift Engineer briefing was delayed and was not conducted prior to the scram.

The ineffective communications between the NSO and the SCRE contributed to this event.

Another contributing cause was attributed to insufficient training. Hot Standby condition, although covered in initial license training, was not selected for inclusion into the Operator Requalification Program since it is an infrequent evolution.

A final contributing cause was attributed to the On-Site Review process. The On-Site Review of the Special Test Procedure was limited in scope. The turbine torsional test was thoroughly reviewed, however, the review did not consider the infrequent operation of the plant in HOT STANDBY.

D. SAFETY ANALYSIS OF EVENT:

The following items have been addressed in the safety analysis of this event:

1. The performance of the nuclear instrumentation (IRMs)
2. The reactivity worth of the control rods that were withdrawn, and,
3. The core reactivity at the time of the event.

The purpose of the IRMs is to provide sufficient intermediate range flux level information under the worst permitted bypass and chamber failure conditions. The scaling arrangement in the IRM subsystem assures that for all unbypassed IRM channels, the scram and rod block trips are no more than a factor of 10 above the IRM level at the time. This assures that, should scram or rod block action be needed due to rapid or unintentional neutron flux increases, the trip signal will be generated before the flux increases by a factor greater than ten providing a conservative margin to fuel damage. During this event, the IRMs performed as expected. The minimum required operable IRMs were available for the Reactor Protection System. IRM 12 was bypassed on RPS Channel A and IRM 17 was bypassed on RPS B. Although IRM 16 had to be manually inserted, it was in place and operable.

An analysis was performed by the Nuclear Fuel Services Department on the reactivity worth of the control rods that resulted in the power increase. The analysis demonstrated that the worth of the control rods was not greater than expected but was sufficient to create the resulting power increase.

Given the additional notch withdrawal which was apparently taken with the reactor already on a positive period, the very short period which scrammed the unit on IRM high high flux is consistent with analytical results. It is also clearly bounded by the FSAR limiting reactivity insertion events at low power (Rod Drop Accident and the Rod Withdrawal Error event from just subcritical which was analyzed for the IRM design basis).

The onsite nuclear engineering group also verified that core reactivity was within the Technical Specification limits by evaluation of the critical rod pattern during the subsequent criticality. These analyses and verification demonstrate that the core is not behaving anomalously.

E. CORRECTIVE ACTIONS:

The immediate corrective actions included placing the mode switch in SHUTDOWN and initiating procedure QGP 2-3, REACTOR SCRAM.

An initial investigation team was formed to review the event. The Station Regulatory Assurance Supervisor was assigned the function of the team leader. Members of the investigation team included two (2) Regulatory Assurance personnel (one, an engineer who holds an active SRO license), the Lead Nuclear Engineer, the Assistant Superintendent for Operations (ASO) and the On-site Nuclear Safety Administrator (who reports to the Corporate Safety Assessment Department). The team began their investigation of the event at approximately 2230 hours on October 27, 1990. Interviews with the Shift Engineer, the SCRE, NSO and Nuclear Engineer were conducted. Shift logs and procedures were also reviewed as part of the investigation.

On October 28, 1990, the BWR Operations General Manager and the corporate Chief Nuclear Engineer reviewed the results of the Station's preliminary investigation and discussed it with NRC personnel from Region III and NRR on a conference call.

Following the completion of the preliminary investigation, the following corrective actions were completed:

1. An in-depth discussion was conducted with the members of the crew involved in the event. The discussion included members of upper Station management, the Chief Nuclear Engineer, and the BWR Operations General Manager. A presentation of the event sequence and the investigation results was provided to the crew members. The crew was requested to provide comments as to the accuracy of the facts surrounding the event. This action was completed on October 28, 1990.

2. Prior to assuming their shift duties, each operating crew was briefed on this event by upper Station management. During these briefings, the need for effective communication and the SCRE oversight function were stressed.

3. Additional management oversight (an Operating Engineer) was assigned to the crew involved in this event until remedial training was completed.

4. The crew involved in this event received remedial training on operating the unit in the HOT STANDBY mode. The training consisted of classroom and simulator training. The training included discussions of teamwork, communications and procedural compliance. This training was completed on Wednesday, October 31, 1990.

5. An independent and in-depth investigation of the event was conducted to augment the preliminary investigation conducted by the Station. The purpose of the augmented review was to ensure a thorough investigation of the event. The investigation reviewed the event and focused on procedures/procedural adherence, command and control of the Control Room, preparation for special tests, and communications, including operating logs and turnover. The conclusions reached by the corporate investigation will be reviewed for potential application to other CECo sites. The augmented review was completed by November 2, 1990.

The following corrective actions are to be completed:

1. Specific training on controlling reactor pressure under different operating conditions (especially when operating in hot standby) will be provided in the Licensed Operator Initial Training and Licensed Requalification Programs. The training will include reactivity management specific to this mode of operation. Training will be conducted for all licensed shift personnel and Nuclear Engineers during the next training cycle. This training will consist of both classroom and simulator and will include lessons learned from Quad Cities and LaSalle Station events. (NTS 2652009006101).

2. This event will be included in both the "Lessons Learned" portion of the operator retraining class as well as required reading. This training will stress the need for effective communication and command and control responsibilities which are expected during control room evolutions. (NTS 2652009006102).

3. Nuclear Engineers will be required to attend the "Lessons Learned" portion of the operator retraining class when this event is discussed. Also, Nuclear Engineers will be required to attend the simulator portion of the hot standby operation training. (NTS 2652009006103).

4. In the event that hot standby operation is required due to plant conditions, augmented management oversight will be provided to all crews. However, the Station does not intend to operate in the hot standby mode until training is completed. (NTS 2652009006104).

5. QGP 2-4 will be enhanced based on the lesson learned from this event. (NTS 2652009006105).

6. The personnel involved in this event will receive counseling on this event, their perceived errors, and how their performance could have been improved. (NTS 2652009006106)

7. A Qualified Nuclear Engineer will be present on startups until at least one bypass valve is opened and during hot standby conditions. (NTS 2652009006107)

8. Procedure QTP 010-4, "Preparation, Performance and Review of Special Operational Tests" will be reviewed and revised as needed to ensure the following are adequately addressed:

   a. identification of line of command and specific responsibilities if other than the normal shift operating organization is involved,

   b. review of procedures to be used to enter and exit test conditions,

   c. the need for special training or crew briefings,

   d. special shift turnover requirements if the test will take more than one shift, and

   e. requirements for procedure validation as specified in QCAP 1100-4. (NTS 2652009006108)

9. The training recommendations of the LaSalle County Station PSE Evaluation Report 90-20, Supplement #1, will be expedited. (NTS 2652009006109)

10. Nuclear Fuel Services will conduct an assessment to identify any potential weaknesses in training on reactivity management. Appropriate actions will be implemented in response to this review. (NTS 2652009006110)

11. A process for inclusion of infrequently performed tasks in the periodic training program will be developed. (NTS 2652009006111)

12. Expedite the proceduralization of the turnover checklists currently under development for the NSOs and SEs to improve shift to shift communications. (NTS 26520090061")

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				
Quad Cities Unit Two	0 | [illegible] | 0 | 2 | 6 | 5	9 | 0	-	0 | 1 | 1	-	0 | 0		0 | 9	OF	0 | 9

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

13. Stress the importance of logging conditions encountered which future shifts should be made aware. (NTS 2652009006113)

14. A committee made up of a cross section of station personnel will be established to address the issue of procedure adherence and what further actions can be undertaken. (NTS 2652009006114)

15. Expectations for control room communications will be reviewed and standards established. The ASO will issue these standards as an Operating Department policy document. (NTS 2652009006115)

F. <u>PREVIOUS EVENTS:</u>

A review of previous reactor scrams at Quad Cities, back to 1985, did not reveal an event similar to this one.

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

There was no component failure associated with this event.