Commonwealth Edison Company
Quad Cities Generating Station
22710 206th Avenue North
Cordova, IL 61242-9740
Tel 309-654-2241

ComEd

LWP 95-074

August 8, 1995

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

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

Subject: Licensee Event Report (LER) 265/94-010 Supplemental Information.

As stated in LER 265/94-010, supplemental information is being provided and is enclosed as Attachment 1. This information constitutes revision 01 to the original LER documentation.

Attachment 2 is a reproduction of the original text of LER 265/94-010.

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."

If there are any questions or comments concerning this letter, please refer them to Nick Chrissotimos, Regulatory Assurance Administrator at 309-654-2241, ext. 3100.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD CITIES NUCLEAR STATION

D. B. Cook for

L.W. Pearce
Station Manager

Attachment 1- LER Supplemental Information
Attachment 2- LER 265/94-010 (copy)

cc: J. Schrage
C. Miller
INPO Records Center
NRC Region III

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ComEd Quad Cities Station - System Engineering Department

To: L.W. Pearce

From: Paul Aitken BRWebb for 8/8/95
Sys Eng Supervisor

Prepared by: Daniel Semeter DJS 8-8-95
CRD System Engineer

Reviewed by: Jason Smith JTS 8/8/95
Lead Nuclear Engineer

Subject: NTS Item #2651809401003 Issue A Supplemental Report When the Results of the GE Analysis are received.

General Electric performed an analysis of two batches of Scram Solenoid Pilot Valve(SSPV) diaphragms. These batches were both manufactured in 1989, with one set having been in service since 1990 and the other set having been in service since 1992.

The testing results of the 1990 batch are as follows. The oxidative stability of these diaphragms was near the end of life. They degrade under temperature conditions of 169° C, compared to "good" diaphragms that would degrade at greater than 200° C. The spring constant of the Buna-N material for these diaphragms was not checked. This is because this batch had already developed hardening and cracks. Therefore, no useful information could be gained from this testing.

The testing results of 1992 batch are as follows. The diaphragms were flexible, but exhibited some localized hardening. The spring constant for these diaphragms was comparable to other diaphragms of similar vintage. The oxidative stability for these diaphragms showed that there was still some remaining service life. The diaphragms did not show degradation until 210° C.

The results of this testing showed that the diaphragms that had a service life of greater than three years had degraded enough to affect operability of the control rods. All of the SSPVs that had diaphragms with a service life of greater than three years were replaced with the new Viton elastomer SSPVs during Q1F35 and Q2F35. The diaphragms that had been in service since 1992 were left installed until Q2R13, when they were replaced with the new Viton elastomer SSPVs. All of the SSPVs on Unit 2 with the Buna-N elastomers have been replaced with the SSPVs with the new Viton elastomers. All of the control rods on Unit 2 were successfully scram timed. Unit One has 53 HCUs with the old style Buna-N elastomer SSPVs. These SSPVs have only been ins service since June of 1994. They will be removed during Q1R14, after 1.5 years of service life. The analysis has showed that three years of service life is acceptable. Following Q1R14, all of the Buna-N elastomer SSPVs will have been replaced in Unit 1.

There will be no change in the course of action and commitments indicated by LER 2-94-010 as a result of this testing. This item can be closed.

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

ATTACHMENT 2

GGC-94-119

September 28, 1994

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) 94-010, Revision 00, for Quad Cities Nuclear Power Plant 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.

The following commitments are being made by this letter:

An expansion of the sampling program of the diaphragms is being performed to gather more data. Material analysis will be performed by General Electric Co.

An accelerated repair/replacement program of all the suspect SSPV diaphragms on Unit 1 and 2 began September 28, 1994.

A supplement to this report will be issued when the results of the General Electric Co. analysis are received.

If there are any questions or comments concerning this letter, please refer them to Nick Chrissotimos, Regulatory Assurance Administrator at 309-654-2241, ext. 3100.

Respectfully,

COMMONWEALTH EDISON
QUAD CITIES NUCLEAR POWER STATION

LJ Tucker

for G.G. Campbell
Station Manager

GGC/TB/plm
Enclosure

cc: J. Schrage
C. Miller
INPO Records Center
NRC Region III

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 5	

Title (4)

Control Rod D-11 Unplanned Scram From Position 48 to 00 During Instrument Maintenance Surveillance

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 8	2 9	9 4	9 4	- 0 1 0 -	0 0	0 9	2 8	9 4		0 5 0 0 0

OPERATING MODE (9)	04	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
POWER LEVEL (10)	9 8	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	
Keven Floming, Regulatory Assurance, Ext. 2789	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
B	A A	S O L	A 6 1 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

ABSTRACT:

On 08-29-94, Unit 2 was in the run mode at 98% rated core thermal power. The Instrument Maintenance Department (IMD) was performing a monthly surveillance when a channel B 1/2 scram signal was intentionally inserted, and caused an unplanned scram of Control Rod Drive (CRD) [AA] D-11 from position 48 to 00.

The unplanned scram of CRD D-11 was due to instrument air leakage past the exhaust port diaphragm of the 117 scram solenoid pilot valve (SSPV) (SOL). This caused the 127 scram outlet valve to open when the channel B 1/2 scram signal was inserted.

Corrective actions included taking Hydraulic Control Unit (HCU) D-11 out-of-service, rebuilding the HCU SSPVs, and rebuilding the 127 scram outlet valve.

LER265\94\010.WPF

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PLANT AND SYSTEM IDENTIFICATION:

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

EVENT IDENTIFICATION: Control Rod D-11 Unplanned Scram from position 48 to 00 during Instrument Maintenance Surveillance.

A. CONDITIONS PRIOR TO EVENT:

Unit: Two	Event Date: August 29, 1994	Event Time: 0953
Reactor Mode: 4	Mode Name: Run	Power Level: 98

This report was initiated by Licensee Event Report 265\94-010.

RUN (4) - In this position the reactor system pressure is at or above 825 psig, and the reactor protection system is energized, with APRM protection and RBM interlocks in service (excluding the 15% high flux scram).

B. DESCRIPTION OF EVENT:

At approximately 0953 on 08-29-94, Unit 2 was in the run mode at 98% rated core thermal power. The Instrument Maintenance Department (IMD) was performing Monthly Low and Low-Low Reactor Water Level Analog Trip System Calibration and Functional Testing (QCIS 200-3).

Per procedure, QCIS 200-3, a 1/2 scram signal was inserted on Reactor Protection System (RPS) [JC] channel B. When the channel B 1/2 scram was received, alarm A-3, ROD DRIFT, annunciated on the 902-5 panel and an unplanned scram of Control Rod Drive (CRD) [AA] D-11 (14-43) from position 48 to 00 was observed. The channel B 1/2 scram was reset at approximately 0954.

An Operator and Shift Foreman (SF) were dispatched to the CRD D-11 Hydraulic Control Unit (HCU) to investigate. The operator checked the scram solenoid pilot valves (SSPV) for air leaks, their associated fuses to determine if they were blown, ensured the solenoids were energized by using a solenoid tester, and measured the scram exhaust line temperature with a thermography gun. No abnormalities were noted; however, a Nuclear Work Request (NWR) deficiency tag was found on the 127 scram outlet valve stating water was leaking by the valve seat.

Immediate corrective actions included notifying the Unit Supervisor (US), Shift Engineer (SE), a Qualified Nuclear Engineer (QNE), and the System Engineer. The IMD surveillance was also suspended pending re-authorization by the US.

The QNE determined no immediate control rod movements were required to adjust neutron flux profile.

At approximately 1230, CRD D-11 HCU was taken out-of-service for Operations and electrically disarmed. The IMD surveillance was re-started at approximately 1250.

NWR Q17779 was initiated to rebuild the 117 and 118 SSPVs. Problem Identification Form (PIF) 94-2140 was initiated to investigate the event. An existing work request, Q16701, addressed the repair of the 127 scram outlet valve.

An Emergency Notification System (ENS) notification of this event was completed at 1259 hours on August 29, 1994 to comply with the requirements of 10 CFR 50.72 (b) (2) (ii).

There were no systems or components inoperable at the beginning of this event which could have contributed to this event.

C. <u>CAUSE OF EVENT</u>:

This report is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv), which requires the reporting of any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF) [JE], including the RPS.

The following is a summary of conclusions and Causal Factors (C/F) relating to problems which may have influenced and/or contributed to equipment malfunctions.

C/F: Equipment Specification, Manufacture and Construction

The unplanned scram of CRD D-11 was due to instrument air leakage past the exhaust port diaphragm of the 117 SSPV. This caused the 127 scram outlet valve to open when a channel B 1/2 scram signal was inserted.

SSPV 117 and 118 were rebuilt under NWR Q17779. All four diaphragms were hard and brittle; one was found to have a tear approximately 180 degrees around the circumference. Two others had minor defects on the surface. CRD D-11 SSPVs were last rebuilt in March of 1990.

This type of material degradation has been identified in General Electric Company (G080) Rapid Information Communication Services Information Letter (RICSIL) NO. 069 dated May 2, 1994. SSPVs that may contain the suspect diaphragm kits have been identified. The diaphragms from a sample of the identified SSPVs have been examined and showed increased hardness.

Scram outlet valve 127 was rebuilt under NWR Q16701. It was determined the leakage did not contribute to this event.

D. <u>SAFETY ANALYSIS OF EVENT</u>:

1. The safety significance with having CRD D-11 inadvertently scram is considered minimal. Because the rod scrammed into "00", there were no control rod blade tip enhancement problems.

   The scram of D-11 provided negative reactivity. However, if the rod were required to scram on a full-core scram, it would have performed its function. Therefore, there was no shutdown margin concern for this event.

2. This failure mode of diaphragm degradation can also result in the failure of a control rod to scram on an individual scram signal. If a SSPV failure caused the failure of a control rod to scram, the backup scram valves would ensure the control rod would insert into the core on a manual or automatic full core scram signal.

- There have been two failures to scram since December 3, 1993. Actions have been implemented to expedite replacement of the suspect SSPV diaphragms to reduce the likelihood of further failures.

- All of the 1990 and 1991 vintage suspect SSPVs have recently experienced hot scram timing. Unit One scram timing was completed 9-2-94. Unit Two scram timing was completed 8-30-94. All of the CRDs with suspect SSPV diaphragms meet the Technical Specification scram time requirements.

- There has been no indication of an adverse trend of the overall Technical Specification average scram insertion times due to SSPV degradation.

- Based on a sample of diaphragms provided by Quad Cities, General Electric reports that the hardness of the Quad Cities SSPV diaphragms were not as severe as that reported by other plants experiencing scram anomalies. General Electric also reports that we should not expect to see a marked increase in the failure rate.

E. <u>CORRECTIVE ACTIONS COMPLETED</u>:

The immediate corrective actions involved the Operations Department sending an operator and Shift Foreman to visually inspect HCU D-11. The operator checked the SSPVs for air leaks, checked for blown fuses, ensured the solenoids were energized, and checked for flow in the scram exhaust line. No abnormalities were noted; however a deficiency tag was found on the 127 scram outlet valve stating water was leaking by the valve seat.

CRD D-11 was taken out-of-service and electrically disarmed.

The SSPVs, 117 and 118, were rebuilt under NWR Q17779; this included the replacement of the diaphragms. The scram outlet valve, 127, was rebuilt under NWR Q16701. This work was completed and CRD D-11 was successfully tested and declared operable on 09-04-94.

CORRECTIVE ACTIONS TO BE COMPLETED:

An expansion of the sampling program of the diaphragms is being performed to gather more data. Material analysis will be performed by General Electric Co. (NTS# 26518094001001).

An accelerated repair/replacement program of all the suspect SSPV diaphragms on Unit 1 and 2 began September 28, 1994. (NTS# 26518094001002).

A supplement to this report will be issued when the results of the General Electric Co. analysis are received. (NTS# 26518094001003).

F. PREVIOUS EVENTS:

General Electric Company RICSIL 069 Rev. 1 has identified this as an industry wide problem.

The Nuclear Tracking System data base listed one LER involving CRD drift or scram at Quad Cities Station for the period from January 1, 1989 to present.

- LER 254/94-001, dated 1-24-94, Rod Drift of Control Rod H-1 From Position 48 to 14 During Instrument Maintenance Surveillance

G. COMPONENT FAILURE DATA:

This event is reportable to the Nuclear Plant Reliability Data System (NPRDS).

The SSPVs are manufactured by the Automatic Switch Company (ASCo) (A610) with manufacturer part number HVA90-405-2J.

The rebuild kits for the ASCo valves are manufacturer part number 204-137.

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL
Quad Cities Nuclear Power Station

Reference Number:	Date: 8/11/95
Subject: Supplemental Report for LER 2/94-010	
Submitted by: Terry Barber	

FOR REVIEW:

	1. Safety Evaluations NOT involving an unreviewed safety question as defined in 10CFR50.59 for:
☐	a. Changes to procedures as described in the Safety Analysis Report.
☐	b. Changes to equipment or systems as described in the Safety Analysis Report.
☐	c. Tests or experiments NOT described in the Safety Analysis Report.
	2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
☐	a. Procedure changes.
☐	b. Equipment or system changes.
☐	c. Tests or experiments.
☐	3. Proposed changes to the Technical Specifications or Operating License.
☐	4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
☐	5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
X	6. All REPORTABLE EVENTS (LERs only).
☐	7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
☐	8. All changes to the Station Emergency Plan prior to implementation.
☐	9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

☐	10. Other OSR Items/Documents NOT addressed above.

This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.