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 1 5

TITLE (4)

Leak Rate From All Valves & Penetrations in Excess of Technical Specifications

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 3	0 7	8 4	8 4	— 0 0 2	— 0 1	0 9	0 4	8 4	NA	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 2

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)	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	
Alex L 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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	J M	P E N	C 3 1 0	Y	B	J M	P E N	C 3 1 0	Y
B	J M	P E N	C 3 1 0	Y	B	J M	P E N	C 3 1 0	Y

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) — X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

While performing Refueling Outage Local Leak Rate Testing, the measured combined leakage rate for all penetrations and valves, except Main Steam Isolation Valves, was found to leak in excess of 293.75 SCFH which is allowed by the plant Technical Specifications. This supplemental report documents repairs made to valves, hatches, and penetrations with unacceptable leak rates and the final results of the Local Leak Rate Testing program.

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NRC Form 366B
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LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

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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	
B	W K	I S V	C 6 8 4	Y							
B	W K	I S V	C 6 8 4	Y							
B	W K	I S V	C 6 8 4	Y							
B	W K	I S V	C 6 8 4	Y							
B	W K	I S V	C 6 8 4	Y							
B	W K	I S V	C 6 8 4	Y							
B	S J	I S V	C 6 8 4	Y							
B	S J	I S V	C 6 8 4	Y							
B	S J	I S V	C 6 8 4	Y							
B	S J	I S V	C 6 8 4	Y							
B	I K	I S V	C 6 8 4	Y							
B	I K	I S V	C 6 8 4	Y							
B	V A	I S V	P 3 4 0	Y							
B	B J	I S V	T 3 0 2	Y							

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Event Description

While performing the Unit One End of Cycle 7 Refueling Outage Local Leak Rate Testing program, the combined measured leakage for all penetrations and valves (JM) (except for MSIV's), that are subject to Type B and C tests, exceeded 0.60 L_A (293.75 SCFH at 48 psig). This limit is established by Technical Specification 3.7.A.2.c, and the testing requirements are established by 10 CFR 50, Appendix J.

The consequence of this occurrence is that it was necessary to repair a number of containment isolation valves and penetrations to bring the combined measured leak rate below the Technical Specification limit prior to resuming power operation. Exceeding the Technical Specification limit does not pose any significant risks or hazards to public safety because the total leakage determined by Type B and C tests does not, in any way, represent a probable leakage from the containment under accident conditions.

There are a number of factors which prevent totaling Type B and C test results to obtain a probable containment leakage. First, many of the Type C tests are performed by pressurizing the volume between isolation valves in series. While the Local Leak Rate Test (LLRT) does give the total leakage for both valves, the maximum (worst case) leakage one would expect from the containment would occur when both valves leak equally. Therefore, the probable containment leakage would be no more than half of the LLRT total for both valves, and, in fact, the leakage could be zero if all the LLRT measured leakage was through only one of the valves. Second, a number of Type C tests are performed on valves in series with other individually tested isolation valves. In this situation, the worst case probable containment leakage would be the minimum of the two LLRT results, not the total of the two. Third, there are also cases where the test boundary for the Type C test consists of three or more isolation valves. In this situation, if the LLRT result shows that only one valve repair is required, the LLRT result following the repair would be the worst possible leakage for any other valve on the boundary. Thus, the "as left" LLRT result would also be the "worst case" leakage from the containment prior to the repair. Fourth, Type B tests, which tests penetrations and double gasketed seals, test two sealing surfaces, one from the pressurized volume to the Primary Containment and another from the pressurized volume to the Secondary Containment. In this case, the "worst case" leakage would be half of the LLRT result.

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

Event Description (continued)

The "worst case" total leakage path calculation as described is still not a true measure of expected leakage during accident conditions. For example, a number of Type C tests are performed on systems which would, under most accident scenarios, be filled with water and pressurized (e.g. Reactor Feedwater and RHRS). These valves, while they may represent a substantial portion of the total measured leakage for Type B and C testing, would contribute nothing to a radiological release under most accident conditions.

Another situation that affects the "worst case" total leakage path calculation is the presence of accessible manual valves in series with isolation valves, such as those present on the HPCI and RCIC steam exhaust lines. If it was determined during an accident that an isolation valve of this type was not functioning properly, the manual valve could be closed to decrease the radioactive release to the Secondary Containment.

In order to show how the above considerations would affect the interpretation of the LLRT results, certain supporting documentation has been prepared. Appendix A of this report contains a listing of all Type B and C Tests performed on Unit One at the End of Cycle 7. For each Type B and C Test, the following information is provided: 1) The "as found" LLRT total measured leak rate; 2) A "worst case" estimate for containment leakage through the leakage path; 3) The "as left" LLRT total measured leak rate; and, 4) A "worst case" estimate of containment leakage following repairs. The "worst case" values for the Feedwater inlet lines take into consideration the fact that the lines are normally filled with water and pressurized and will contribute nothing to a radiological release. The "worst case" value for the HPCI steam exhaust line assumes that the manual isolation valve is closed.

As can be seen from the summary, exceeding the total measured leakage for Type B and C Tests as specified in Technical Specification 3.7.A.2.c, does not in itself demonstrate that the Primary Containment would have leaked more than its allowable limit of L_A (489.59 SCFH at 48 psig).

In addition to Primary Containment, other engineering safeguards are designed to mitigate the consequences of a radiological release during accident conditions. These systems are the Emergency Core Cooling System (ECCS), the Emergency Diesel Generators, the Secondary Containment, the Standby Gas Treatment System, and the Off-Gas "hold-up" piping and chimney. In the unlikely event that a radiological release should occur during an accident, the Quad-Cities Generating Station Emergency Plan has been found to be adequate for protecting public health and safety.

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

Cause

As a result of the Local Leak Rate Testing program, there were a number of valves, a Drywell penetration, and several hatches that were repaired during the End of Cycle 7 Refueling and Maintenance Outage. These items are listed in Appendix B along with a description of the work required. The types of repairs listed in Appendix B, with the exception of the Drywell penetration, are normal types of maintenance which is periodically required due to normal use and wear. Some of the system factors which contribute to necessary maintenance are: 1) Foreign material in piping fluid; 2) Size of valves (large valves tend to leak more than smaller valves); 3) Presence of high pressure steam which can cause steam cutting of disc and seating surfaces; 4) Frequency of valve operation; 5) Length of time since previous maintenance; and, 6) Valve and operator design.

The testable bellows on Core Spray piping Penetration X-16A was replaced during the refueling outage. While the leakage of this penetration (21.0 SCFH at 48 psig) only represented a leak of 4.3% of L_A, the leakage was abnormal for this type of penetration and replacement was deemed necessary. The cause of this leakage was a failure of the penetration bellows manufactured by Pathway Bellows Company.

Corrective Action

The immediate corrective action for the items requiring repair are listed in Appendix B. While all of the repairs performed would not have been required by the Technical Specification limit, repairs were made where leakage values showed deterioration from previous leak rate results.

The intent of the Local Leak Rate Test program is to determine where repairs are required to ensure preventative maintenance prior to containment degradation.

A Type A Test (Integrated Leak Rate Test) was performed near the end of the refueling outage. This test found the containment leakage to be 0.230 weight %/day. This result further demonstrates the integrity of the Primary Containment, and therefore, no further corrective action was deemed necessary.

APPENDIX A

LLRT RESULTS FOR UNIT ONE

WITH THROUGH LEAKAGES

BEFORE AND AFTER REPAIRS

LLRT DESCRIPTION	AS FOUND LEAK RATE	AS FOUND WORST CASE THROUGH LEAKAGE	AS LEFT LEAK RATE	AS LEFT WORST CASE THROUGH LEAKAGE	NOTES
'A' Main Steam Line	4.6	2.3	4.6	2.3	1
'B' Main Steam Line	104.8	34.5	11.8	2.3	3
'C' Main Steam Line	448.4	31.7	13.8	2.3	3
'D' Main Steam Line	467.9	1.7	1.2	0.0	3
Main Steam Line Drain	338.2	68.5	0.1	0.05	1,2
Primary Sample	0.0	0.0	0.0	0.0	
'A' Feedwater Inlet	Note 9 / Note 10	Note 11	3.6/33.4	Note 11	
'B' Feedwater Inlet	Note 9 / 563.8	Note 11	0.0/0.0	0.0	
RHRS to Radwaste	5.2/5.2	5.2	5.2/5.2	5.2	2
'A' RHRS Containment Spray	7.0	3.5	7.0	3.5	1
'A' RHRS Return	0.0	0.0	6.0	6.0	
'A' RHRS Supp Chamber Spray	3.0	1.5	3.0	1.5	1
'B' RHRS Containment Spray	1.5	0.75	1.5	0.75	1
'B' RHRS Return	0.0	0.0	4.6	4.6	
'B' RHRS Supp Chamber Spray	1.4	0.7	0.7	0.35	1
RHRS Shutdown Cooling Suction	3.1	1.55	3.1	1.55	1
RHRS Head Spray	0.0	0.0	0.4	0.0	2
Cleanup Suction	5.0	2.5	5.0	2.5	1
RCIC Steam Supply	0.1	0.05	0.4	0.2	1

LLRT DESCRIPTION	AS FOUND LEAK RATE	AS FOUND WORST CASE THROUGH LEAKAGE	AS LEFT LEAK RATE	AS LEFT WORST CASE THROUGH LEAKAGE	NOTES
RCIC Condensate Drain	3.0	3.0	3.0	3.0	
RCIC Turbine Exhaust	4.0	4.0	4.0	4.0	
Drywell & Supp Chamber Purge	2.1	1.05	2.1	1.05	1
'A' Supp Chamber Vent	0.0	0.0	0.0	0.0	
'B' Supp Chamber Vent	10.0	5.0	10.0	5.0	1
Drywell & Supp Chamber Supply Air Purge	1.8	0.9	1.8	0.9	1
Drywell & Supp Chamber Exhaust	81.0	27.0	27.0	13.5	1,6
Drywell Floor Drain Sump Discharge	75.6	37.8	0.3	0.15	1
Drywell Equipment Drain Sump Discharge	16.2	8.1	3.2	1.6	1,5
HPCI Steam Supply	1.2	0.6	0.0	0.0	
HPCI Condensate Return	0.0	0.0	0.0	0.0	
HPCI Steam Exhaust	Note 9	4.0	4.0	4.0	12
Drywell Pneumatic Suction	1.9/1.7	1.7	1.9/1.7	1.7	2
'A' Oxygen Analyzer Suction	0.0/0.0	0.0	0.0/0.0	0.0	
'B' Oxygen Analyzer Suction	0.6/0.7	0.6	0.6/0.7	0.6	2
'C' Oxygen Analyzer Suction	2.5/2.6	2.5	2.5/2.6	2.5	2
'D' Oxygen Analyzer Suction	1.5/0.6	0.6	1.5/0.6	0.6	2
Oxygen Analyzer Return	15.0/4.9	4.9	4.0/3.8	3.8	2

LLRT DESCRIPTION	AS FOUND LEAK RATE	AS FOUND WORST CASE THROUGH LEAKAGE	AS LEFT LEAK RATE	AS LEFT WORST CASE THROUGH LEAKAGE	NOTES
TIP #1	0.0	0.0	0.0	0.0	
TIP #2	0.0	0.0	0.0	0.0	
TIP #3	0.9	0.9	0.5	0.5	
TIP #4	0.0	0.0	0.7	0.7	
TIP #5	0.3	0.3	0.0	0.0	
TIP Purge	4.2	4.2	4.2	4.2	
ACAD (2599-2A, 23A)	0.3	0.15	0.3	0.15	1
ACAD (2599-2B, 23B)	2.2/0.1	0.1	2.2/0.1	0.1	2
ACAD (2599-3A, 24A)	1.3/0.0	0.0	1.3/0.0	0.0	2
ACAD (2599-3B, 24B)	5.0/0.0	0.0	5.0/0.0	0.0	2
ACAD (2599-4A, 5A)	0.6/0.0	0.0	0.6/0.0	0.0	2
ACAD (2599-4B, 5B)	2.1/1.5	1.5	2.1/1.5	1.5	2
CAM (2499-1A, 2A)	0.0	0.0	0.0	0.0	
CAM (2499-1B, 2B)	0.0	0.0	0.0	0.0	
CAM (2499-3A, 4A)	0.0/17.0	0.0	0.0	0.0	2
CAM (2499-3B, 4B)	0.0/18.5	0.0	0.0	0.0	2
X-7A	0.0	0.0	0.0	0.0	
X-7B	1.2	0.6	1.2	0.6	7
X-7C	0.0	0.0	0.0	0.0	
X-7D	0.0	0.0	0.0	0.0	

LLRT DESCRIPTION	AS FOUND LEAK RATE	AS FOUND WORST CASE THROUGH LEAKAGE	AS LEFT LEAK RATE	AS LEFT WORST CASE THROUGH LEAKAGE	NOTES
X-8	0.0	0.0	0.0	0.0	
X-9A	0.0	0.0	0.0	0.0	
X-9B	0.0	0.0	0.0	0.0	
X-10	0.1	0.05	0.1	0.05	7
X-11	0.3	0.15	0.3	0.15	7
X-12	6.0	3.0	6.0	3.0	7
X-2	8.7	4.35	8.7	4.35	13
X-13A	0.1	0.05	0.1	0.05	7
X-13B	0.0	0.0	0.0	0.0	
X-14	0.0	0.0	0.0	0.0	
X-23	1.8	0.9	1.8	0.9	7
X-24	0.0	0.0	0.0	0.0	
X-25	2.7	1.35	2.7	1.35	7
X-26	0.2	0.1	0.2	0.1	7
X-36	0.0	0.0	0.0	0.0	
X-47	0.0	0.0	0.0	0.0	
X-17	0.0	0.0	0.0	0.0	
X-16A	21.0	10.5	---	---	8
X-16B	8.0	4.0	8.0	4.0	7

LLRT DESCRIPTION	AS FOUND LEAK RATE	AS FOUND WORST CASE THROUGH LEAKAGE	AS LEFT LEAK RATE	AS LEFT WORST CASE THROUGH LEAKAGE	NOTES
X-100A	0.3	0.15	0.3	0.15	7
X-100B	0.0	0.0	0.0	0.0	
X-100C	0.0	0.0	0.0	0.0	
X-100D	0.0	0.0	0.0	0.0	
X-100E	0.0	0.0	0.0	0.0	
X-100F	0.0	0.0	0.0	0.0	
X-100G	0.0	0.0	0.0	0.0	
X-101A	0.3	0.15	0.3	0.15	7
X-101B	0.3	0.15	0.3	0.15	7
X-101D	0.0	0.0	0.0	0.0	
X-102A	0.3	0.15	0.3	0.15	7
X-103	0.0	0.0	0.0	0.0	
X-104B	0.0	0.0	0.0	0.0	
X-104C	0.0	0.0	0.0	0.0	
X-104F	0.0	0.0	0.0	0.0	
X-105A	0.0	0.0	0.0	0.0	
X-105B	0.0	0.0	0.0	0.0	
X-105C	0.0	0.0	0.0	0.0	
X-105D	0.4	0.2	0.4	0.2	7

LLRT DESCRIPTION	AS FOUND LEAK RATE	AS FOUND WORST CASE THROUGH LEAKAGE	AS LEFT LEAK RATE	AS LEFT WORST CASE THROUGH LEAKAGE	NOTES
X-107A	0.0	0.0	0.0	0.0	
X-227A	0.0	0.0	0.0	0.0	
X-227B	0.0	0.0	0.0	0.0	
X-1	0.0	0.0	0.0	0.0	
X-6	0.0	0.0	0.0	0.0	
X-4	105.0	52.5	0.0	0.0	7
X-35A	0.0	0.0	0.0	0.0	
X-35B	0.0	0.0	0.0	0.0	
X-35C	0.0	0.0	0.0	0.0	
X-35D	0.0	0.0	0.0	0.0	
X-35E	0.0	0.0	0.0	0.0	
X-35F	0.0	0.0	0.0	0.0	
X-35G	0.0	0.0	0.0	0.0	
X-200A	0.0	0.0	0.0	0.0	
X-200B	0.0	0.0	0.0	0.0	
Drywell Head Flange	30.0	15.0	0.0	0.0	7
SL-1	83.3	41.65	0.0	0.0	7
SL-2	2.7	1.35	0.0	0.0	7
SL-3	5.0	2.5	0.0	0.0	7

LLRT DESCRIPTION	AS FOUND LEAK RATE	AS FOUND WORST CASE THROUGH LEAKAGE	AS LEFT LEAK RATE	AS LEFT WORST CASE THROUGH LEAKAGE	NOTES
SL-4	0.0	0.0	0.0	0.0	
SL-5	0.5	0.25	0.0	0.0	7
SL-6	0.0	0.0	0.0	0.0	
SL-7	0.0	0.0	0.0	0.0	
SL-8	6.0	3.0	0.0	0.0	7
TOTAL	2524.3*	405.45	235.1	98.0	

*Does not include values for CV 1-220-58A, CV 1-220-58B, CV 1-220-62A, and CV 1-2301-45.

Notes

1. For the LLRT shown, the two isolation valves in series were tested by pressurizing the volume between the valves with the volumes external to the test volume drained and vented. The "worst case" through leakage from the containment based on the LLRT result would be one-half of the LLRT result.

2. The two isolation valves in series were tested individually. The "worst case" through leakage from the containment would be the minimum of the two LLRT results.

3. The Main Steam Isolation Valves were tested by pressurizing between the two valves. The result of this test is the "as found leak rate". The outboard valves of the B, C, and D lines were then tested individually. These results were then subtracted from the respective "as found leak rate" to determine a leakage for each valve individually. The "worst case" through leakage is, therefore, the minimum value of the two individual valve leakages for each line.

4. The Feedwater Check Valves were modified by M-4-1-80-27 to add additional hold down clamps to the seat. The valves were retested following the modification work and other repairs as described in Appendix B.

5. The isolation valves on the Drywell Equipment Drain Sump Pump Discharge line were modified by M-4-1-83-19. This modification rotated the valves so the operators are above the valves instead of below the valves. This modification should improve valve reliability based on these valve's maintenance experience. The valves were retested following the modification.

6. The Drywell and Suppression Chamber Exhaust LLRT consists of pressurizing between six containment isolation valves. Leakage through two of the valves is required to have a containment leakage path. The "as found" LLRT result was 81.0 SCFH. A visual inspection of the AO 1-1601-24 valve showed leakage from around the disc operating shaft. The packing around the shaft was replaced, and no repairs were required for any other valves. The volume was retested, and the LLRT result was 27.0 SCFH. The latter test verifies that no other valve in the volume boundary could leak more than 27.0 SCFH, so this value is used as the "worst case" as found through leakage. Since two valves must leak to form a leakage path from the containment, the possible leakage following the repair cannot exceed half of the LLRT result, or 13.5 SCFH.

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

Notes (continued)

7. A testable penetration or double gasketed seal represents a test of two sealing boundaries. The leakage can be from the pressurized volume to the containment, or it can be from the pressurized volume to the outside of the containment. Therefore, the "worst case" through leakage is half of the LLRT result.

8. The bellows on the X-16A Penetration was replaced and modification tested during the Type A Test as required by 10 CFR 50, Appendix J, Section IV.A.

9. The volume defined by the valve and a manual isolation valve could not be pressurized during the "as found" LLRT.

10. The valve was disassembled prior to the performance of the "as found" LLRT. This has been documented in Deviation Investigation Report D-4-1-84-23.

11. The "worst case" through leakage for the Feedwater Check Valves is assumed to be 0.0 SCFH because the Feedwater lines will be full of water and pressurized during most accident scenarios.

12. The "worst case" through leakage for the High Pressure Coolant Injection (HPCI) steam exhaust line is assumed to be 4.0 SCFH because of the manual isolation valve which would be accessible during accident conditions. No repairs were made to this valve, and the volume between it and the HPCI Steam Exhaust Check Valve had a final leak rate of 4.0 SCFH. Thus, with the manual valve closed, the containment leakage from the volume would be no more than 4.0 SCFH.

13. The leakages shown for X-2 (Drywell Personnel Interlock) are the values after conversion to PA (48 psig). The test is performed at 10 psig as allowed by Technical Specification 3.7.A.2.d. The conversion ratio used is from the Laminar Flow Model (Ref. ORNL-NSIC-5, Oak Ridge National Laboratory, Aug. 1965). The through leakages are half the LLRT result.

APPENDIX B

SUMMARY OF PRIMARY CONTAINMENT

ISOLATION VALVE REPAIRS

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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Repaired Item	Description of Repairs
MO 1-220-1	Replaced valve body and valve stem
MO 1-220-2	Lapped seat and disc; rebuilt operator
AO 1-1601-24	Disc operating shaft packing replaced
AO 1-2001-3	Lapped seat and disc
AO 1-2001-4	Lapped seat and disc; adjusted stroke
AO 1-2001-15	Rotated operator (Modification M-4-1-83-19); lapped seat and disc
AO 1-2001-16	Rotated operator (Modification M-4-1-83-19); lapped seat and disc; adjusted stroke
CV 1-2301-45	Replaced Check Valve like-for-like
AO 8803	Lapped seat and disc and repacked
AO 8804	Lapped seat and disc and repacked
CV 1-220-58A	Installed additional seat hold down clamps (Modification M-4-1-80-27)
CV 1-220-62A	Installed additional seat hold down clamps; rebuilt valve
CV 1-220-58B	Installed additional seat hold down clamps; replaced disc and seat
CV 1-220-62B	Installed additional seat hold down clamps; replaced disc and seat
X-4	Replaced gaskets
Drywell Head Flange	Replaced gaskets
Shear Lug Inspection Hatches	Replaced "O"-Rings
X-16A	Replaced existing double-ply bellows with a single-ply bellows. Test cover to be installed at a later date.

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

NJK-84-263

September 4, 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 (LER) 84-002, Revision 1, for Quad-Cities Nuclear Power Station.

This supplemental report is submitted to you in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(ii), to document the causes and corrective actions taken regarding our leak rate from the Local Leak Rate Testing program exceeding the Technical Specifications.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

N. J. Kalivianakis

N. J. Kalivianakis
Station Superintendent

NJK:DBC/bb

Enclosure

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

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