LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
QUAD-CITIES, NUCLEAR POWER STATION, UNIT Two	0 5 0 0 0 2 6 5	1 of 1 7

Title (4) Leak Rate From All Valves and Penetrations on Unit Two in Excess of Tech Spec 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
1 0	1 2	8 6	8 6	0 1 4	0 1	0 6	1 0	8 7		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	
Tom Crippes Technical Staff Engineer, Ext. 2151	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
B	S J	I S V	C 6 8 4	Y	X	B N	I S V	C 6 8 4	Y
X	J M	P E N	C 3 1 0	Y	X	J M	I S V	P 3 4 0	Y

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
__ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

On October 11, 1986, Quad Cities Unit Two was shutdown for refueling. On October 12, 1986, while performing refuel outage local leak rate testing, the measured combined leakage for all penetrations and valves, except Main Steam Isolation Valves, was found to be in excess of the Technical Specification limit of 293.75 SCFH ($0.60L_a$). This report documents the repairs made to valves and penetrations with unacceptable leak rates and the final results of the Local Leak Rate Testing program. This report is submitted in accordance with 10CFR50.73(a)(2)(ii) which requires the 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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NRC Form 366B
(9-83)

LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

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

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

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	JM	ISV	C684	Y					
X	IK	ISV	C635	Y					
X	IK	ISV	C635	Y					

NRC FORM 366B
(9-83)

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].

EVENT IDENTIFICATION:

The leak rate from all valves and penetrations (excluding MSIVs) is in excess of the Technical Specification limit of $0.60L_a$ or 293.75 SCFH as defined in Technical Specification 3.7.A.2.

A. CONDITIONS PRIOR TO EVENT:

Unit: Two	Event Date: October 12, 1986	Event Time: 0300
Reactor Mode: 1	Mode Name: Shutdown	Power Level: 0%

This report was initiated by Deviation Report D-4-2-86-56.

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 October 11, 1986, Unit Two was shutdown for the end of cycle eight refueling and maintenance outage. At 0300 hours on October 12, 1986, while performing refueling outage Local Leak Rate Testing, the measured combined leakage rate for all penetrations and valves, except Main Steam Isolation Valves [SB], was found to be in excess at 293.75 SCFH ($0.60L_a$). Tables 1 and 2 beginning on page 9 of this report provide details about the required repairs and/or adjustments (RAs) made to the equipment that caused the excessive leakage. Note that some of the RAs were not due to excessive leakage, but were the result of preventive maintenance. The valve leakage before and after the RAs and an explanation of the work performed is provided in Table 1. For valves where the RAs were initiated due to Local Leak Rate Test (LLRT) results, rates are shown in the Comment section with details provided in the corrective action section of this report.

C. APPARENT CAUSE OF EVENT:

This report is being submitted to comply with the requirements of 10CFR50.73(a)(2)(ii) which requires the reporting of any event or condition that resulted in the condition of the nuclear power plant, including its principle safety barrier, being seriously degraded.

The cause for each specific problem can be found in Table 2 of this report beginning on page 10. In general, the causes identified were normal wear or design deficiency.

The first step to a good corrective action or maintenance program is to determine why the valve in question leaked. The answer to that question is not always obvious when dealing with valves that are sometimes quite large or when the air leakages are small but require repair due to regulatory limitations. At Quad Cities, we believe that we have a good program for diagnosing valve problems and facilitating repairs through the use of Station Procedure QMP 800-18 and the checklist QMP 800-S15. When any safety related and/or primary containment [NH] isolation valve is disassembled, a Quality Control Inspector performs a thorough inspection of the valve in order to determine the root cause of the valve leakage (or any other problems mandating the repair). An additional inspection is performed during reassembly of the valve. We believe that this method of diagnostics and control on these types of repairs meet or exceed any prevailing standard within the industry.

In addition, Quad Cities maintains on file the LLRT results for every primary containment isolation valve and penetration dating back to plant startup and trends those results. The station's willingness to repair valves or penetrations that exhibit low, but equipment specific high or increasing leakages over past LLRT results, demonstrates a sincere effort to meet the requirements of 10 CFR 50, Appendix J.

Because of the stringent testing requirements of the above regulation and problems encountered industry wide in meeting those requirements, the corrective action portion of this report has been prepared to identify "chronic" problems experienced at Quad Cities. Actions taken in the past and future plans are discussed.

The specific action taken this refuel outage on all valves with RAs due to LLRT leakage is given below in Table 2 beginning on page 11. The note numbers can be referenced back to Table 1 to identify the valves.

D. SAFETY ANALYSIS OF EVENT:

The consequence of this occurrence is that it was necessary to repair a number of containment isolation valves 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 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 containments could 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.

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 [SJ] and Residual Heat Removal System (RHRS) [BO]). 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.

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 [EK], the Secondary Containment, the Standby Gas Treatment System [BH].

E. <u>CORRECTIVE ACTION:</u>

The problems listed below required some type of maintenance due to excessive local leak rates. The immediate action taken for many of these problems was sufficient corrective action because the leakage involved was small and no pattern of chronic failure existed. The items of special concern, however, are the feedwater check valves and the drywell head which have a history of excessive leakage and/or large leakage rates. These two problems along with the rest of the problems listed below will be discussed in detail here concerning corrective actions required to correct the problem.

1. Inboard and outboard feedwater (FW) [SJ] check valves (CV 2-220-58A, 62A)

2. Reactor Core Isolation Cooling (RCIC) [BN] steam supply valve (MO 2-1301-16)

3. Drywell and suppression chamber purge valve [VL] (AO 2-1601-22)

4. Oxygen analyzer isolation valves [IK] (FCV 2-8801C, FCV 2-8802C)

5. Drywell head

6. Suppression chamber to reactor building vacuum breaker [BF] (CV 2-1601-31B)

1. Feedwater Check Valves

As an immediate corrective action, both the Feedwater Check valve (CV 2-220-58A, 62A) internals were replaced like for like with Crane valves model #159-262-101. Both valves were successfully Local Leak Rate Tested (LLRT) and returned to service on November 24, 1986.

The failure of these valves to give good LLRT results is well documented at Quad Cities and at other stations throughout the industry. The primary problem with these valves continues to be focused on the fact that in order to isolate a high pressure water line, a higher air pressure is going to be required in order to seat the disk prior to testing.

The station has initiated an Action Item Request (AIR No. 85-12) to Station Nuclear Engineering Department (SNED) to investigate this problem and to determine a solution. The current status of this AIR is that procedures QMP-800-28 and QMP 800-S22 for preventive maintenance on these feedwater check valves are being developed. These procedures include cleaning the disk and seat assembly. The procedure also includes reducing the dimensional tolerance between the disc and seat ring bushing inside diameter (I.D.) and hinge pin outside diameter (O.D.). This will allow for better seating of the check valve disk.

2. RCIC Steam Supply Valve (MO 2-1301-16)

The immediate corrective action for the RCIC Steam Supply valve (MO 2-1301-16) leakage was to machine the inner bonnet and fitted wedge, and to replace the valve stem, seal ring, and packing. Also, Mechanical Maintenance replaced the packing on test tap valve 2-1301-18A. The MO 2-1301-16 & 17 (downstream steam isolation) valves were successfully LLRT and returned to service on January 15, 1987.

3. Drywell & Suppression Chamber Purge Valve (MO 2-1601-22)

The immediate corrective action was to replace the MO 2-1601-22 valve like for like with a Henry Pratt valve model #D1200G. The test volume was then successfully LLRT and returned to service on November 18, 1986.

4. Oxygen Analyzer Isolation Valves (FCV 2-8801C & FCV 2-8802C)

The immediate corrective action for both the FCV 2-8801C and the FCV 2-8802C valves was that the valves were removed and the internals were cleaned. The valve seats in both valves were lapped. On January 2, 1987, the valves were successfully LLRT and returned to service.

5. Drywell Head

On October 12, 1986, prior to the Integrated Primary Containment Leak Rate Test (IPCLRT), Technical Staff personnel Local Leak Rate Tested the Drywell Head Flange and the leakage was discovered to be greater than 60 SCFH. On October 13, 1986, the IPCLRT was conducted in order to determine the effects of a faulty Drywell Head Flange Seal. As a result, the IPCLRT failed with a leak rate of 464.06 SCFH at a 95 percent confidence level, which was above the acceptable limit of 367.2 SCFH per QTS 150-1. Maintenance personnel replaced the Drywell Head Gasket with a new type of gasket (Garlock #8364). The original gasket was AMS 3301 40 durometer red silicone material. Chicago Bridge & Iron Company recommended that Garlock #8364 gasket material be used. On October 14, 1986, the drywell head flange was successfully leak rate tested with 0.0 SCFH leakage and on October 14 and 15, 1986 the IPCLRT was also successfully completed with a leak rate of 177.11 SCFH at a 95 percent confidence level which is below the 367.2 SCFH limit. After the IPCLRT was completed, the Drywell Head Flange was again tested to determine the effect of an IPCLRT on the drywell head gasket. As a result, the leakage was found to be 0.0 SCFH.

Presently there is a two phase investigation being performed to find a better gasket material for use in the Drywell Heads. Phase One: Station Nuclear Engineering (SNED) is looking into different types of gasket materials (Silicone and EPDM) and how they compare to the materials we have used and are presently using. Also, SNED has sent samples of the two gasket materials we have used to Battel Testing Lab for testing (including radiation testing). The test results will then be compared with the test results from other types of gasket materials. Phase Two: Nuclear Services Technical (NST) is conducting a test at Systems Operational Analysis Department (SOAD). They are testing four different types of gasket material. These materials are being tested in a fixture that was fabricated to resemble the configuration of the drywell head flange. The material is to be heated in an oven for a set period of time and subjected to specific tests. Upon collection of all data, a decision will be made on what gasket material should be used in the drywell head.

6. Suppression Chamber to Reactor Building Vacuum Breaker (CV 2-1601-31B)

The immediate corrective action for this valve was to disassemble the valve and clean the seat. Also, the valve packing and O-rings were replaced. The valve was reassembled and successfully LLRT on January 15, 1987.

F. PREVIOUS EVENTS:

Recent previous leak rate failures at Quad Cities Station are documented in Licensee Event Reports (LER) 254/86-001 and 265/85-007. In addition, LER 265/86-015 documents the IPCLRT failure which occurred on October 13, 1986.

G. COMPONENT FAILURE DATA:

See Tables 1 and 2.

TABLE 1

Description/ Valve #	Component Manufacturer and Model	As Found/ As Left (SCFH)	Comments
Main Steam Line Drain (MO 2-220-2)	Crane Co. (L200)	*23.68/*28.8	Retested after preventive maint - not due to LLRT
A FW Outboard Check Valve (CV 2-220-62A)	Crane Valve (159-262-101)	47.5/12.2	Note 2
A FW Inboard Check Valve (CV 2-220-58A)	Crane Valve (159-262-101)	333.6/0.52	Note 1
RHRS Injection (MO 2-1001-29B)	Crane Co. (783-U)	12.4/14.5	Retested after preventive maint - not due to LLRT
RCIC Steam Supply (MO 2-1301-16)	Crane Co. (783-U)	*23.5/*8.6	Note 3
DW & Suppression Chamber Purge (AO 2-1601-22)	Henry Pratt (2 FII)	*41.28/*4.13	Note 4
Suppression Chamber to Rx Building Vacuum Breakers (CV 2-1601-31B)	Crane Co. (47-1/2-U)	*14.76/*14.25	Note 8
Oxygen Analyzer Isolation (FCV 2-8801C)	Copes Vulcon Inc. (D-100-60)	50.0/5.0	Note 5
Oxygen Analyzer Isolation (FCV 2-8802C)	Copes Vulcon Inc. (D-100-60)	40.0/0.1	Note 6
Drywell Head	Chicago Bridge & Iron	>60(Would not pressurize)/0.0	Note 7
Traversing Incore Probe (TIP) Ball Valve [IG] (733-1)	General Pneumatics Corp. (608 KW J06-3)	0.2/0.25	Retested after preventive maint - not due to LLRT
TIP Ball Valve (733-2)	General Pneumatics Corp. (608 KW J06-3)	0.4/0.85	Retested after preventive maint - not due to LLRT
TIP Ball Valve (733-3)	General Pneumatics Corp. (608 KW J06-3)	0.5/0.0	Retested after preventive maint - not due to LLRT
TIP Ball Valve (733-4)	General Pneumatics Corp. (608 KW J06-3)	0.0/2.2	Retested after preventive maint - not due to LLRT
TIP Ball Valve (733-5)	General Pneumatics Corp. (608 KW J06-3)	0.9/0.0	Retested after preventive maint - not due to LLRT

TABLE 1 - Cont.

Description/ Valve #	Component Manufacturer and Model	As Found/ As Left (SCFH)	Comments
Containment Atmosphere Monitor (CAM) System [IK] (SO 2-2499-1A)	Target Rock (1/2 SMS-S-1)	0.0/0.0	Retested after EQ maintenance
CAM System (SO 2-2499-2A)	Target Rock (1/2 SMS-S-1)	0.0/0.0	Retested after EQ maintenance
CAM System (SO 2-2499-1B)	Target Rock (1/2 SMS-S-1)	0.0/0.0	Retested after EQ maintenance
CAM System (SO 2-2499-2B)	Target Rock (1/2 SMS-S-1)	0.0/0.0	Retested after EQ maintenance
CAM System (SO 2-2499-3A)	Target Rock (1/2 SMS-S-1)	0.0/0.0	Retested after EQ maintenance
CAM System (SO 2-2499-4A)	Target Rock (1/2 SMS-S-1)	0.0/0.0	Retested after EQ maintenance
CAM System (SO 2-2499-3B)	Target Rock (1/2 SMS-S-1)	0.0/0.0	Retested after EQ maintenance
CAM System (SO 2-2499-4B)	Target Rock (1/2 SMS-S-1)	0.0/0.0	Retested after EQ maintenance

* Leak rate included all valves in test volume.

TABLE 2

Note No.	Discussion
1	Work Request Q53226 was initiated to repair the inboard feedwater check valve CV 2-220-58A. The valve was found to have a badly worn valve seat, guide, and disc due to a worn bushing caused from normal wear. Also, Work Request Q53227 was written to repair test tap valve 2-220-115A which had a packing leak that contributed to the leak rate within the test volume.

Leakage History	Feedwater Check Valve (CV 2-220-58A)
07-13-71	4.60 SCFH
12-30-74	97.00 SCFH
02-08-75	8.80 SCFH
10-12-76	1.70 SCFH
10-19-76	0.00 SCFH
02-03-78	1.56 SCFH
12-26-79	Unable to Pressurize
01-16-80	7.80 SCFH
09-21-81	1.03 SCFH
10-05-83	267.30 SCFH
01-09-84	0.52 SCFH
04-10-85	1921.70 SCFH
05-04-85	16.00 SCFH
11-07-86	333.60 SCFH
11-24-86	0.52 SCFH

CONCLUSION: The feedwater check valves are large 18" check valves on the feedwater lines and have an erratic test history. The main reason for this is that the valve does not seat when tested with 48 psig of air. All feedwater check valves are considered a chronic problem. Reference the corrective action section of this report.

TABLE 2 - (Cont.)

Note No.	Discussion
2	Work Request Q53154 was initiated to repair the outboard feedwater check valve CV 2-220-62A. The check valve O-ring and seal ring were found to be defective due to normal wear from use.

Leakage History	Feedwater Check Valve (CV 2-220-62A)
07-14-71	4.00 SCFH
12-30-74	472.00 SCFH
03-17-75	0.00 SCFH
10-12-76	271.00 SCFH
10-22-76	5.16 SCFH
02-06-78	513.80 SCFH
02-14-78	9.50 SCFH
12-26-79	Unable to Pressurize
01-16-80	0.00 SCFH
09-22-81	1140 SCFH
10-07-81	6.20 SCFH
10-06-83	2.70 SCFH
01-06-84	5.26 SCFH
04-10-85	10.85 SCFH
11-05-86	47.50 SCFH
11-24-86	12.20 SCFH

CONCLUSION: The feedwater check valves are large 18" check valves on the feedwater lines and have an erratic test history. The main reason for this is that the valve does not seat when tested with 48 psig of air. All feedwater check valves are considered a chronic problem. Reference the corrective action section of this report.

TABLE 2 (Cont.)

Note No.	Discussion
3	Work Request Q54642 was initiated to inspect and repair RCIC Steam Supply Valve (MO 2-1301-16). The valve was found to have a wedge that was out of adjustment. The wedge fit tightly at the bottom of the seating surface but fit loosely at the top of the seating surface. Also, Work Request Q54468 was written to repair test tap valve 2-1301-18A. This valve had a packing leak which added to the leakage between the MO 2-1301-16 & 17 valves.

Leakage History	RCIC Steam Supply Valve Boundary (MO 2-1301-16 & 17)
07-26-71	2.10 SCFH
02-19-75	5.90 SCFH
09-11-76	19.68 SCFH
01-15-78	17.50 SCFH
11-26-79	82.10 SCFH
02-28-80	6.50 SCFH
09-07-81	29.00 SCFH
12-01-81	46.00 SCFH
09-06-83	20.40 SCFH
01-18-84	3.35 SCFH
03-17-85	4.10 SCFH
04-27-85	2.00 SCFH
01-02-87	23.50 SCFH
01-15-87	8.60 SCFH

CONCLUSION: The leakage history of these valves do not appear to be a major problem. The last significant leakage detected was four cycles ago. The valve inspection and corrective action required do not indicate that these valves have a chronic problem requiring further corrective action at this time. Reference the corrective action section of this report.

TABLE 2 (Cont.)

Note No.	Discussion
4	Work Request Q51932 was initiated to repair or replace the Drywell/Torus Purge Valve (AO 2-1601-22). The valve appeared to have a worn valve seat caused by normal wear during plant operation.

Leakage History	Drywell Suppression Chamber Purge Valve Boundary (AO 2-1601-21, 22, 55, 56)
08-21-71	6.70 SCFH
08-23-72	15.5 SCFH
04-19-73	6.50 SCFH
05-04-73	14.4 SCFH
10-17-73	14.4 SCFH
09-15-76	61.22 SCFH
10-11-76	11.75 SCFH
01-19-78	Unable to pressurize
03-06-78	4.13 SCFH
12-03-79	136.10 SCFH
01-17-80	14.50 SCFH
09-29-81	53.60 SCFH
12-20-81	10.32 SCFH
09-27-83	68.10 SCFH
01-10-84	24.80 SCFH
03-23-85	33.00 SCFH
10-16-86	41.28 SCFH
11-18-86	4.13 SCFH

CONCLUSION: Over the previous years, the Drywell/Suppression Chamber Purge Valve (AO 2-1601-21, 22, 55, 56) boundary has shown a history of high leak rates. The AO 2-1601-21 & 22 butterfly valves appear to have more maintenance done to them than do the AO 2-1601-55 gate valve and the AO 2-1601-56 butterfly valve. However, the last time any of these four valves have been replaced was on March 2, 1978, when the AO 2-1601-21 butterfly valve was replaced with a like for like rebuilt valve. Reference the corrective action section of this report.

TABLE 2 (Cont.)

Note No.	Discussion
5	Work Request Q52872 was initiated to inspect and repair oxygen analyzer valve (AO 2-8801C). The valve was found to have a worn valve seat and a large quantity of internal rust due to normal wear and use.

Leakage History	O_2 Analyzer Isolation (AO 2-8801C)
09-16-76	1.20 SCFH
01-18-78	10.50 SCFH
12-06-79	11.00 SCFH
09-29-81	4.40 SCFH
10-03-83	6.00 SCFH
03-29-85	36.50 SCFH
05-22-85	13.00 SCFH
10-21-86	50.00 SCFH
01-02-87	5.00 SCFH

CONCLUSION: This oxygen analyzer valve does not represent a serious source of containment leakage because the line is provided with a second isolation valve (AO 2-8802C). With both isolation valves closed, the leak rate was reduced to 07.0 SCFH which falls within safe limits with respect to La.

TABLE 2 (Cont.)

Note No.	Discussion
6	Work Request Q52871 was initiated to inspect and repair oxygen analyzer valve (AO 2-8802C). The valve was found to have a worn valve seat and a large quantity of internal rust due to normal wear and use.

Leakage History	O_2 Analyzer Isolation (AO 2-8802C)
09-16-76	13.00 SCFH
01-18-78	>30.00 SCFH
02-09-78	0.05 SCFH
12-06-79	0.60 SCFH
09-29-81	0.40 SCFH
10-03-83	16.00 SCFH
11-29-83	1.40 SCFH
03-29-85	9.70 SCFH
05-03-85	6.50 SCFH
10-21-86	40.00 SCFH
01-02-87	0.10 SCFH

CONCLUSION: This oxygen analyzer valve does not represent a serious source of containment leakage because the line is provided with a second isolation valve (AO 2-8801C). With both isolation valves closed, the leak rate was reduced to 37.0 SCFH which falls within safe limits with respect to La.

TABLE 2 (Cont.)

Note No.	Discussion
7	Work Request Q52677 was initiated to investigate and repair the drywell head. The apparent cause for this problem has been attributed to a deterioration in the sealing ability of the drywell head flange seal material. This is documented in Licensee Event Report (LER) 265/86-015.

Leakage History	Drywell Head Flange
08-27-71	0.00 SCFH
10-13-72	0.65 SCFH
11-05-75	1.40 SCFH
09-10-76	0.00 SCFH
10-19-76	1.60 SCFH
03-08-78	0.20 SCFH
11-25-79	0.65 SCFH
12-23-81	0.00 SCFH
09-05-83	7.50 SCFH
02-07-84	0.00 SCFH
03-17-85	0.00 SCFH
05-24-85	0.00 SCFH
10-12-86	>60 SCFH (Prior to IPCLRT)
10-14-86	0.00 SCFH (After repairs)
10-15-86	0.00 SCFH (Immediately after IPCLRT)
01-16-87	0.00 SCFH

CONCLUSION: There have been previous LLRT failures of the Unit One Drywell Head. These failures have subsequently forced Quad Cities to fail the overall Integrated Leak Rate Test (ILRT). Thus, an investigation into the use of an alternative gasket material was initiated. Reference the corrective action section of this report.

TABLE 2 (Cont.)

Note No.	Discussion
8	Work Request Q52662 was initiated to investigate and repair the Suppression Chamber to Reactor Building Vacuum Breaker (CV 2-1601-31B). Maintenance personnel discovered that the internal valve body surfaces, valve seat, and gasket surfaces were dirty. Also, the packing needed to be replaced around the hinge pin.

Leakage History	Suppression Chamber to Rx Bldg. Vacuum Breakers Valve Boundary (AO 2-1601-20B, CV 2-1601-31B)
08-19-71	0.08 SCFH
04-16-73	1.91 SCFH
10-17-73	0.00 SCFH
01-28-75	0.00 SCFH
09-12-76	2.29 SCFH
01-18-78	110.74 SCFH
02-15-78	0.73 SCFH
12-03-79	0.76 SCFH
09-23-81	19.90 SCFH
09-30-81	10.70 SCFH
09-12-83	7.10 SCFH
03-23-85	13.99 SCFH
05-30-85	13.99 SCFH
10-10-86	14.76 SCFH
01-19-87	14.25 SCFH

CONCLUSION: As can be seen in the above data, the leak rate between AO 2-1601-20B and CV 2-1601-31B valve boundary appears to have increased over the past six years. Since the leak rate was left within acceptable limits, no further maintenance was required. However, this particular volume will be given special attention during the next Unit 2 refueling outage which could require that AO 2-1601-20B be replaced.

SHEET 7

REV. 1 4/86
PGK

EVENT SUMMARY AND CAUSE CODES

DVR Number
04-2-86-056 Supp. 1

- [ ] Lost generation
- [ ] Cost > $25,000
- [ ] Hazard or Spill
- [ ] Personnel injury
- [ ] Reactor trip
- [ ] ESF actuation
- [x] NRC reportable
- [x] LER
- [ ] PSE
- [ ] NRC violation, level___
- [ ] GSEP event, class_____
- [x] Tech Spec LCO
- [ ] Potential or future loss
- SALP functional area___

Component type — Failure mode — Department

	Component type		Failure mode		Department		
X	C	M	M	2	MM		Wear, erosion
X	C	M	M	9	MM		out of adjustment
X	C	M	M	8	MM		Dirty

Licensed? L or blank — Level — Department — Type — Detail code

	Licensed?	Level	Department	Type	Detail code	
A						
A						
A						

Type — Detail Code — Department

	Type	Detail Code	Department	
B	D	5	MM	
B				
B				

	Type	Detail code	
C			

Type of deficiency — Detail code — Procedure type

	Type of deficiency	Detail code	Procedure type
D			
D			
D			

Type — Detail code — Department

	Type	Detail code	Department	
E				
E				
E				

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

RLB-87-83

June 10, 1987

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 please find Licensee Event Report (LER) 86-014, Revision 01, 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 the reporting of any event or condition that resulted in the condition of the nuclear power plant, including its principle safety barrier, being seriously degraded.

The original LER 86-014 stated that the local leak rate testing program had found leakage in excess of Technical Specification limits, but did not provide a complete summary pending completion of the testing program and corrective actions. This report addresses all valves and penetrations that had repairs performed to reduce the leakage total to within the Technical Specification limit.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

R L Bax
R. L. Bax
Station Manager

RLB/MSK/rk

Enclosure

cc: I. Johnson
R. Higgins
INPO Records Center
NRC Region III

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