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Quad Cities Nuclear Power Station
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RLB-93-054

March 26, 1993

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) 93-007, 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)(B). 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

R. L. Bax
Station Manager

RLB/TB/plm

Enclosure:

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

IE22 1/1

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
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Title (4)

B Loop Main Steam Isolation Valves Exceeded Technical Specification Leakage Limits For Containment Isolation Valves.

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 3	0 7	9 3	9 3	0 0 7	0 0	0 3	2 7	9 3		0 5 0 0 0

OPERATING MODE (9): 1

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)	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)	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 Kunzmann, Technical Staff, 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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
X Yes (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 6	2 5	9 3

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

ABSTRACT:

During performance of combined Local Leak Rate Test of the "B" loop MSIV's, the leakage exceeded the 11.5 scfh Tech Spec limit of T.S. 3.7.A.2.e. The measured leakage was 89.32 scfh. Further testing will be done to identify the leakage path. A supplemental report will be issued when the cause of the leakage is identified and repairs have been completed.

This report is being submitted to comply with 10 CFR 50.73 (a)(2)(ii)(B).

DVR 392

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
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TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

PLANT AND SYSTEM IDENTIFICATION:

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

EVENT IDENTIFICATION: B loop main steam isolation valves exceeded technical specification leakage limits for containment isolation valves.

A. CONDITIONS PRIOR TO EVENT:

Unit: Two	Event Date: March 7, 1993	Event Time: 0922
Reactor Mode: 1	Mode Name: SHUTDOWN	Power Level: 0%

This report was initiated by Deviation Report D-4-2-93-019.

SHUTDOWN Mode (1) - In this position, a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection trip systems have been deenergized for 10 seconds prior to permissive for manual reset.

B. DESCRIPTION OF EVENT:

On March 7, 1993, Quad Cities Unit Two was shutdown for the cycle 12 refueling and maintenance outage. At 0922 hours, while Local Leak Rate Testing (LLRT), the "B" Main Steam Isolation Valves, it was determined that the measured combined leakage rate of 89.32 standard cubic feet per hour (scfh) was in excess of the individual MSIV Technical Specification (3.7.A.2.a.3) limit of 11.5 scfh.

An individual LLRT of the outboard MSIV AO-2-203-2B measured a rate of 76.7 scfh.

The inboard MSIV AO-2-203-1B measured a rate of 12.62 scfh. Both the combined test and the individual tests were performed using the more conservative method of pressure decay.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(ii)(B). 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 leakage cannot be determined until repairs have been completed and the valves have been retested. A supplemental report documenting these repairs and any corrective actions taken will be issued.

D. SAFETY ANALYSIS OF EVENT:

The safety consequences of this event were minimal since LLRT is a conservative method for quantifying containment leakage. It is realistic to expect the leakage to be equal to the lesser leakage of the two valves, which when tested with the more conservative pressure decay method only exceeded the Technical Specification requirement of 11.5 scfh with a leakage of 12.62 scfh.

E. CORRECTIVE ACTIONS:

No corrective actions have been taken at this time. A supplemental report will be issued which will document the corrective actions taken to bring the MSIV leakage below the required limit (NTS# 2652009301901).

F. PREVIOUS EVENTS:

254/89/014 Leak rate from all valves and penetrations including MSIVs on Unit 1 in excess of Technical Specification Limit.

G. COMPONENT FAILURE DATA:

Component failure data is not available at this time since repairs have not been completed. Failure data will be included in the Supplemental Report.