NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Quad-Cities Nuclear Power Station, Unit One	0 5 0 0 0 2 5 4	1 OF 0 3

TITLE (4)
Unit One Main Steam Isolation Valves Failed Local Leak Rate Tests

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	1 6	8 4	8 4	— 0 0 4	— 0 0	0 4	0 6	8 4		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(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, NRC Form 366A)*
[ ]	20.405(a)(1)(iii)	[ ]	50.73(a)(2)(i)	[ ]	50.73(a)(2)(viii)(A)		
[ ]	20.405(a)(1)(iv)	XX	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
Alex Misak	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
X	S B	I S V	N 4 1 7	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

X YES *(If yes, complete EXPECTED SUBMISSION DATE)* [ ] NO

EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	0 9	0 1	8 4

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

On March 16. 1984, five Main Steam Isolation Valves on Unit One were found to leak in excess of the 11.5 SCFH limit given in Technical Specification 3.7.A.2.-a.3. The excessive leakages for valves AO 1-203-1B, AO 1-203-2B, AO 1-203-1C, AO 1-203-2C, and AO 1-203-2D were identified during Local Leak Rate Testing performed while Unit One was shutdown for End of Cycle Seven Refueling. Causes for the excessive leakages have not yet been determined. Repairs will be completed and leakages brought to within Technical Specification requirements prior to Cycle Eight startup. A supplemental report will be issued at that time.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Unit One Quad-Cities Nuclear Power Station	0 5 0 0 0 2 5 4	8 4	— 0 0 4 —	0 0	0 2 OF 0 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Event Description

At 1030 hours on March 16, 1984, Main Steam Isolation Valves AO 1-203-1B, AO 1-203-2B, AO 1-203-1C, AO 1-203-2C, and AO 1-203-2D 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. Unit One was shutdown for End of Cycle Seven Refueling and maintenance when the excessive leakages were discovered during the performance of Local Leak Rate Testing. The leakages for the individual valves are as follows:

Valve	Leakage
AO 1-203-1B	34.5 SCFH
AO 1-203-2B	70.3 SCFH
AO 1-203-1C	31.7 SCFH
AO 1-203-2C	416.7 SCFH
AO 1-203-2D	466.2 SCFH

For each line with excessively leaking valves, the inboard valve had the lower leakage and would limit the through-line leakage. The through-line leakage for each line is as follows:

Line	Through-Line Leakage
B	34.5 SCFH
C	31.7 SCFH
D	1.7 SCFH

The through-line leakage of the D line is 1.7 SCFH due to the fact that tests showed the AO 1-203-1D valve leaked 1.7 SCFH. Because this line's leakage is this small, the amount of leakage during an accident would not be excessive.

The through-line leakages for the B and C lines are greater than the 11.5 SCFH limit for any one valve. However, during testing, the inboard valves, which are the limiting valves in each case, are pressurized such that the pressure forces the disc away from the seat. During an accident, Reactor pressure would act in the opposite direction and force the disc into the seat tighter, thereby reducing the leakage to a value lower than what is indicated from the tests.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Unit One Quad-Cities Nuclear Power Station	0 5 0 0 0 2 5 4	8 4	— 0 0 4 —	0 0	0 3 OF 0 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Cause

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

Corrective Action

No corrective action has been taken as of this date. All repairs will be completed prior to unit startup, and a supplemental report will be submitted at that time describing the repairs and "as-left" leakages.

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

NJK-84-128

April 6, 1984

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 Number (LER) 84-004 for Quad-Cities Nuclear Power Station.

This report is submitted to you in accordance with the requirements of the Code of Federal Regulations, T tle 10, Part 50.73(a)(2)(ii), as a condition that resulted in the principle safety barriers being degraded.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

L. F. Gerner for
N. J. Kalivianakis
Station Superintendent

NJK:PDK/bb

Enclosure

cc B. Rybak
A. Morrongiello
INPO Records Center
NRC Region III

IE-22
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