Commonwealth Edison
Quad Cities Nuclear Power Station
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Cordova, Illinois 61242
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RLB-92-031

January 31, 1992

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) 92-002, 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)(ii). Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear plant being in a condition that was outside the design basis of the plant.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD CITIES NUCLEAR POWER STATION

Gary Spell for RLB
R. L. Bax
Station Manager

RLB/TB/plm

Enclosure

cc: J. Schrage
T. Taylor
INPO Records Center
NRC Region III

STMGR 276

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
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Title (4) Exceedance of Technical Specification Local Leak Rate Test Limit 0.6 La While Testing the Containment Purge Four Valve Volume 2-1601-21, 22, 55 and 56

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		
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OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
	2	___ 20.402(b)	___ 20.405(c)	___ 50.73(a)(2)(iv)	___ 73.71(b)
POWER LEVEL (10)	0 0 0	___ 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)	X 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)
David P. Kunzmann, LLRT Coordinator Ext. 2162	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
X	Yes (If yes, complete EXPECTED SUBMISSION DATE)	___ NO		0 8	0 1	9 2

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

ABSTRACT:

On January 1, 1992, Quad Cities Unit Two was shutdown for refueling and maintenance (Q2R11). On January 3, 1992 at 0715 hours while performing Local Leak Rate Testing (LLRT) of the containment purge four valve volume 2-1601-21, 22, 55 and 56, it was determined that the Technical Specification 3.7.A.2.a.2 limit of 293.75 SCFH (0.6 La) was exceeded.

An Emergency Notification System (ENS) phone call was completed on January 3, 1992 at 1029 (EST) hours in accordance with 10CFR50.72(b)(2)(i).

The cause of the excessive leakages will not be known until repairs have been completed. A supplemental report will be issued to address the causes and corrective actions taken to bring the combined leakage to within Technical Specification limits. This report is being submitted to comply with 10CFR50.73(a)(2)(ii).

PLANT AND SYSTEM IDENTIFICATION:

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

EVENT IDENTIFICATION: Exceedance of Technical Specification Local Leak Rate Test Limit 0.6 La While Testing The Containment Purge Four Valve Volume 2-1601-21, 22, 55 and 56.

A. CONDITIONS PRIOR TO EVENT:

Unit: Two Event Date: January 3, 1992 Event Time: 0715
Reactor Mode: 2 Mode Name: REFUEL Power Level: 00%

This report was initiated by Deviation Report D-4-02-92-02.

REFUEL (2) - In this position interlocks are established so that one control rod only can be withdrawn when flux amplifiers are set at the proper sensitivity level and refueling crane is not over the reactor. Also, the trip from the turbine control valves, turbine stop valves, main steam isolation valves, and condenser vacuum are bypassed. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

B. DESCRIPTION OF EVENT:

On January 1, 1992, Quad Cities Unit Two was shutdown for the end of cycle 11 for refueling and maintenance. On January 3, 1992 at 0715 hours while performing Local Leak Rate Testing (LLRT) of the containment purge four valve volume 2-1601-21, 22, 55, and 56, it was identified that the as found leakage was 1395 scfh. Therefore, the Technical Specification 3.7.A.2.a.2 limit of 293.75 SCFH (0.6 La) combined leakage from all valves [NH] [ISV] and penetrations [NH] [PEN] except Main Steam [SB] Isolation Valves [ISV] (MSIV), was exceeded.

On January 3, 1992 at 1029 (EST) hours an Emergency Notification System (ENS) phone call was completed in accordance with 10CFR50.72(b)(2)(i).

C. APPARENT CAUSE OF EVENT:

This report is being submitted to comply with the requirements of 10CFR50.73 (a)(2)(ii) which states that the licensee shall report any event or condition that resulted in the condition of the Nuclear Power Plant, including its principal safety barriers, being seriously degraded or that resulted in the Nuclear Plant being in a condition that was outside the design basis of the plant

The cause of the excessive leakages cannot be determined until repairs have been completed and the valves and components have been retested. A supplemental report documenting the repairs and corrective actions taken will be issued.

D. SAFETY ANALYSIS OF EVENT:

The safety consequences of the event were minimal since the maximum pathway leakages are used for comparison to the total allowable leakage for Type B and C tests (0.6 La), which is a conservative method. The summation of the maximum pathway leakage rates assumes that for each potential air leakage through primary containment [NH], the best valve or barrier fails and the leakage through the pathway equals the leakage of the worst valve or barrier.

The summation of the minimum pathway leakages, which yields a more realistic total leakage, is used for determining the acceptance of the Type A test results. This method assumes that the best valve or barrier for each pathway remains intact. Although not all of the required as found Type B and C testing is complete, the total as found minimum pathway leakage for the pathways already tested are still within the allowable limit (0.75La).

The minimum pathway leakage will not be known until the containment purge four valve volume can be re-tested after a modification is made to the test method. This modification will be completed during this current outage.

E. CORRECTIVE ACTIONS:

The cause of the excessive leakage will not be known until repairs have been completed. The immediate corrective action was to initiate work requests to repair 2-1601-21 (Drywell purge) valve. A supplemental report will be issued to document the causes and corrective actions taken to bring the combined and the individual leakages below Technical Specification limits (NTS #265 200 92 00201).

F. PREVIOUS EVENTS:

254/90-029	Leak rate from all valves and penetrations including MSIVS on Unit 1 exceeded the Technical Specification Limit.
265/90-003	Leak Rate from all valves and penetrations including MSIVs on Unit 2 exceeded the Technical Specification Limit.
254/89-014	Leak Rate from all valves and penetrations including MSIVs on Unit One in excess of Technical Specification Limit.

265/88-007 Leak rate from all valves and penetrations excluding MSIVs on Unit Two in excess of Technical Specification Limit.

254/87-016 Leak rate from all valves and penetrations excluding MSIVs on Unit One in excess of Technical Specification Limit.

265/86-014 Leak rate from all valves and penetrations exluding MSIVs on Unit Two in excess of Technical Specification Limit.

These are the most recent related events; other similar events have occurred prior to 1986.

G. COMPONENT FAILURE DATA:

Component failure data will not be available until repairs have been completed. This data will be included in a supplemental report which will be completed following valve repairs.