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ATTACHMENT 2

ComEd

LWP-95-034

March 29, 1995

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Reference: Quad Cities Nuclear Power Station
Docket Number 50-265, DPR-30, Unit 2

Enclosed is Licensee Event Report (LER) 95-001, Revision 00, for Quad Cities Nuclear Power Station.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(ii)(B). The licensee shall report any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear plant being in a condition that was outside the design basis of the plant.

The following commitments are being made by this letter:

A supplemental report will be issued which will document the valves and penetrations that had excessive leakage, what caused the leakage, and the corrective actions taken to bring all excessive primary containment leakages below the required LLRT limits.

If there are any questions or comments concerning this letter, please refer them to Nick Chrissotimos, Regulatory Assurance Supervisor at 309-654-2241, ext. 3100.

Respectfully,

COMMONWEALTH EDISON
QUAD CITIES NUCLEAR POWER STATION

D.B. Cook for

L. W. Pearce
Station Manager

LWP/TB/lls

Enclosure

cc: J. Schrage
C. Miller
INPO Records Center
NRC Region III

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LICENSEE EVENT REPORT (LER) Form Rev. 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Quad Cities Unit Two	0 5 0 0 0 2 6 5	1 of 0 4

Title (4)

Valve/Penetration Technical Specification Leakage Limits Exceeded During Local Leak Rate Testing

Event Date (5)			LER Number (6)					Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year		Sequential Number		Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
												0 5 0 0 0
0 3	0 7	9 5	9 5	–	0 0 1	–	0 0	0 4	0 5	9 5		0 5 0 0 0

OPERATING MODE (9)	02	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)
POWER LEVEL (10)	0 0 0		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	
Nick Chrissotimos, Regulatory Assurance, Ext. 3100	3 0 9	6 5 4 - 2 2 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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

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

A. ABSTRACT

At approximately 1630 on March 7, 1995, Quad Cities Unit-2 was shutdown for the cycle 13 refueling and maintenance outage. After Local Leak Rate Testing (LLRT) the "C" Main Steam Isolation Valves (MSIV), it was determined that the measured leakage rate of 13.6 standard cubic feet per hour (SCFH) in the outboard MSIV AO-2-203-2C exceeded the individual MSIV Technical Specification (3.7.A.2.a.3) leakage limit of 11.5 SCFH.

The inboard MSIV AO-2-203-1C had a measured leakage rate of 0.8 SCFH.

On March 8, 1995, at 1329 hours while performing LLRT on the Unit-2 Inboard Feedwater (FW) check valve 2-220-58B, the valve failed with an unquantified leakage.

The unquantified leakage exceeds the 0.6 La (293.75 SCFH) combined leakage limits specified in Technical Specification 3.7.A.2.a.2. The outboard feedwater 2-220-62B check valve had a measured leakage rate of 0.0 SCFH.

The root cause of the excessive leakages will not be known until the components have been disassembled and inspected. A supplemental report will be issued when all as found LLRT's are complete and the cause of excessive primary containment leakages are identified.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2511 MWt rated core thermal power.

EVENT IDENTIFICATION:

A. CONDITIONS PRIOR TO EVENT:

Unit: Two	Event Date: March 7, 1995	Event Time: 1630
Reactor Mode: 2	Mode Name: Refuel	Power Level: 0

This report was initiated by Licensee Event Report 265\95-001.

REFUEL (2) - In this position interlocks are established so that one control rod only may be withdrawn when flux amplifiers are set at the proper sensitivity level and the refueling crane is not over the reactor. Also, the trip from the turbine control valves, turbine stop valves, main steam isolation valves, and condenser vacuum are bypassed. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

B. DESCRIPTION OF EVENT:

At approximately 1630 on March 7, 1995, Quad Cities Unit 2 was shutdown for the cycle 13 refueling and maintenance outage. After Local Leak Rate Testing (LLRT) the "C" Main Steam Isolation Valves (MSIV) [SB] [ISV], it was determined that the measured leakage rate of 13.6 standard cubic feet per hour (SCFH) in the outboard MSIV AO-2-203-2C exceeded the individual MSIV Technical Specification (3.7.A.2.a.3) leakage limit of 11.5 SCFH.

The inboard MSIV AO-2-203-1C had a measured leakage rate of 0.8 SCFH.

On March 8, 1995, at 1329 hours while performing LLRT on the Unit 2 inboard feedwater (FW) check valve [SJ] [FCV] 2-220-58B, the valve failed with an unquantified leakage. The outboard feedwater 2-220-62B check valve had a measured leakage of 0.0 SCFH.

The unquantified leakage exceeds the 0.6 La (293.75 SCFH) combined leakage limits specified in Technical Specification 3.7.A.2.a.2 (combined leakage rate of all penetrations and valves, except the MSIV's, subject to type B and C tests of 10CFR, Appendix J).

The MSIV, the FW check valve, and any other primary containment valves or penetrations that fail LLRT with excessive leakage will be reported under the supplemental report to this LER.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with the requirements of the 10CFR 50.73(a)(2)(ii)(B). Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers being seriously degraded, or that resulted in the nuclear plant being in a condition that was outside the design basis of the plant.

The root cause of the excessive leakages will not be known until the components have been disassembled and inspected. A supplemental report will be issued when all as found LLRT's are complete and the cause of excessive primary containment leakages are identified.

D. SAFETY ANALYSIS OF EVENT:

The safety consequences of this event were minimal since LLRT maximum pathway is a conservative method for quantifying containment leakage. Containment leakage is specified by two separate leak pathways. The minimum pathway and the maximum pathway. The minimum pathway adds up all the leakages of the best valves of each pathway. The maximum path adds up all the leakages of the worst valves of each pathway, assuming the best valves of each pathway fail.

The actual leakage under accident conditions would be less than that determined by the LLRT maximum pathway. Where only the maximum pathway limit is exceeded as in this case, more than one valve is present in a line, and it is realistic to expect the leakage to be equal to the lesser leakage of the two valves.

E. CORRECTIVE ACTIONS:

A supplemental report will be issued which will document the valves and penetrations that had excessive leakage, what caused the leakage, and the corrective actions taken to bring all excessive primary containment leakages below the required LLRT limits (NTS# 265180950010l).

F. **PREVIOUS EVENTS**:

A three year Nuclear Tracking System (NTS) historical search, for LER's associated with valve LLRT testing exceeding Technical Specifications, was conducted. The results of the search are included below.

LER #	Description
265/92-002	Unit-2 Valves 2-1601-21(22)(55)(56) Failed LLRT Leakage Limit.
254/92-020	Unit-1 Tech Spec Containment LLRT Exceeded Due To Various Component Failures.
254/93-007	Unit-1 B-Loop Main Steam Isolation Valve Local Leak Rate Exceeded 11.5 SCFH.
265/93-025	Unit-2 A-Loop Main Steam Isolation Valve Local Leak Rate Exceeded 11.5 SCFH.
254/94-005	Unit-1 A-Loop Main Steam Isolation Valve Local Leak Rate Exceeded 11.5 SCFH.

G. **COMPONENT FAILURE DATA**:

Component failure data is not available at this time since repairs have not been completed. Failure data will be included in the Supplemental Report.