Commonwealth Edison
Quad-Cities Nuclear Power Station
Post Office Box 216
Cordova, Illinois 61242
Telephone 309/654-2241

NJK-76-33

RECEIVED
2-10-76
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

February 2, 1976

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

Reference: Quad-Cities Nuclear Power Station
Docket No. 50-254, DPR-29, Unit One
Appendix A, Section 6.6.B.2.b

Enclosed please find Reportable Occurrence Report No. RO 50-254/76-2 for Quad-Cities Nuclear Power Station. This report is submitted to you in accordance with the requirements of Technical Specification 6.6.B.2.

This report identifies eight occurrences whereby excessive leakages were measured with respect to primary containment isolation valves during local leak rate testing. To date, all primary containment isolation valves and testable penetrations have been initially tested. The following represents a tabular summary of the leak rate test failures covered in this report:

SYSTEM	VALVES TESTED	MEASURED LEAKAGE	CORRECTED LEAKAGE
Main Steam Drains	MO-1-220-1&2	47.7 SCFH	—
Main Steam Isolation Valves	AO-1-203-1B	21.0 SCFH	—
	AO-1-203-1C	70.9 SCFH	—
Drywell & Torus Purge Valves	AO-1-1601-21, 22, 55, 56	197.1 SCFH	10.32 SCFH
Oxygen Analyzer Valves	AO-1-8802B	25.9 SCFH	—
HPCI Steam Exhaust Check Valve	CV-1-2301-45	84.4 SCFH	—
Feedwater Check Valves	CV-1-220-58B	3026 SCFH	—
	CV-1-220-62A	3040 SCFH	—
	CV-1-220-62B	378 SCFH	—

FEB 5 1976

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SYSTEM	VALVES TESTED	MEASURED LEAKAGE	CORRECTED LEAKAGE
RHRS Containment Spray Valves	MO-1-1001-34A, 36A,37A	249 SCFH	6.05 SCFH
Cleanup Pump Suction Valves	MO-1-1201-2&5	Could not pressurize	2.28 SCFH

Those valves which have not been repaired to date will be repaired prior to unit startup at the completion of the current refueling outage. A supplemental report will be sent upon completion of all corrective actions regarding the above valves. Details will be provided as to the cause, repairs, and corrected leak rates measured after repair.

Very truly yours,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

N. J. Kalivianakis
Station Superintendent

NJK/LFG/lk

cc: G. A. Abrell

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

	LICENSEE NAME	LICENSE NUMBER	LICENSE TYPE	EVENT TYPE
01	I L Q A D 1 (7 8 9–14)	0 0 - 0 0 0 0 0 - 0 0 (15–25)	4 1 1 1 1 (26–30)	0 1 (31 32)

	CATEGORY	REPORT TYPE	REPORT SOURCE	DOCKET NUMBER	EVENT DATE	REPORT DATE
01 CON'T	MI (57 58)	L (59)	L (60)	0 5 0 - 0 2 5 4 (61–68)	0 1 0 3 7 6 (69–74)	0 2 0 2 7 6 (75–80)

EVENT DESCRIPTION

02 WHILE PERFORMING LOCAL LEAK RATE TESTS ON THE MAIN STEAM

03 LINE DRAIN VALVES, MO-1-220-1 & MO-1-220-2, IT WAS DETERMINED

04 THAT THE COMBINED LEAKAGE OF THE TWO VALVES WAS 47.7 SCFH,

05 WHICH IS IN EXCESS OF THE 18.36 SCFH LIMIT ALLOWED BY

06 TECHNICAL SPECIFICATION 4.7.A.2.i(2)(b). OF

	SYSTEM CODE	CAUSE CODE	COMPONENT CODE	PRIME COMPONENT SUPPLIER	COMPONENT MANUFACTURER	VIOLATION
07	H B (9 10)	E (11)	V A L V E X (12–17)	N (43)	C 6 6 5 (44–47)	N (48)

CAUSE DESCRIPTION

08 THE APPARENT CAUSE OF THE EXCESSIVE LEAKAGE

09 THROUGH VALVE MO-1-220-2 WAS BETWEEN THE VALVE

10 BODY SEAT AND THE DISK. 3" GATE VALVE MOD. # 783-11

	FACILITY STATUS	% POWER	OTHER STATUS	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION
11	H (9)	0 0 0 (10–12)	NA (13–44)	B (45)	SURVEILLANCE TEST (46–80)

	FORM OF ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY	LOCATION OF RELEASE
12	Z (9)	Z (10)	NA (11–44)	NA (45–80)

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION
13	0 0 0 (9–11)	Z (12)	NA (13–80)

PERSONNEL INJURIES

	NUMBER	DESCRIPTION
14	0 0 0 (9–11)	NA (12–80)

OFFSITE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

	TYPE	DESCRIPTION
16	Z (9)	NA (10–80)

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 Event Description (Cont'd) - THE TWO VALVES LEAK TESTED,

19 VALVE MO-1-220-2 WAS THE ONLY ONE LEAKING. (20-50-265/76-2)

NAME: LARRY L. HENSON PHONE: 309-654-2241 (EXT. 247)

1A

REPORT NUMBER: RO 50-254/76-2

REPORT DATE: February 2, 1976

OCCURRENCE DATE: January 3, 1976

FACILITY: Quad-Cities Nuclear Power Station
Cordova, Illinois 61242

IDENTIFICATION OF OCCURRENCE:

Excessive leakage measured from primary containment isolation valves MO-1-220-1 and MO-1-220-2 during local leak rate testing.

CONDITIONS PRIOR TO OCCURRENCE:

Unit One was in the REFUEL mode for a scheduled refueling outage.

DESCRIPTION OF OCCURRENCE:

On January 3, 1976 a local leak rate test was performed on the Main Steam Line Drain Valves MO-1-220-1 and MO-1-220-2 by pressurizing the pipe volume between these valves to a pressure of 48 psig. It was determined that the combined leakage of these two valves was 47.7 SCFH, which exceeded the 18.36 SCFH limit (5% L_{to}) as given by Technical Specification 4.7.A.2.i(2)(b) for any one isolation valve. Further testing and investigations revealed that valve MO-1-220-2 was the source of the excessive leakage. Work Request No. 270-76 was issued to repair this valve.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

Equipment Failure

The apparent cause of the excessive leakage measurement was a leak path between the valve body seat and the valve disk for MO-1-220-2. The actual cause of this leak path is undetermined at this time.

ANALYSIS OF OCCURRENCE:

Main Steam Line Drain Valves MO-1-220-1 and MO-1-220-2 are normally closed during reactor power operation, and are open during startups to drain the main steam lines. The safety implications of this occurrence are minimized by the fact that MO-1-220-1 was found not to be leaking, and primary containment integrity was not compromised. The leakage path through this line could have been further isolated by closing valves MO-1-220-4 and 1-220-92. This would have prevented any leakage from reaching the condenser.

Further, this occurrence did not hinder the steam drain valves from closing in the event of a Group I containment isolation signal, and therefore, primary containment was not jeopardized. Also, no abnormal radiation exposures would have been experienced by plant personnel and the health and safety of the public would not have been affected by this occurrence.

CORRECTIVE ACTION:

The Maintenance Department will initiate repairs of MO-1-220-2 as soon as practicable during the current refueling outage. Upon completion of this work, a local leak rate test will be performed to verify that the leakage from these valves is within Technical Specification limitations. Details regarding the cause and corrective actions taken to resolve this occurrence will be sent in a supplemental report at the end of the current refueling outage.

FAILURE DATA:

Neither MO-1-220-1 nor MO-1-220-2 have had excessive leakage rates during prior leak testing investigations. MO-2-220-1 and MO-2-220-2 have likewise never leaked excessively.

Valve MO-1-220-2 is a 3" carbon steel gate valve manufactured by Crane Company, Model 783-U, serial number 91809A.

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | |

(PLEASE PRINT ALL REQUIRED INFORMATION)

01 LICENSEE NAME: I L Q A D 1 | LICENSE NUMBER: 00-00000-00 | LICENSE TYPE: 4 1 1 1 1 | EVENT TYPE: 0 1

01 CON'T CATEGORY: M I | REPORT TYPE: L | REPORT SOURCE: L | DOCKET NUMBER: 050-0254 | EVENT DATE: 010476 | REPORT DATE: 020276

EVENT DESCRIPTION

02 Main Steam Isolation Valve (MSIV) local

03 leak rate testing revealed excessive leakage

04 through MSIV 1-203-1B. Once the valve

05 was repaired it was successfully tested.

06 (20 50-254/76-2)

07 SYSTEM CODE: C D | CAUSE CODE: E | COMPONENT CODE: V A L V E X | PRIME COMPONENT SUPPLIER: N | COMPONENT MANUFACTURER: C 6 6 5 | VIOLATION: N

CAUSE DESCRIPTION

08 The apparent cause was leakage around

09 the valve stem. The corrective action

10 was to tighten the valve packing and retime the valve.

11 FACILITY STATUS: M | % POWER: 000 | OTHER STATUS: NA | METHOD OF DISCOVERY: B | DISCOVERY DESCRIPTION: SURVEILLANCE TEST

12 FORM OF ACTIVITY RELEASED: Z | CONTENT OF RELEASE: Z | AMOUNT OF ACTIVITY: NA | LOCATION OF RELEASE: NA

PERSONNEL EXPOSURES

13 NUMBER: 000 | TYPE: Z | DESCRIPTION: NA

PERSONNEL INJURIES

14 NUMBER: 000 | DESCRIPTION: NA

OFFSITE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

16 TYPE: Z | DESCRIPTION: NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 Cause Description (cont'd) - 20" Crane "Y" Globe

19 valve, Ser. No. 856834-39

NAME: [signature] PHONE: 309-654-2241 (243)

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

	LICENSEE NAME (9–14)	LICENSE NUMBER (15–25)	LICENSE TYPE (26–30)	EVENT TYPE (31–32)
01	I L Q A D 1	0 0 - 0 0 0 0 0 - 0 0	4 1 1 1 1	0 1

	CATEGORY (57–58)	REPORT TYPE (59)	REPORT SOURCE (60)	DOCKET NUMBER (61–68)	EVENT DATE (69–74)	REPORT DATE (75–80)
01 CON'T	M I	L	L	0 5 0 - 0 2 5 4	0 1 0 4 7 5	0 2 0 2 7 6

EVENT DESCRIPTION

02 Main Steam Valve (MSIV) local leak rate testing revealed
03 excessive leakage through MSIV 1-203-1C. Repairs
04 will be performed as soon as practical during
05 the current outage.
06

	SYSTEM CODE (9–10)	CAUSE CODE (11)	COMPONENT CODE (12–17)	PRIME COMPONENT SUPPLIER (43)	COMPONENT MANUFACTURER (44–47)	VIOLATION (48)
07	C D	E	V A L V E X	N	C 6 6 5	N

CAUSE DESCRIPTION

08 The actual cause is undetermined at this time. At the
09 end of the current refueling outage a supplemental report will
10 detail the cause and corrective actions for the MSIV

	FACILITY STATUS (9)	% POWER (10–12)	OTHER STATUS (13–44)	METHOD OF DISCOVERY (45)	DISCOVERY DESCRIPTION (46–80)
11	H	0 0 0	NA	B	SURVEILLANCE TEST

	FORM OF ACTIVITY RELEASED (9)	CONTENT OF RELEASE (10)	AMOUNT OF ACTIVITY (11–44)	LOCATION OF RELEASE (45–80)
12	Z	Z	NA	NA

PERSONNEL EXPOSURES

	NUMBER (9–11)	TYPE (12)	DESCRIPTION (13–80)
13	0 0 0	Z	NA

PERSONNEL INJURIES

	NUMBER (9–11)	DESCRIPTION (12–80)
14	0 0 0	NA

OFFSITE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

	TYPE (9)	DESCRIPTION (10–80)
16		NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 NA

19 NA

NAME: Orin M Howerson PHONE: 309-654-2241 (146)

GPO 881.667

REPORT NUMBER: RO 50-254/76-2

REPORT DATE: February 2, 1976

OCCURRENCE DATE: January 4, 1976

FACILITY: Quad-Cities Nuclear Power Station
Cordova, Illinois 61242

IDENTIFICATION OF OCCURRENCE:

Main Steam Isolation Valve (MSIV) local leak rate testing revealed excessive leakage through MSIVs AO-1-203-1B and AO-1-203-1C.

CONDITIONS PRIOR TO OCCURRENCE:

Unit One was in the REFUEL mode for reactor refueling.

DESCRIPTION OF OCCURRENCE:

On January 4, 1976 a local leak rate test was performed on Unit One MSIV AO-1-203-1B. A leakage of 21.0 SCFH was measured. On the same day, a leak rate of 70.88 SCFH was determined through AO-1-203-1C. These leakages were in excess of the 11.5 SCFH allowable leakage for any one MSIV as given by Technical Specification 4.7.A.2.i(2)(c). The leakage measurements were obtained by pressurizing the volume enclosed by the "B" and "C" Main Steam Line inboard and outboard MSIV's through pressure test connections to a test pressure of 25 psig. With no water head on the inboard side, the test yielded a combined two-valve (inboard and outboard) leakage of 31.1 SCFH for the "B" steam line and 72.6 SCFH for the "C" steam line.

The above test was again performed with a water head in excess of the 25 psig on the inboard side of the AO-1-203-1B and 1C MSIVs. The second test verified that outboard AO-1-203-2B MSIV had an acceptable leakage of 10.1 SCFH while the inboard AO-1-203-1B MSIV had a leakage of 21.0 SCFH. Work Request 27-76 was issued to repair valve AO-1-203-1B. The second test revealed that outboard AO-1-203-2C MSIV also had an acceptable leakage of 1.72 SCFH. Therefore, valve AO-1-203-1C had a leak rate of 70.9 SCFH. Work request no. 28-76 was issued to repair this valve.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

Equipment Failure

The apparent cause of the leakage through AO-1-203-1B was determined to be loose packing, thereby affording a leak path up around the valve stem. A soap bubble solution was used to check for packing leaks with valve AO-1-203-1B under pressure, and bubbles were observed in this case. The cause of the leakage through AO-1-203-1C is postulated to be a leak path between the seating surface of the valve body and the pilot stem. This has been the cause of several past occurrences whereby MSIV's have leaked excessively. The 1C MSIV will be disassembled and repaired, and details relative to the cause of the leakage will be sent in a supplementary letter.

ANALYSIS OF OCCURRENCE:

In the event of a main steam line break outside the primary containment, the total leakage possible through the "B" steam line would have been 10.0 SCFH, and total leakage through the "C" steam line would have been 1.72 SCFH.

The excessive leakage rates did not in any way render the MSIVs inoperable, nor was the ability of the valves to perform their design function affected. Upon receipt of a Group I isolation signal, these valves would have shut in the required time, and performed the isolation function.

The Main Steam Isolation Valves utilize straight line flow to provide a good flow pattern and upstream pressure to aid in valve closure by tilting the actuator toward the upstream side of the valve. The balancing feature of the valve makes it possible to take advantage of the upstream pressure to aid in holding the valve closed and to have the advantage of requiring a small actuator cylinder to open the valve. This is accomplished by allowing the full upstream line pressure to bleed into the chamber above the plug through the balancing part to exert a force on the plug internals in a direction to hold it against the seat.

This design, combined with the fact that the MSIVs are open during normal operation, minimizes the safety implications of this occurrence. Therefore, the health and safety of the public were not affected.

CORRECTIVE ACTION:

Work Request 27-76 was issued for AO-1-203-1B and the valve packing was tightened. The valve was subsequently retested and a leakage of 0.27 SCFH was determined. Valve AO-1-203-1C will be repaired as soon as practicable and a subsequent local leak rate test will be performed to verify acceptable leakage. Details concerning the repairs and leak rate test will be provided in a supplementary report.

FAILURE DATA:

Excessive leakage measurements have been determined for the MSIVs on both units in the past.

Unit Two MSIV's 2-203-1B and 2-202-2B were leak rate tested on April 27, 1973 during an investigation of missing parts on electromatic relief valve 2-203-3E. The measured leakage was 52.4 SCFH and repairs were made on the valve pilot stems and seats. A subsequent leak test was performed and the measured leakage was 3.25 SCFH.

The Unit One MSIV's were leak rate tested on April 1, 1974 with valves AO-1-203-2C, 1D, and 2D indicating leak rates of 216.0 SCFH, 34.7 SCFH, and 34.7 SCFH, respectively. Surface defects were noted on the valve pilot stems. After repairs, a re-test resulted in leakages of 8.64 SCFH, 5.76 SCFH, and 0.0 SCFH respectively for valves AO-1-203-2C, 1D, and 2D.

The Unit Two MSIV's were leak tested in December of 1974. The leakages for the combined inboard and outboard valves for steam lines "A" and "B" were in excess of the Technical Specification limitations. Valve AO-2-203-2C also leaked excessively. Slight warpage was the contributing cause of the leakages. All leaking valves were repaired and acceptable leak rate values were measured during re-testing.

The MSIV's are 20" Y-type globe valves manufactured by Crane Company.

LICENSEE EVENT REPORT

CONTROL BLOCK: [| | | | |] (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

01 LICENSEE NAME (7–14)	ILQAD1
LICENSE NUMBER (15–25)	00-00000-00
LICENSE TYPE (26–30)	41111
EVENT TYPE (31–32)	01
01 CON'T CATEGORY (57–58)	MI
REPORT TYPE (59)	L
REPORT SOURCE (60)	L
DOCKET NUMBER (61–68)	050-0254
EVENT DATE (69–74)	010476
REPORT DATE (75–80)	020276

EVENT DESCRIPTION

02 WHILE PERFORMING LOCAL LEAK RATE TESTS ON THE
03 UNIT 1 DRYWELL & SUPPRESSION CHAMBER PURGE VALVES
04 DURING REFUELING, A LEAK RATE OF 197.1 SCFH WAS
05 MEASURED. THIS WAS GREATER THAN THE ALLOWABLE 5% L_{To}
06 FOR ANY ONE ISOLATION VALVE. LEAKAGE WAS FROM FLANGES AT

07 SYSTEM CODE (9–10)	SA
CAUSE CODE (11)	E
COMPONENT CODE (12–17)	PIPEXX
PRIME COMPONENT SUPPLIER (43)	A
COMPONENT MANUFACTURER (44–47)	G257
VIOLATION (48)	N

CAUSE DESCRIPTION

08 THE LEAKAGE WAS PREDOMINANTLY OBSERVED TO BE
09 COMING FROM 2 FLANGES, AS DETERMINED BY USING A SOAP
10 SOLUTION. FLANGE BOLTS WERE TIGHTENED. RE-TEST LEAKAGE WAS 10.32 SCFH

11 FACILITY STATUS (9)	H
% POWER (10–12)	000
OTHER STATUS (13–44)	NA
METHOD OF DISCOVERY (45)	B
DISCOVERY DESCRIPTION (46–80)	SURVEILLANCE TEST
12 FORM OF ACTIVITY RELEASED (9)	Z
CONTENT OF RELEASE (10)	Z
AMOUNT OF ACTIVITY (11–44)	NA
LOCATION OF RELEASE (45–80)	NA

PERSONNEL EXPOSURES

13 NUMBER (9–11)	000
TYPE (12)	Z
DESCRIPTION (13–80)	NA

PERSONNEL INJURIES

14 NUMBER (9–11)	000
DESCRIPTION (12–80)	NA

OFFSITE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

16 TYPE (9)	Z
DESCRIPTION (10–80)	NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 EVENT DESCRIPTION (CONT'D) AO-1-1601-22 & AO-1-1601-56. WORK
19 REQUEST WAS WRITTEN TO TIGHTEN FLANGE BOLTS (RO 50-254/76-2)

NAME: LAWRENCE F. GERNER PHONE: 309-654-2241 (x158)

GPO 881-667

REPORT NUMBER: RO 50-254/76-2

REPORT DATE: February 2, 1976

OCCURRENCE DATE: January 4, 1976

FACILITY: Quad-Cities Nuclear Power Station
Cordova, Illinois 61242

IDENTIFICATION OF OCCURRENCE:

Excessive leakage measured from primary containment isolation valves AO-1-1601-21, 22, 55 and 56 during local leak rate testing.

CONDITIONS PRIOR TO OCCURRENCE:

Unit One was in the REFUEL mode for a refueling outage.

DESCRIPTION OF OCCURRENCE:

At 10:00 a.m. on January 4, 1976, a local leak rate test was performed on the Unit One drywell and suppression chamber purge valves AO-1-1601-21, AO-1-1601-22, AO-1-1601-55, and AO-1-1601-56. The test consisted of pressurizing the piping volume enclosed by these valves to 48 psig and then measuring the leakage. The leakage was determined to be 197.1 SCFH which exceeded the limit of 5% L_{to} (18.36 SCFH) for any one primary containment isolation valve, as set forth in Section 4.7.A.2.i of the Technical Specifications.

During the performance of the test, with the pipe volume under pressure, all valve packings, welds, and flanges were checked for leakage using a soap bubble solution. Extensive bubble formation was noticed at the flanged junctions of the pressurized piping on valves AO-1-1601-22 and AO-1-1601-56. This leakage observed from the flanges was thus determined to be the predominant leakage path from the pipe volume. No other leakage was observed while using the soap bubble solution. The seating surface of valve AO-1-1601-22 was also checked with soap bubble solution and no leakage was noticed. Work Request No. 36-76 was written to tighten the flanges as a means to stop the leakage.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

Equipment Failure

The cause of the excessive leakage was determined to be the fact that the two flange connections were not fully tightened, and thus afforded a leakage path when pressurized. Possible pipe vibration may have caused the connections to loosen up.

ANALYSIS OF OCCURRENCE:

The leakage detected was at valves AO-1-1601-22 and AO-1-1601-56. However, inboard primary containment isolation valve AO-1-1601-21 was observed to be completely intact at the flanges. Therefore, leakage from the drywell to the reactor building would have been minimal during accident conditions.

Each primary containment isolation valve involved with this particular test was completely capable of performing its intended function in its required manner, and would have operated properly after having received a Group II isolation signal.

This occurrence did not in any way reduce the ability of the primary containment isolation system to function as designed, and since the inboard isolation valve was fully intact, this occurrence did not reduce the reliability of the primary containment. Therefore, reactor safety and the health and safety of the public were not jeopardized by this occurrence.

CORRECTIVE ACTION:

Mechanical Maintenance personnel tightened the leaking flanges by taking up on all the bolts holding the flanges together. A subsequent local leak rate test was run on the Drywell and Suppression Chamber Purge Valves on January 9, 1976. The measured leakage was 10.32 SCFH, which was acceptable.

FAILURE DATA:

The subject primary containment isolation valves have been tested for leakage during prior outages, and have been found to leak excessively on two occasions.

On April 16, 1973, a leak rate test on the Unit 2 purge valves revealed that AO-2-1601-22 leaked excessively. The valve butterfly disc was found to be slightly mis-aligned.

On May 21, 1974, the Unit 1 purge valves were tested and the measured leakage was 26.1 SCFH. In this case, the rubber seat for AO-1-1601-21 was dirty, and there were fitting leaks from instrument lines adjoining the purge piping.

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

01	LICENSEE NAME (9–14)	LICENSE NUMBER (15–25)	LICENSE TYPE (26–30)	EVENT TYPE (31–32)
7 8	I L Q A D 1	0 0 - 0 0 0 0 0 - 0 0	4 1 1 1 1	0 1

01 CON'T	CATEGORY (57–58)	REPORT TYPE (59)	REPORT SOURCE (60)	DOCKET NUMBER (61–68)	EVENT DATE (69–74)	REPORT DATE (75–80)
7 8	M I	L	L	0 5 0 - 0 2 5 4	0 1 0 4 7 6	0 2 0 2 7 6

EVENT DESCRIPTION

02 LOCAL LEAK RATE TESTING DURING THE UNIT ONE REFUELING OUTAGE
03 SHOWED THAT VALVE 1-8802-B LEAKED AT A RATE IN EXCESS OF
04 THAT ALLOWED BY THE TECHNICAL SPECIFICATIONS. A work
05 request was issued to inspect & repair the valve
06 (RO 50-254/76-?)

07	SYSTEM CODE (9–10)	CAUSE CODE (11)	COMPONENT CODE (12–17)	PRIME COMPONENT SUPPLIER (43)	COMPONENT MANUFACTURER (44–47)	VIOLATION (48)
7 8	X X	E	V A L V E X	A	C 6 3 5	N

CAUSE DESCRIPTION

08 The specific cause of the equipment Failure will be
09 determined when the valve is inspected for repair.
10 Additional information will be supplied as it becomes available

11	FACILITY STATUS (9)	% POWER (10–12)	OTHER STATUS (13–44)	METHOD OF DISCOVERY (45)	DISCOVERY DESCRIPTION (46–80)
7 8	H	0 0 0	NA	3	SURVEILLANCE TEST

12	FORM OF ACTIVITY RELEASED (9)	CONTENT OF RELEASE (10)	AMOUNT OF ACTIVITY (11–44)	LOCATION OF RELEASE (45–80)
7 8	Z	Z	NA	NA

PERSONNEL EXPOSURES

13	NUMBER (9–11)	TYPE (12)	DESCRIPTION (13–80)
7 8	0 0 0	Z	NA

PERSONNEL INJURIES

14	NUMBER (9–11)	DESCRIPTION (12–80)
7 8	0 0 0	NA

OFFSITE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

16	TYPE (9)	DESCRIPTION (10–80)
7 8	Z	NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 Cause Description (cont'd) - 1/2" globe valve provided by
19 Copes Vulcan Ser# 6710-58?63-69-6

NAME: Donald R Anderson PHONE: 309-654-2241 (Ex247)

GPO 881-667

REPORT NUMBER: RO 50-254/76-2

REPORT DATE: February 2, 1976

OCCURRENCE DATE: January 4, 1976

FACILITY: Quad-Cities Nuclear Power Station
Cordova, Illinois 61242

IDENTIFICATION OF OCCURRENCE:

Excessive leakage measured from primary containment isolation valve AO-1-8802B during local leak rate testing.

CONDITIONS PRIOR TO OCCURRENCE:

Unit One was in the REFUEL mode for a refueling outage.

DESCRIPTION OF OCCURRENCE:

On January 4, 1976, leak rate testing was performed on the primary containment oxygen analyzer valves. Valve AO-1-8802B, the outboard isolation valve on sample line 1-8802-1/2"-H, was tested individually by uncoupling the sample line at the analyzer cabinet, and pressurizing the line and valve to a test pressure of 48 psig. The measured leakage was 25.9 SCFH, thereby exceeding the Technical Specification limit of 18.36 SCFH for any one primary containment isolation valve. All other oxygen analyzer valves tested satisfactory.

A soap bubble solution was used to check for fitting and packing leaks. Some packing leakage was detected, and operating personnel tightened down on the packing. A re-test was performed on January 7, 1976 and the leakage was measured to be 19.3 SCFH. Since the Technical Specification limit had still been exceeded, Work Request No. 87-76 was issued to disassemble, inspect, and repair valve AO-1-8802B.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

Equipment Failure

The apparent cause of the excessive leakage from isolation valve AO-1-8802B is postulated to be dirty seating surfaces. To date, this valve has not been taken apart. The valve will be repaired prior to the commencement of unit startup, and a subsequent local leak rate test will be performed. The results of these repairs and re-test will be given in a supplemental report.

ANALYSIS OF OCCURRENCE:

The leakage through valve AO-1-8802B was greater than the other seven oxygen analyzer valves tested. The following are the test results for these valves:

AO-1-8801A	-	0.18 SCFH
AO-1-8802A	-	1.99 SCFH
AO-1-8801B	-	0.17 SCFH
AO-1-8801C	-	0.47 SCFH
AO-1-8802C	-	1.92 SCFH
AO-1-8801D	-	1.77 SCFH
AO-1-8802D	-	2.06 SCFH

Valve AO-1-8802B is in series with valve AO-1-8801B. Since the latter valve had a leakage of 0.17 SCFH, the total leakage through line 1-8802-1/2" -H was minimal.

Furthermore, this isolation valve was fully capable of performing its function of closing after having been given a Group II primary containment isolation signal. Thus, this occurrence did not jeopardize the safety of the reactor nor the health and safety of the public.

CORRECTIVE ACTION:

Corrective action will consist of disassembly of the valve and cleaning of the valve seating surfaces. Following a valve operability exercise, a local leak rate test will be performed to verify compliance with the Technical Specifications.

FAILURE DATA:

The AO-1-8802B valve was tested for leakage on April 10, 1974, and could not be pressurized to 48 psig due to its inability to seat fully. The valve was disassembled, cleaned, and reassembled. The packing was adjusted to facilitate smooth operation. A leak test was performed on April 16, 1974 and the measured leakage was 6.58 SCFH.

Valve AO-1-8802B is a 1/2" stainless steel air operated globe valve manufactured by Blaw-Knox Corp., Copes-Vulcan Division, serial number 6710-58363-59-1.

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

	LICENSEE NAME (9–14)	LICENSE NUMBER (15–25)	LICENSE TYPE (26–30)	EVENT TYPE (31–32)
01 (7–8)	ILQAD1	00-00000-00	41111	01

	CATEGORY (57–58)	REPORT TYPE (59)	REPORT SOURCE (60)	DOCKET NUMBER (61–68)	EVENT DATE (69–74)	REPORT DATE (75–80)
01 CON'T (7–8)	MI	L	L	050-0254	010576	020276

EVENT DESCRIPTION

02 Local leak rate testing revealed an excessive leak rate of 84.42

03 SCFH through the HPCI exhaust check valve. A defective gasket

04 between the valve seat and body was discovered and replaced

05 with the unit in cold shutdown for a

06 refueling outage. In the event of high torus pressure

	SYSTEM CODE (9–10)	CAUSE CODE (11)	COMPONENT CODE (12–17)	PRIME COMPONENT SUPPLIER (43)	COMPONENT MANUFACTURER (44–47)	VIOLATION (48)
07 (7–8)	SF	E	VALVEX	A	C502	N

CAUSE DESCRIPTION

08 A defective gasket was discovered between the valve seat

09 and body. A new gasket was formed on-site and used in

10 the valve repair.

	FACILITY STATUS (9)	% POWER (10–12)	OTHER STATUS (13–44)	METHOD OF DISCOVERY (45)	DISCOVERY DESCRIPTION (46–80)
11 (7–8)	H	000	NA	B	SURVEILLANCE Test

	FORM OF ACTIVITY RELEASED (9)	CONTENT OF RELEASE (10)	AMOUNT OF ACTIVITY (11–44)	LOCATION OF RELEASE (45–80)
12 (7–8)	Z	Z	NA	NA

PERSONNEL EXPOSURES

	NUMBER (9–11)	TYPE (12)	DESCRIPTION (13–80)
13 (7–8)	000	Z	NA

PERSONNEL INJURIES

	NUMBER (9–11)	DESCRIPTION (12–80)
14 (7–8)	000	NA

OFFSITE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

	TYPE (9)	DESCRIPTION (10–80)
16 (7–8)	Z	NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 Event Discription (cont'd) - a manual stop check valve exists which could

19 have been closed to prevent backflow to the HPCI turbine (cont'd)

NAME: MICHAEL P. FLASCH PHONE: 309-654-2241 EXT. 247

76-2

REPORT NUMBER: RO 50-254/76-2

REPORT DATE: February 2, 1976

OCCURRENCE DATE: January 5, 1976

FACILITY: Quad-Cities Nuclear Power Station
Cordova, Illinois 61242

IDENTIFICATION OF OCCURRENCE:

Excessive leakage through the High Pressure Coolant Injection System (HPCI) steam exhaust check valve 1-2301-45, as determined during local leak rate testing.

CONDITIONS PRIOR TO OCCURRENCE:

The Unit One reactor was in the REFUEL mode for refueling. The HPCI turbine exhaust manual valve, 1-2301-74, was closed.

DESCRIPTION OF OCCURRENCE:

While performing local leak rate tests on primary containment isolation valves, it was determined that the HPCI steam exhaust check valve had a measured leak rate of 84.42 SCFH which was in excess of the 18.36 SCFH limit allowed by Technical Specification 4.7.A.2.i.(2)(b) (5% L_{to}).

The leak rate test was performed by pressurizing the pipe volume between valves 1-2301-45 and 74 to 48 psig through a 3/4 inch pressure test point.

Work Request 42-76 was issued to initiate repairs.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

Equipment Failure

The cause of the excessive leakage through the check valve was failure of the valve gasket between the valve body seat ring and the disc.

ANALYSIS OF OCCURRENCE:

When operating under accident conditions, the HPCI exhaust passes through check valve 1-2301-45 and stop check valve 1-2301-74, and into the suppression chamber. In the event of high suppression chamber pressure, the stop check valve, 1-2301-74, could be closed to prevent back flow through the check valve to the HPCI turbine.

The back leakage through check valve 1-2301-45 does not render the HPCI system inoperable. Therefore, safe plant operation was not jeopardized. Since the HPCI room is not normally manned and is enclosed by the secondary containment, no abnormal radiation exposures would be experienced by plant personnel and public health and safety would not be affected.

CORRECTIVE ACTION:

The Maintenance Department disassembled the check valve and found that the gasket on the valve seat was cracked and chipped in numerous places. As soon as final repair of the valve is complete, another leak rate test will be conducted. The results of the repairs and subsequent leak test will be provided in a supplemental report.

FAILURE DATA:

On December 30, 1974, the Unit Two HPCI exhaust check valve was found to leak at the rate of 62.2 SCFH. The excessive leakage was corrected by remachining the valve seat, resulting in a post-repair leakage of 0.4 SCFH.

This check valve is a 24 inch swing check manufactured by Mission Valve and Pump Co., Model #15 SMF B402.

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

	LICENSEE NAME	LICENSE NUMBER	LICENSE TYPE	EVENT TYPE
01	ILQAD1	00-00000-00	41111	01

	CATEGORY	REPORT TYPE	REPORT SOURCE	DOCKET NUMBER	EVENT DATE	REPORT DATE
01 CON'T	MI	L	L	050-0254	011076	020276

EVENT DESCRIPTION

02 Local leak rate testing revealed excessive leakage through Feed
03 Water Check valves 1-220-58B, 62A & 62B. These valves have leaked
04 in the past, and viton "O" rings were installed on the valve
05 seating surfaces. The deterioration of the viton seals is now under
06 evaluation by CeCo Station Nuclear Engineering & Water Chemistry departments.

	SYSTEM CODE	CAUSE CODE	COMPONENT CODE	PRIME COMPONENT SUPPLIER	COMPONENT MANUFACTURER	VIOLATION
07	CH	E	VALVEX	N	C665	N

CAUSE DESCRIPTION

08 The apparent cause of excessive leakage is deterioration
09 of the viton seals which are located between the valve body
10 and trim assembly. F.W. check valves are 18 in., Manufactured

	FACILITY STATUS	% POWER	OTHER STATUS	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION
11	H	000	NA	B	SURVEILLANCE TEST

	FORM OF ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY	LOCATION OF RELEASE
12	Z	Z	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION
13	000	Z	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION
14	000	NA

OFFSITE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

	TYPE	DESCRIPTION
16	Z	NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 Event Description (cont'd) - (RO 50-254/76-2)

19 Cause Description (cont'd) - by Crane Co., Model 783-U

NAME: J.C. Vahrenwald [illegible] PHONE: 309-654-2241 - Ext. 248

REPORT NUMBER: RO 50-254/76-2

REPORT DATE: February 2, 1976

OCCURRENCE DATE: January 8, 1976

FACILITY: Quad-Cities Nuclear Power Station
Cordova, Illinois 61242

IDENTIFICATION OF OCCURRENCE:

Excessive leakage measured from feedwater check valves 1-220-58B, 1-220-62A, and 1-220-62B during local leak rate testing.

CONDITIONS PRIOR TO OCCURRENCE:

Unit One was in the REFUEL mode for refueling outage.

DESCRIPTION OF OCCURRENCE:

At 7:35 p.m. on January 8, 1976, feedwater check valve 1-220-58B failed to pass the local leak rate test limitation of 18.36 SCFH at 48 psig as allowed by Technical Specification 4.7.A.2.i(2)(b). The test was performed by closing manual valve 1-220-57B and pressurizing the pipe volume between this valve and the check valve to 48 psig. The as-found leakage of valve 1-220-58B was 3026 SCFH. Work Request 2755-75 had previously been written to repair the 1-220-58B valve.

At 10:00 a.m. on January 10, 1976 and at 12:00 noon on January 17, 1976, feedwater check valves 1-220-62A and 1-220-62B were also found to leak excessively when tested at 48 psig. The measured leakage for these valves was 3040 SCFH for the 1-220-62A valve and 378 SCFH for the 1-220-62B valve. Work Requests 120-76 and 189-76 were issued to repair the 1-220-62B and 1-220-62A valves, respectively.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

Equipment Failure

The cause of excessive leakage is believed to be deteriorated viton seals which are located between the valve body and trim assembly. These viton seals and related sealing surfaces will be evaluated by the CECO Station Nuclear Engineering Department and the Generating Stations Department Water Chemistry Group.

ANALYSIS OF OCCURRENCE:

If the unlikely situation is assumed where a feedwater line would rupture between valves 1-220-58A and 1-220-62A or valves 1-220-58B and 1-220-62B, the inboard check valves in either case would determine the amount of leakage.

This leakage would be contained by the drywell or in the least case by the Main Steam Isolation Valve (MSIV) room and reactor building secondary containment. The main contents of this leakage would be feedwater, steam from the reactor steam space, and fission product gases. The water could leak only until the vessel level was lowered to the feedwater spargers. Any steam that leaked would condense, and this water or condensate would be contained inside the plant by the floor drain system. Any gases that escaped would be handled by the Standby Gas Treatment Systems. At no time would the fuel be in danger of being uncovered due to this situation. When deinerted, a drywell entry could be made and the 1-220-57A or 57B valves could be closed preventing any further leakage. This type of worst-case analysis puts the plant in an inoperable status, but no radioactivity would be released to the environs and the health and safety of the public would not be affected.

CORRECTIVE ACTION:

Corrective action to be taken will be dependent upon the recommendations and suggestions made by the Station Nuclear Engineering Department and the Water Chemistry Group. Following completion of their analyses, and review of their recommendations, repairs will be made on the two feedwater check valves. A leak rate test will then be performed on these valves to verify acceptable leakage. Details relative to the corrective actions taken and the final leakage values will be provided in a supplemental report.

FAILURE DATA:

The feedwater check valves for Unit 1 had been leak tested once, prior to the current outage, in April 1974. As-found leakages ranged from 646.5 SCFH to 1332.0 SCFH for valves 1-220-58A, 58B, 62A, and 62B. A modification was initiated to replace the stainless steel seat rings with Viton components. Following completion of this change, leak rates ranged from 0.0 SCFH to 11.1 SCFH.

The Unit 2 feedwater check valves were tested in December 1974, and were found to have leakages ranging from 97 SCFH to 2647 SCFH. The same modification was performed on these valves as was done for Unit 1. Leak rate measurements made subsequent to this modification showed leakages ranging from 0.0 SCFH to 10.3 SCFH.

The feedwater check valves are 18" check valves manufactured by Crane Co., Model 783-U.

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

	LICENSEE NAME	LICENSE NUMBER	LICENSE TYPE	EVENT TYPE
01	I L Q A D 1	0 0 - 0 0 0 0 0 - 0 0	4 1 1 1 1	0 1

	CATEGORY	REPORT TYPE	REPORT SOURCE	DOCKET NUMBER	EVENT DATE	REPORT DATE
01 CON'T	M I	L	L	0 5 0 - 0 2 5 4	0 1 1 4 7 6	0 2 0 2 7 6

EVENT DESCRIPTION

02 MO 1-1001-37A failed to pass local leak rate requirments

03 as specified in Tech. Specs. The "Limitorque" valve operator

04 torque switch was readjusted, and the MO 1-1001-37A

05 valve was tested satifactorily. Failed leak rate

06 test with 249 SCFH - passed with 6.048 SCFH. MO 1-1001-37A

	SYSTEM CODE	CAUSE CODE	COMPONENT CODE	PRIME COMPONENT SUPPLIER	COMPONENT MANUFACTURER	VIOLATION
07	C F	E	V A L V O P	A	L 2 0 0	N

CAUSE DESCRIPTION

08 Valve operator torque switch was out of adjustment.

09 The "limitorque" valve operator is on 6"

10 globe valve by Crane Co. (Limitorque MOD. SMB)

	FACILITY STATUS	% POWER	OTHER STATUS	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION
11	H	0 0 0	NA	B	SURVEILLANCE Test

	FORM OF ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY	LOCATION OF RELEASE
12	Z	Z	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION
13	0 0 0	Z	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION
14	0 0 0	NA

OFFSITE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

	TYPE	DESCRIPTION
16	Z	NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 Event Discription (cont'd) - was operable and could

19 have provided primary containment. (RO 50-254/76-2)

NAME: Ted L. kou PHONE: 1-309-654-2241 (Ext. 252)

REPORT NUMBER: RO 50-254/76-2

REPORT DATE: February 2, 1976

OCCURRENCE DATE: January 4, 1976

FACILITY: Quad-Cities Nuclear Power Station
Cordova, Illinois 61242

IDENTIFICATION OF OCCURRENCE:

Excessive leakage measured from primary containment isolation valves MO-1-1001-34A, 36A, and 37A during local leak rate testing.

CONDITIONS PRIOR TO OCCURRENCE:

Unit 1 was in the REFUEL mode for a refueling outage.

DESCRIPTION OF OCCURRENCE:

At 11:00 p.m. on January 14, 1976, the piping volume between Residual Heat Removal System (RHRS) valves MO-1-1001-34A, MO-1-1001-36A, and MO-1-1001-37A failed to meet local leak rate test limitations as specified in Section 4.7.A.2.i(2)(b) of the Technical Specifications.

These valves were leak rate tested by pressurizing the piping volume between the valves with air at 48 psig pressure. The leak rate was determined to be 249 SCFH.

To determine which valves were leaking, the piping volume was pressurized with clean demineralized water. Water was found leaking through the 1-1001-37A valve to the suppression chamber spray header. The test coordinator then had an equipment operator manually tighten down on the MO-1-1001-37A valve using the valve motor handwheel, and the flow from the suppression chamber spray header stopped. From this it was concluded that the valve seating surfaces were good, but the motor operator for the MO-1-1001-37A was not allowing the valve to fully seat following a CLOSE signal. A torque switch problem was suspected and Work Request No. 163-76 was issued to repair the motor operator.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

Equipment Failure

The apparent cause of this occurrence is designated as equipment failure. The MO-1-1001-37A valve operator torque switch required readjustment. The improper close torque setting on the switch caused the valve operator to stop prior to complete seating of the valve disc.

ANALYSIS OF OCCURRENCE:

Valve MO-1-1001-37A is used only under post accident conditions to provide water to the pressure suppression chamber spray nozzles for the purpose of reducing pressure. It is interlocked closed during LPCI injection, and can only be used after actuation of a keylock switch and if drywell pressure is greater than 1 psig. This valve is normally exercised during the monthly containment cooling valve operability surveillance test.

Although the volume enclosed by RHRS valves MO-1-1001-34A, MO-1-1001-36A, and MO-1-1001-37A was found to experience excessive leakage, the cause was attributed to only one valve. The volume could have been successfully isolated due to the fact that MO-1-1001-34A, which is located upstream of MO-1-1001-37A, was not found to be leaking excessively. Valve MO-1-1001-37A was at all times fully capable of performing its intended function in the required manner.

Therefore, based on the fact that this valve is operated, other than for test purposes, only during post accident conditions, and the fact that MO-1-1001-34A was intact to provide additional isolation capability, the safety of the reactor and the health and safety of the general public were in no way affected by this occurrence.

CORRECTIVE ACTION:

The corrective action to prevent repetition of this occurrence was to re-adjust the torque switch. After an operability test, MO-1-1001-37A was leak tested in conjunction with MO-1-1001-34A and MO-1-1001-36A on January 27, 1976. The measured leakage was 6.048 SCFH, which was less than the 18.36 SCFH (5% L_{to}) limit given in the Technical Specifications.

FAILURE DATA:

In April, 1973 a program was started to replace an older model Torque Switch with a new model Torque Switch having a stronger return spring. The MO-1-1001-37A valve motor operator torque switch had been changed under Work Request 2772-73.

Valve MO-1-1001-37A has been tested for leakage on a prior occasion, and the results were satisfactory. There have been no other problems with the Limitorque valve operator on this valve since the new type torque switch was installed.

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

01 LICENSEE NAME (9–14)	I L Q A D 1
LICENSE NUMBER (15–25)	0 0 - 0 0 0 0 0 - 0 0
LICENSE TYPE (26–30)	4 1 1 1 1
EVENT TYPE (31–32)	0 1

01 CON'T CATEGORY (57–58)	M I
REPORT TYPE (59)	L
REPORT SOURCE (60)	L
DOCKET NUMBER (61–68)	0 5 0 - 0 2 5 4
EVENT DATE (69–74)	0 1 2 6 7 6
REPORT DATE (75–80)	0 2 0 2 7 6

EVENT DESCRIPTION

02 Volume between Unit 1 cleanup suction isolation
03 valves 1-1201-2 and 1-1201-5 failed local leak rate
04 test. Maintenance determined that valve 1-1201-2
05 operator torque switch malfunction prevented
06 valve from fully closing. Torque switch was

07 SYSTEM CODE (9–10)	C G
CAUSE CODE (11)	E
COMPONENT CODE (12–17)	V A L V O P
PRIME COMPONENT SUPPLIER (43)	A
COMPONENT MANUFACTURER (44–47)	L 2 0 0
VIOLATION (48)	N

CAUSE DESCRIPTION

08 Inadequate design of operator torque switch
09 by Limitorque (MOD# SMB-00). Replaced
10 torque switch.

11 FACILITY STATUS (9)	H
% POWER (10–12)	0 0 0
OTHER STATUS (13–44)	NA
METHOD OF DISCOVERY (45)	B
DISCOVERY DESCRIPTION (46–80)	SURVEILLANCE Test

12 FORM OF ACTIVITY RELEASED (9)	Z
CONTENT OF RELEASE (10)	Z
AMOUNT OF ACTIVITY (11–44)	NA
LOCATION OF RELEASE (45–80)	NA

PERSONNEL EXPOSURES

13 NUMBER (9–11)	0 0 0
TYPE (12)	Z
DESCRIPTION (13–80)	NA

PERSONNEL INJURIES

14 NUMBER (9–11)	0 0 0
DESCRIPTION (12–80)	NA

OFFSITE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

16 TYPE (9)	Z
DESCRIPTION (10–80)	NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 EVENT DESCRIPTION (CONT'D) - replaced & satisfactorily
19 tested volume between valves. (RO 50-254/76-2)

NAME: Ted Lihou PHONE: 1-309-654-2241 (Ext 252)

GPO 881-667

REPORT NUMBER: RO 50-254/76-2

REPORT DATE: February 2, 1976

OCCURRENCE DATE: January 26, 1976

FACILITY: Quad-Cities Nuclear Power Station
Cordova, Illinois 61242

IDENTIFICATION OF OCCURRENCE:

Excessive leakage measured from primary containment isolation valves MO-1-1201-2 and MO-1-1201-5 during local leak rate testing.

CONDITIONS PRIOR TO OCCURRENCE:

Unit 1 was in the REFUEL mode for a scheduled refueling outage.

DESCRIPTION OF OCCURRENCE:

At 1:00 p.m. on January 26, 1976, the piping volume between Reactor Water Cleanup System pump suction valves MO-1-1201-2 and MO-1-1201-5 could not be pressurized to 48 psig for local leak rate testing. The test coordinator then had an equipment operator manually tighten down on the MO-1-1201-2 valve using the valve motor handwheel. The volume was again tested, and pressurized to 48 psig. The leak rate was found to be 2.48 SCFH, which was within limits specified in Section 4.7.A.2.i(2)(b) of the Technical Specifications. From this it was concluded that the valve seating surfaces were good, but the motor operator for the MO-1-1201-2 valve was not allowing the valve to fully seat after having been given a CLOSE signal. A torque switch problem was suspected and Work Request No. 295-76 was issued to repair the motor operator.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

Equipment Failure

The apparent cause of this occurrence is designated as equipment failure. The electricians determined on January 27, 1976 that the torque switch on the valve was defective. The mode of failure on this particular torque switch was a weak return spring. It was discovered that an old model SMB-00 torque switch was still installed on the 1-1201-2 valve. This older model torque switch was then replaced with the new type which was used in the SMB-00 and SMB-000 torque switch replacement program of April 1973 (W.R. 5411-74).

ANALYSIS OF DEVIATION:

The safety implications of this deviation are minimized by the fact that valve MO-1-1201-2 would have closed on a group three primary containment isolation signal. Additionally, the other Reactor Cleanup isolation valve MO-1-1201-5 was shown not to have been leaking excessively. Therefore, the total actual leakage from the containment through this piping would have been small. Primary containment integrity was not compromised, and both MO-1-1201-2 and MO-1-1201-5 were fully capable of performing their intended function in the required manner. Thus, the health and safety of the public were not affected by this occurrence.

CORRECTIVE ACTION:

The corrective action to prevent repetition of this occurrence was to replace the defective torque switch with a new one with a stronger return spring. After an operability test, MO-1-1201-2 was leak tested in conjunction with MO-1-1201-5 on January 27. The measured leakage was 2.28 SCFH, which was less than the 18.36 SCFH (5% L_{to}) limit given in the Technical Specifications.

FAILURE DATA:

Valve MO-1-1201-2 had been tested for leakage on a prior occasion, and the results were satisfactory. Also, there have not been any past problems with the valve motor operator or torque switch.

Valve MO-2-1201-2 experienced a problem on December 20, 1974, whereby the valve would not close during a quarterly primary containment isolation valve exercise surveillance test. The torque switch had malfunctioned, and was replaced.