LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
QUAD-CITIES, NUCLEAR POWER STATION, UNIT 1	0 5 0 0 0 2 5 4	1 of 0 3

Title (4) Leak Rate for MSIV (Main Steam Isolation Valve) in excess of Technical Specification limit

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	8 6	8 6	0 0 2	0 0	0 1	3 0	8 6		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	
Nicos P. Digrindakis, Technical Staff Engineer Ext. 2158	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
X	J M	I S V	C 6 8 4	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
X Yes (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 5	0 1	8 6

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

On January 7, 1986, three Main Steam Isolation Valves [JM] leaked in excess of the 11.5 SCFH limit allowed by Technical Specification 3.7.A.2.a.3. The excessive leakage for valves AO-1-203-2A, AO-1-203-2B, and AO-1-203-2C were identified during Local Leak Rate Testing performed while Unit One was shutdown for the end of Cycle Eight Refueling and Maintenance outage. Exact causes for the excessive leakages have not been determined yet. Repairs will be completed and leakages will be brought to within Technical Specification requirements prior to the Unit Startup. A supplemental report will be issued at that time.

This report is being submitted to you in accordance with the requirements of 10 CFR 50.73(a)(2)(ii), which requires reporting of any event or condition that resulted in the condition of the nuclear power plant including its principle safety barrier, being seriously degraded.

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PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2511 MWt rated core thermal power. Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

IDENTIFICATION OF OCCURRENCE:

Leakage through Main Steam Isolation Valves AO-1-203-2A, AO-1-203-2B, and AO-1-203-2C is in excess of Technical Specification limit of 11.5 SCFH as determined by Local Leak Rate Testing.

Discovery Date: 01-07-86

Report Date: 01-30-86

This report was initiated by Deviation Report D-4-1-86-6

CONDITIONS PRIOR TO OCCURRENCE:

SHUTDOWN Mode(1) - Rx Power 0% - Unit Load 0 MWe

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.

DESCRIPTION OF OCCURRENCE:

On January 7, 1986, Unit One was in cold shutdown for the end of Cycle Eight Refueling and Maintenance Outage. At 1530 hours, while performing the Main Steam Isolation Valves [JM] Local Leak Rate Test, QTS 100-3, Valves AO-1-203-2A, AO-1-203-2B, and AO-1-203-2C were found to leak in excess of the 11.5 standard cubic feet per hour (SCFH) limit specified in Technical Specification 3.7.A.2.a.3. The leakage for the individual valves are as follows.

VALVE	LEAKAGE
AO-1-203-2A	292.7 SCFH
AO-1-203-2B	38.0 SCFH
AO-1-203-2C	82.9 SCFH

This report is submitted in accordance with the requirements of 10 CFR 50.73(a)(2)(ii).

APPARENT CAUSE OF OCCURRENCE:

The cause of these excessive leakages have not yet been determined. A supplemental report will be submitted following repair of the valves detailing the cause of the leakages.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					Page (3)		
		Year	///	Sequential Number	///	Revision Number			
Quad Cities Unit 1	0 5 0 0 0 2 5 4	8 6	-	0 0 2	-	0 0	0 3	OF	0 3

TEXT

ANALYSIS OF OCCURRENCE:

From the preliminary results of leak rate testing, leakages of the A, B, and C Steam lines are minimal due to the fact that the outboard valves AO-1-203-2A, AO-1-203-2B, and AO-1-203-2C were shown to have the majority of the leakage. Therefore, the safety implications of this occurrence were minimal.

CORRECTIVE ACTION:

No corrective action has been taken as of this date. All repairs will be completed prior to the Unit Startup, and the supplemental report will describe the repairs and the "as-left" leakages.

FAILURE DATA:

At Quad Cities Station there have been 46 Main Steam Isolation Valves that have failed Local Leak Rate Testing. The last failure occurred March 17, 1985.

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

NJK-86-25

January 30, 1986

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

Reference: Quad-Cities Nuclear Power Station
Docket Number 50-254, DPR-29, Unit One

Enclosed please find Licensee Event Report (LER) 86-002, Revision 00, for Quad-Cities Nuclear Power Station.

This report is submitted to you in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(ii), which requires reporting of any event or condition that resulted in the condition of the nuclear power plant, including its principle safety barrier, being seriously degraded.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

N. J. Kalivianakis
Station Manager

NJK/MSK/dak

Enclosure

cc: J. Wojnarowski
A. Madison
INPO Records Center
NRC Region III

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