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Quad Cities Nuclear Power Station
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DCD

RLB-90-307

December 12, 1990

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

Reference: Quad Cities Nuclear Power Station
Docket Number 50-254, DPR-29, Unit One

Enclosed is Licensee Event Report (LER) 90-029, 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)(i)(B): The licensee shall report any operation or condition prohibited by the plant's Technical Specifications.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD CITIES NUCLEAR POWER STATION

R.A. Robey for
R. L. Bax
Station Manager

RLB/MJB/jmt

Enclosure

cc: R. Stols
T. Taylor
INPO Records Center
NRC Region III

IE22

3216H

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

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

Title (4)
Exceedence of Technical Specification Local Leak Rate Test Limit 0.6 La While Testing Feedwater Check Valve 1-220-62B, Cause to be Determined.

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
										0 5 0 0 0
1 1	1 5	9 0	9 0	0 2 9	0 0	1 2	1 2	9 0		0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 0

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)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	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	
J. Rolfes, Tech Staff Engineer Ext. 2160	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
X	S J	I S V	C 6 8 4	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
X Yes (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 2	2 8	9 1

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

ABSTRACT:

On November 12, 1990, Quad Cities Unit One was shutdown for the end of cycle 11 refueling and maintenance outage. On November 15, 1990, at 0700 hours while performing local leak rate testing (LLRT) of the Drywell Personnel Air Lock, it was determined that the Technical Specification 3.7.A.2.d. leakage limit of 18.4 standard cubic feet per hour (SCFH), 0.0375La, was exceeded.

On November 15, 1990, at 2325 hours while performing LLRT on the Feedwater Check Valves, valve 1-220-62B could not be pressurized. The Technical Specification 3.7.A.2.a.2, limit of 293.75 SCFH (0.6La) was exceeded.

An Emergency Notification System (ENS) phone call was completed on November 16, 1990 at 0318 hours in accordance with 10CFR50.72(b)(2)(i).

The cause of the excessive leakages will not be known until repairs have been completed. A supplemental report will be issued to address the causes and corrective actions taken to bring the combined leakage to within Technical Specification limits. This report is being submitted to comply with 10CFR50.73(a)(2)(i)(B).

PLANT AND SYSTEM IDENTIFICATION:

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

EVENT IDENTIFICATION: Exceedence of Technical Specification Local Leak Rate Test Limit 0.6 La While Testing Feedwater Check Valve 1-220-62B, Cause to be Determined.

A. CONDITIONS PRIOR TO EVENT:

Unit: One	Event Date: November 15, 1990	Event Time: 2325
Reactor Mode: 2	Mode Name: REFUEL	Power Level: 00%

This report was initiated by Deviation Report D-4-1-90-122

Refuel Mode (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:

On November 12, 1990 Unit One was shutdown for the end of cycle 11 refueling and maintenance outage.

On November 15, 1990, at 0700 hours while performing QTS 100-50, Local Leak Rate Test (LLRT) of the Drywell personnel air lock [NH][AL], the as found leakage rate was 27.5 standard cubic feet per hour (SCFH). A leak was found on an electrical conduit [CND] which penetrates the outboard side of the air lock. This exceeded the Technical Specification 3.7.A.2.d. limit of 18.4 SCFH (0.0375La).

On November 15, 1990, at 2325 hours while performing Temporary Procedure 6329, LLRT of the Unit One Feedwater Check Valves [SJ][ISV], the outboard valve, 1-220-62B, could not be pressurized. Therefore, the leakage rate of 1-220-62B was undetermined. The Technical Specification 3.7.A.2.a.2 limit of 293.75 SCFH (0.6La) combined leakage from all valves [NH][ISV] and penetrations [NH][PEN] except Main Steam [SB] Isolation Valves [ISV](MSIV), was exceeded.

On November 16, 1990, at 0318 hours an Emergency Notification System (ENS) phone call was completed in accordance with 10CFR50.72(b)(2)(i).

C. APPARENT CAUSE OF EVENT:

This report is being submitted to comply with the requirements of 10CFR50.73(a)(2)(i)(B) which states that the licensee shall report any operation or condition prohibited by the plant's Technical Specifications.

The cause of the excessive leakages cannot be determined until repairs have been completed and the valves and components have been retested. A supplemental report documenting the repairs and corrective actions taken will be issued.

D. SAFETY ANALYSIS OF EVENT:

The safety consequences of the event were minimal since the maximum pathway leakages are used for comparison to the total allowable leakage for Type B and C tests (0.6La), which is a conservative method. The summation of the maximum pathway leakage rates assumes that for each potential air leakage through primary containment [NH], the best valve or barrier fails and the leakage through the pathway equals the leakage of the worst valve or barrier.

The summation of the minimum pathway leakages, which yields a more realistic total leakage, is used for determining the acceptance of the Type A test results. This method assumes that the best valve or barrier for each pathway remains intact. Although not all of the required as found Type B and C testing is complete, the total as found minimum pathway leakage for the pathways already tested are still within the allowable limit (0.75La).

The actual leakage of the feedwater lines under accident conditions is expected to be less than the tested value since the lines would be filled and pressurized with water rather than air, and the lines could possibly be isolated using non-primary containment isolation (PCI) valves [NG][ISV] on the upstream side of containment. The Standby Gas Treatment (SBGT)[BH] system was operable to provide an additional safety barrier.

E. CORRECTIVE ACTIONS:

The cause of the excessive leakages will not be known until repairs have been completed. The immediate corrective action was to initiate work requests to repair the 1-220-62B valve and the conduit for the Drywell Air Lock. A supplemental report will be issued to document the causes and corrective actions taken to bring the combined and the individual leakages below Technical Specification limits (NTS 2542009012201).

F. PREVIOUS EVENTS:

254/89-014 Leak Rate from all valves and penetrations Including MSIVs on Unit One in excess of Technical Specification Limit.

265/88-007 Leak rate from all valves and penetrations excluding MSIVs on Unit Two in excess of Technical Specification limit.

254/87-016 Leak rate from all valves and penetrations excluding MSIVs on Unit One in excess of Technical Specification limit.

265/86-014 Leak rate from all valves and penetrations excluding MSIVs on Unit Two in excess of Technical Specification limit.

These are the most recent related events; other similar events have occurred prior to 1986.

G. COMPONENT FAILURE DATA:

Component failure data will not be available until repairs have been completed. This data will be included in a supplemental report which will be completed following valve repairs.