NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 Licensee Code (2)	I L Q A D 2
License Number (3)	0 0 0 - 0 0 0 - 0 0 0
License Type (4)	4 1 1 1 1
CAT (5)	
CON'T 01 Report Source (6)	L
Docket Number (7)	0 5 0 0 0 2 6 5
Event Date (8)	0 9 1 2 8 3
Report Date (9)	0 3 1 9 8 4

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 On September 12, 1983, while performing refueling outage Local Leak Rate Testing,
03 the total measured combined leakage rate for all penetrations and valves, except
04 Main Steam Isolation Valves, was found to exceed 293.75 SCFH (0.60 L_a). Repairs
05 were performed during the refueling outage to give a total measured leak rate (less
06 MSIV's) of 238.15 SCFH (0.486 wt %/day).
07
08

09	System Code (11)	Cause Code (12)	Cause Subcode (13)	Component Code (14)	Comp. Subcode (15)	Valve Subcode (16)
	S A	E	B	V A L V E X	X	X

(17) LER/RO Report Number	Event Year		Sequential Report No.		Occurrence Code	Report Type		Revision No.
	8 3	—	0 1 5	/	0 3	L	—	1

Action Taken (18)	Future Action (19)	Effect on Plant (20)	Shutdown Method (21)	Hours (22)	Attachment Submitted (23)	NPRD-4 Form Sub. (24)	Prime Comp. Supplier (25)	Component Manufacturer (26)
B	Z	Z	Z	0 0 0 0	Y	Y	N	Z 9 9 9

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 See Causes and Corrective Actions for valves and penetrations with excessive leakage
11 in the attached report. Also, reference Appendix C of the attached report for
12 specific repairs required by the Refueling Outage Local Leak Rate Testing program.
13
14

15	Facility Status (28): H	% Power (29): 0 0 0	Other Status (30): NA	Method of Discovery (31): B	Discovery Description (32): Local Leak Rate Testing
16	Activity Released (33): Z	Content of Release (34): Z	Amount of Activity (35): NA		Location of Release (36): NA
17	Personnel Exposures Number (37): 0 0 0	Type (38): Z	Description (39): NA		
18	Personnel Injuries Number (40): 0 0 0	Description (41): NA			
19	Loss of or Damage to Facility Type (42): Z	Description (43): NA			
20	Publicity Issued (44): N	Description (45): NA			

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GPO 917-926

ID/TS8-T

I. LER NUMBER: 83-15/03L-1

II. LICENSEE NAME: Commonwealth Edison Company
Quad Cities Nuclear Power Station

III. FACILITY NAME: Unit Two

IV. DOCKET NUMBER: 50-254

V. EVENT DESCRIPTION:

While performing the Unit Two end of Cycle 6 Refueling Outage Local Leak Rate Testing program, the measured combined leakage rate for all penetrations and valves (except for MSIV's) that are subject to Type B and C tests exceeded 0.60 L_a (293.75 SCFH at 48 PSIG). The limit is established by Technical Specification 3.7.A.2.c. and the testing requirements are established by 10 CFR 50, Appendix J.

VI. PROBABLE CONSEQUENCES OF THE OCCURRENCE

The consequence of this occurrence is that it was necessary to repair a number of containment isolation valves to bring the total combined measured leak rate below the technical specification limit prior to resuming power operations. Exceeding the limit does not pose any significant risks or hazards to public safety because the total for all 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 that prevent totaling Type B and C test results to obtain a probable containment leakage. For example, there are a number of Type C tests that are performed by pressurizing between isolation valves in series. While the local leak rate test (LLRT) does give a total leakage of both valves, the maximum (worst case) leakage one would expect from the containment is when both valves leak equally. Therefore, the probable containment leakage would be no more than 1/2 the LLRT result and could, in fact, be nothing at all if the leakage during the LLRT was from only one of the valves. There are a number of Type C tests performed on valves in series with other individually tested isolation valves. For leakage paths where the valves can be tested in this manner, the worst case leakage from the containment would be the minimum of the two LLRT results, not the total of both tests. There are also cases where the test boundary is defined by three or more isolation valves. If as a result of the LLRT only one valve requires repair, but leakage through at least two of the valves is required to form a leakage path from the containment, then the LLRT result after the single valve repair would represent a worst possible leakage for any other valve on the boundary. In this case the "as left" LLRT result would also be the "worst case" leakage from the containment prior to the repair. There are also Type B tests where the LLRT result exceeds the worst case expected containment leakage. An example would be when the LLRT pressurizes a double gasketed seal or penetration. The LLRT gives a total leakage for both sealing

surfaces, (one from the pressurized volume to the containment and a second from the pressurized volume to the secondary containment) while the primary containment through leakage would be some value between no leakage and 1/2 the LLRT result.

In order to show how just the above considerations would change how the LLRT results could be interpreted, certain supporting documentation has been prepared. Appendix A of this report contains a listing of all Type B and C tests performed on Unit Two at the end of Cycle 6. 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 following repairs (if any), and 4) a "worst case" estimate of containment leakage following repairs. Appendix B provides a summary for all Type B and C testing. As is readily apparent from the summary, exceeding the limit on total measured leakage for all Type B and C tests, except MSIV's, 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).

There are also a number of reasons why the "worst case" total leakage path calculation in Appendix B would be misleading in predicting a containment leakage during an accident. For example, there are a number of Type C tests performed on containment isolation valves that are for systems that would, under most accident scenarios, be water filled and pressurized (e.g. reactor feedwater inlet and RHRS). These valves, while they may represent a substantial portion of the total measured leakage for all Type B and C tests, would contribute nothing to a radiological release under most accident conditions.

Other engineering safeguards, besides primary containment, that are designed to mitigate the consequences of a radiological release during accident conditions are the redundant emergency core cooling systems (ECCS), the emergency diesel generators, the Secondary Containment building, standby gas treatment systems (SBGTS), 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 (GSEP) has been found to be adequate for protecting the public health and safety.

VII. CAUSE

As a result of the Local Leak Rate Testing program there were a number of valves and a drywell penetration that were repaired during the Cycle 6 refueling outage. These items are listed in Appendix C with the work request number and a description of the necessary repairs. The types of repairs listed in Appendix C, with the exception of the drywell penetration, are normal types of maintenance that is periodically required for these valves due to normal use and wear. Some of the system factors that contribute to necessary valve maintenance are: 1) the quantity of foreign material present in the piping fluid, 2) size of the valve (large valves tend to leak more than small valves), 3) presence of high pressure steam, which can cause steam cutting of seat or disc surfaces, 4) frequency of valve operation or other operating

conditions, 5) length of time since the last maintenance on the valve, and 6) the design of the valve and operator.

The testable bellows on core spray piping penetration X-16B was replaced during the refueling outage. While the leakage of this penetration (19.0 SCFH at 48 PSIG) only represented a leak of 3.9% of L_a, (and a worst case through leakage of 1/2 this amount) the leakage is abnormal for this type of penetration. The cause of this leakage is a failure of the penetration bellows manufactured by Pathway Bellows Company. This was the first failure of its type at Quad Cities Station. Following replacement of the bellows it will be analyzed to determine the mode of failure.

VIII. CORRECTIVE ACTION

The immediate corrective action for the items requiring repair are listed in Appendix C. While all of the repairs listed would not have been required by the Technical Specification limit, repairs were made where leakage valves showed a deterioration from past leak rate results. For example, the bellows on the core spray penetration X-16B was replaced. Attempts were first made to weld repair pinhole leaks in the outer surface of the bellows but were unsuccessful. Then the bellows was pressurized with Helium and a detector was used to determine if the inner surface of the bellows also leaked. Because Helium was detected inside the drywell, indicating some through leakage, the decision was made to replace the bellows.

The intent of the Local Leak Rate Testing program is to allow an operating margin to the Technical Specification limit (total combined measured leak rate for all Type B and C test equaled 238.15 SCFH or 0.486 of L_a following all outage repairs), and to target repairs that will insure preventative maintenance prior to containment deterioration. The success of the program is demonstrated in the relatively low through leakage total found in Appendix B.

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 0.385 weight %/day. This result further demonstrates the integrity of the primary containment. Therefore, no further corrective action is deemed necessary at this time.

ID/TS8-U

APPENDIX A

LLRT RESULTS FOR UNIT TWO

WITH THROUGH LEAKAGES

BEFORE AND AFTER REPAIRS

FALL, 1983

LLRT RESULTS AND THROUGH
LEAKAGE ESTIMATES - 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	43.8	9.24	9.24/2.30	2.30	1
'B' MSL	4.60	2.30	4.60	2.30	2
'C' MSL	2.30	1.15	2.30	1.15	2
'D' MSL	9.22	4.61	9.22	4.61	2
MSL Drain	39.27	19.64	7.46	3.73	2
Primary Sample	0.35	0.18	0.35	0.18	2
'A' Feedwater Checks	267.3/2.7	2.70	0.52/5.26	0.52	3,4
'B' Feedwater Checks	1.0/362.0	1.0	25.80/28.50	25.80	3,4
RHR to Radwaste	0.00	0.00	0.00	0.00	
'A' Drywell Spray	3.08	1.54	3.08	1.54	2
'A' RHR Return	0.00	0.00	0.00	0.00	
'A' Torus Cooling/ Spray	0.00	0.00	0.00	0.00	
'B' Drywell Spray	7.88	3.94	7.88	3.94	2
'B' RHR Return	2.07	2.07	2.07	2.07	
'B' Torus Cooling/ Spray	2.4/163.83	2.40	1.20	0.60	5/2
Shutdown Cooling	0.00	0.00	0.00	0.00	
Head Spray	0.38	0.19	0.38	0.19	2
Cleanup Suction	2.61/1.76	1.76	2.39	1.20	3/2
RCIC Steam Supply	20.40	4.90	3.35	1.68	6/2
RCIC Vac. Pump Ex.	5.03	5.03	5.03	5.03	
RCIC Turbine Ex.	24.90	24.90	5.78	5.78	
DW/Torus Purge	68.10	24.80	24.80	12.40	7
'A' Torus Vent	6.70	3.35	6.70	3.35	2
'B' Torus Vent	7.10	3.55	7.10	3.55	2
DW/Torus Supply Purge	0.00	0.00	0.00	0.00	
DW/Torus Ex. Purge	9.00	4.50	9.00	4.50	2
DW Floor Drain Sump	8.10	4.05	2.81	1.41	2,8
DW Eq. Drain Sump	0.02	0.01	7.35	3.68	2,8
HPCI Steam Supply	2.30	1.15	2.30	1.15	2
HPCI Drain Pot	6.76	6.76	6.10	6.10	9
HPCI Steam Ex.	0.00	0.00	0.00	0.00	
DW Pneumatic (4720)	0.00	0.00	0.00	0.00	
DW Pneumatic (4721)	0.00		0.00		3
O_2 Anal. (8801A)	0.00	0.00	0.00	0.00	
O_2 Anal. (8802A)	0.00		0.00		3
O_2 Anal. (8801B)	0.00	0.00	0.00	0.00	
O_2 Anal. (8802B)	0.00		0.00		3
O_2 Anal. (8801C)	6.00	6.00	6.00		3
O_2 Anal. (8802C)	16.00		1.40	1.40	3
O_2 Anal. (8801D)	0.00	0.00	0.00	0.00	
O_2 Anal. (8802D)	0.00		0.00		3
O_2 Anal. (8803,4)	0.60/10.00	0.60	0.60/10.00	0.60	3

LIRT DESCRIPTION	AS FOUND LEAK RATE	AS FOUND WORST CASE THROUGH LEAKAGE	AS LEFT LEAK RATE	AS LEFT WORST CASE THROUGH LEAKAGE	NOTES
TIP-1	0.25	0.25	0.25	0.25	
TIP-2	0.40	0.40	0.40	0.40	
TIP-3	0.70	0.70	0.70	0.70	
TIP-4	0.00	0.00	0.00	0.00	
TIP-5	4.50	4.50	4.50	4.50	
TIP-Purge Check	7.50	7.50	7.50	7.50	
CAM (2499-1A,2A)	0.00	0.00	0.00	0.00	
CAM (2499-3A,4A)	0.00	0.00	0.00	0.00	
CAM (2499-1B,2B)	0.00	0.00	0.00	0.00	
CAM (2499-3B,4B)	0.00	0.00	0.00	0.00	
ACAD (2599-2A,23A)	5.00/0.00	0.00	5.00/0.00	0.00	3
ACAD (2599-3A,24A)	0.90/15.10	0.90	0.30/0.00	0.00	3
ACAD (2599-2B,23B)	1.50/4.00	1.50	1.50/4.00	1.500	3
ACAD (2599-3B,24B)	0.00	0.00	0.00	0.00	
ACAD (2599-4A,5A)	1.80	0.90	1.80	0.90	2
ACAD (2599-4B,5B)	6.50	3.25	6.50	3.25	2
SRM/IRM Purge	0.00	0.00	0.00	0.00	
X-1	0.00	0.00	0.00	0.00	
X-2	62.76	31.38	3.44	1.72	10,11
X-4	0.00	0.00	0.00	0.00	
X-6	0.00	0.00	0.00	0.00	
X-35A	0.00	0.00	0.00	0.00	
X-35B	0.00	0.00	0.00	0.00	
X-35C	0.00	0.00	0.00	0.00	
X-35D	0.00	0.00	0.00	0.00	
X-35E	0.00	0.00	0.00	0.00	
X-35F	0.00	0.00	0.00	0.00	
X-35G	0.00	0.00	0.00	0.00	
X-200A	0.00	0.00	0.00	0.00	
X-200B	0.00	0.00	0.00	0.00	
Drywell Head	7.50	3.75	0.00	0.00	11
SL-1	0.05	0.03	0.05	0.03	11
SL-2	0.00	0.00	0.00	0.00	
SL-3	0.00	0.00	0.00	0.00	
SL-4	0.00	0.00	0.00	0.00	
SL-5	0.00	0.00	0.00	0.00	
SL-6	0.00	0.00	0.00	0.00	
SL-7	0.00	0.00	0.00	0.00	
SL-8	0.00	0.00	0.00	0.00	
X-7A	0.00	0.00	0.00	0.00	
X-7B	0.00	0.00	0.00	0.00	
X-7C	0.00	0.00	0.00	0.00	
X-7D	0.00	0.00	0.00	0.00	
X-8	0.00	0.00	0.00	0.00	
X-9A	0.00	0.00	0.00	0.00	
X-9B	0.55	0.28	0.55	0.28	11
X-10	0.00	0.00	0.00	0.00	
X-11	0.00	0.00	0.00	0.00	
X-12	3.20	1.60	3.20	1.60	11
X-13A	0.00	0.00	0.00	0.00	

LLRT DESCRIPTION	AS FOUND LEAK RATE	AS FOUND WORST CASE THROUGH LEAKAGE	AS LEFT LEAK RATE	AS LEFT WORST CASE THROUGH LEAKAGE	NOTES
X-13B	0.00	0.00	0.00	0.00	
X-14	0.95	0.48	0.95	0.48	11
X-23	0.00	0.00	0.00	0.00	
X-24	0.00	0.00	0.00	0.00	
X-25	1.95	0.98	1.95	0.98	11
X-26	1.40	0.70	1.40	0.70	11
X-36	0.00	0.00	0.00	0.00	
X-47	0.00	0.00	0.00	0.00	
X-17	1.40	0.70	1.40	0.70	11
X-16A	6.00	3.00	6.00	3.00	11
X-16B	19.00	9.50			11,12
X-100B	0.05	0.03	0.05	0.03	11
X-100C	0.20	0.10	0.20	0.10	11
X-100E	0.20	0.10	0.20	0.10	11
X-100F	0.00	0.00	0.00	0.00	
X-100G	0.35	0.18	0.35	0.18	11
X-101A	0.20	0.10	0.20	0.10	11
X-101B	0.20	0.10	0.20	0.10	11
X-101D	0.20	0.10	0.20	0.10	11
X-102B	0.15	0.08	0.15	0.08	11
X-103	0.25	0.13	0.25	0.13	11
X-104A	0.00	0.00	0.00	0.00	
X-104B	0.00	0.00	0.00	0.00	
X-104C	0.25	0.13	0.25	0.13	11
X-104D	0.25	0.13	0.25	0.13	11
X-104F	0.15	0.08	0.15	0.08	11
X-105C	0.25	0.13	0.25	0.13	11
X-106A	0.05	0.03	0.05	0.03	11
X-106B	0.25	0.13	0.25	0.13	11
X-107A	0.25	0.13	0.25	0.13	11
X-107B	0.30	0.15	0.30	0.15	11
X-227A	0.00	0.00	0.00	0.00	
X-227B	0.00	0.00	0.00	0.00	

NOTES FOR APPENDIX A

1. The total leakage of both valves when pressurizing between the inboard and outboard valve was 43.80 SCFH. When the outboard valve was tested individually its leakage was 34.56 SCFH. The difference between the two results is the leakage of the inboard valve (9.24 SCFH at 25 PSIG). This value would also represent the "worst case" through leakage for the 'A' main steam line.

2. 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 1/2 the LLRT result (i.e. assume that each valve leaks equally maximizing the through leakage).

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

4. The feedwater check valves were modified by M-4-2-80-27 to add additional hold down clamps to seat. The valves were re-tested after the modification was complete.

5. The 36B and 37B valves were tested together with 34B (but without draining and venting the volume external to the 34B). The leakage was 163.83 SCFH. Then only the 36B valve was repaired. Then all three valves (34B, 36B, and 37B) were tested. The leakage was 2.40 SCFH. This means that the worst possible leakage through the 34B (without any repairs to it) was 2.40 SCFH. The two possible leakage paths from the containment are 34B-36B and 34B-37B. The worst case through leakage in the "as found" condition is the leakage of the 34B or 2.40 SCFH.

6. The two steam supply valves, 2-1301-16, 17, were tested together by pressurizing between the valves. The result was 20.40 SCFH. Manual assistance (by cranking down on the valve operator) was applied to the 2-1301-17 valve only. The LLRT was then repeated with the result of 4.90 SCFH. It is conservative to assume that the result of the second test was entirely the leakage of the 2-1301-16 valve, which was not assisted in any way. The 2-1301-17 valve was then repaired and no work was performed on the 2-1301-16 valve. The volume was again tested with the result of 3.35 SCFH. Note 2 applies to the through leakage for the "as left" result.

7. The drywell/torus purge LLRT consists of pressurizing between four primary containment valves. Leakage through two of the valves is required to have a containment leakage path. The "as found" LLRT result was 68.10 SCFH. A visual inspection revealed leakage at the seating surface of the AO 2-1601-22 valve. Adjustments were made to this valve without any repairs to the other three valves and the LLRT was repeated with the result of 24.80 SCFH. The latter test verifies that none of the unrepaired valves can leak more than 24.80 SCFH so this value is used as the "as found" through leakage. Since two valves must leak to form a leakage

path the possible leakage following the repair cannot exceed 1/2 the final LLRT result or 12.40 SCFH.

8. The isolation valves on the sump pump discharge lines were modified (M-4-2-83-7, M-4-2-83-19). This modification relocated the valves to allow the operators to be installed with the valve stem vertical, instead of horizontal. This modification should improve valve reliability and LLRT results, based on maintenance experience on these valves. The valves were re-tested following the modification.

9. This valve was re-tested because of the modification to the piping changing the test volume and primary containment boundary.

10. The doors on the personnel interlock required adjustment to the packing glands on the operator shafts. The leakages shown are the values after conversion to P_a (48 PSIG). The test is performed at 10 PSIG as allowed by Technical Specification. The conversion ratio used is from the laminar flow model (Ref. ORNL-NSIC-5, Oak Ridge National Laboratory, Aug., 1965). The through leakages shown are 1/2 the LLRT result.

11. 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 containment. Therefore, the worst case is when both boundaries leak equally. The through leakages are then 1/2 the LLRT result.

12. The bellows on the X-16B penetration was replaced and modification tested during the Type A test as required by 10 CFR 50, Appendix J, Section IV.A.

ID/TS8-W

APPENDIX B

SUMMARY OF ALL CONTAINMENT

LEAK RATE TESTING DURING

UNIT TWO REFUELING OUTAGE

FALL, 1983

SUMMARY OF ALL CONTAINMENT
LEAK RATE TESTING DURING
UNIT TWO REFUEL OUTAGE
FALL, 1983

	AS FOUND (SCFH) LLRT (TOTAL MEASURED)	AS FOUND (SCFH) WORST CASE THROUGH LEAKAGE	AS LEFT (SCFH) LLRT (TOTAL MEASURED)	AS LEFT (SCFH) WORST CASE THROUGH LEAKAGE
(1) MSIV's @ 25 PSIG	59.92	17.30	27.66	10.36
(2) MSIV's converted to 48 PSIG*	94.67	27.33	43.70	16.37
(3) All Type C Tests (Except MSIV's)	1095.79	144.92	215.66	109.40
(4) All Type B Tests	108.31	54.23	22.49	11.32
TOTAL (2 + 3 + 4)	1298.77	226.48	281.85	137.09

(1) Type A Test (Integrated Leak Rate Test) = 0.385 wt %/day

(2) Upper Confidence Limit of Type A Test Result = 0.392 wt %/day

(3) Correction for Unvented Volumes During Type A Test = 0.079 wt %/day

(4) Correction for Repairs Prior to Type A Test (As Found - As Left) = 0.183 wt %/day $\left(\frac{226.48 - 137.09}{489.59}\right)$

(5) Correction for Leak Repaired During Type A Test = 0.020 wt %/day

TOTAL (2 + 3 + 4 + 5) 0.674 wt %/day (As Found ILRT Result)

*Leak Rate at 25 PSIG converts to Leak Rate at 48 PSIG using CONVERSION RATIO OF 1.58. REFERENCE ORNL - NISC - 5, Oak Ridge National Laboratory, Aug. 1965, page 10.55.

APPENDIX C

APPENDIX C

REPAIRED ITEM	WORK REQUEST	DESCRIPTION OF REPAIRS
MO 2-220-1	Q25256	Lapped seat and disc; replaced stem and packing; replaced seal ring (not re-useable).
MO 2-220-2	Q29175	Replaced disc and disc guide; machined seat; replaced stem and packing; replaced seal ring (not re-useable).
CV 2-1301-41	Q29176 Q31191	Lapped seat; replaced hinge pin and disc; replaced all seals and gaskets.
MO 2-1001-36B	Q26942	Replaced plug, plug nut, and plug guide; Seat built up with Stellite and re-machined (nick in seat was indication of foreign material); stem and packing replaced.
MO 2-1301-17	Q28957	Removed, cleaned, lubricated and inspected Limitorque valve operator; replaced bent de-clutch rod.
CV 2-220-62B	Q14498	Seat O-ring replaced; modification M-4-2-80-27 (Add'l hold down clamps on seat) installed.
AO 2-1602-22	Q28843	Adjusted valve disc so as to seal uniformly on the seat. The disc was slightly off center with respect to the seat.
FCV 2-8802C	Q28972	Lapped valve seat and cleaned disc.
CV 2-220-58A	Q14500 Q29006 Q30940	Seat O-ring replaced; modification M-4-2-80-22 (Add'l hold down clamps on seat) Installed
CV 2-2599-24A	Q29128	Cleaned disc guide and seat. Replaced gasket (Not re-useable).
Penetration X-16B	Q30484 Q31163	Weld repair of bellows failed; bellows was replaced.

DMB

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NJK-84-94

March 19, 1984

J. Keppler, Regional Administrator
Office of Inspection and Enforcement
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Reference: Quad-Cities Nuclear Power Station
Docket Number 50-265, DPR-30, Unit Two
Appendix A, Section 4.7.A.2

Enclosed please find Reportable Occurrence Number (RO) 83-15/03L-1 for Quad-Cities Nuclear Power Station.

This report is submitted to you in accordance with the requirements of Technical Specification 6.6.B.2.b, as a condition leading to operation in a degraded mode permitted by a limiting condition for operation.

The original Reportable Occurrence report (RO) 83-15/03L-0, stated that the Local Leak Rate Testing program had found leakage in excess of Technical Specification limits, but did not provide a complete summary. This report addresses all valves and penetrations which had excess leakage and the repairs which were performed to reduce containment leakage to within Technical Specification limits.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

N. J. Kalivianakis

N. J. Kalivianakis
Station Superintendent

NJK:JRW/bb

Enclosure

cc B. Rybak
A. Morrongiello
INPO Records Center

IE22

APR 4 1984