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Quad Cities Nuclear Power Station
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RLB-93-025

February 9, 1993

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) 92-020, Revision 01, 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). 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.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD CITIES NUCLEAR POWER STATION

R. L. Bax
R. L. Bax
Station Manager

RLB/TB/plm

Enclosure

cc: J. Schrage
T. Taylor
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 One	0| 5| 0| 0| 0| 2| 5| 4	1 of 0 7

Title (4)

Tech Spec Containment Leakage Limit 0.6 La Exceeded

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| 9	2| 1	9| 2	9| 2	0 | 2 | 0	0 | 1	0 | 2	1| 1	9| 3		0| 5| 0| 0| 0| |

OPERATING MODE (9): 2

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	
David P. Kunzmann, LLRT Coordinator, Ext. 2162	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
X	S | J	F| C| V	C |6 |6 |5	Y	X	C | C	F| C| V	C| 6| 6| 5	Y
					X	B | B	F| C| V	A| 0| 3| 7	Y

SUPPLEMENTAL REPORT EXPECTED (14)

___ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

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

ABSTRACT:

On September 20, 1992, Quad Cities Unit One was shutdown for refueling maintenance (Q1R12). On September 21, 1992 at 11:30 hours, while performing Local Leak Rate Testing (LLRT) of the Atmospheric Containment, Atmospheric Dilution (ACAD) to Standby Gas Treatment (SBGT) Containment Isolation Valve AO-1-2599-5B [IK] [BH] it was determined that the Technical Specification 3.7.A.2.a.2 limit of 293.75 SCFH (0.6 La) was exceeded.

An Emergency Notification System (ENS) phone call was completed on September 21, 1992 at 1306 (EST) hours in accordance with 10CFR50.72(b)(2)(i).

The cause of the excessive leakages was identified and repairs have been completed. Additional failed valves were identified during subsequent testing of the remaining isolation valve volumes. This report is being submitted to comply with 10CFR50.73(a)(2)(ii).

LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

Form Rev 2.0

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Quad Cities Unit One	0 5 0 0 0 2 5 4	9 2	- 0 2 0 -	0 1	0 2 OF 0 7

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	F C V	C 6 6 5	No					
X	S J	F C V	C 6 6 5	No					
X	S J	F C V	C 6 6 5	No					
X	B O	F C V	C 6 6 5	No					
X	B R	F C V	C 6 6 5	No					
X	L F	F C V	H 0 3 7	No					
X	C G	F C V	P 3 4 0	No					
X	C G	F C V	P 3 4 0	No					
X	C G	F C V	P 3 4 0	No					
X	C G	F C V	P 3 4 0	No					
X	C G	F C V	P 3 4 0	No					
X	C G	F C V	P 3 4 0	No					
X	C C	F C V	C 6 6 5	No					

NRC Form 366B (9-83)

DVR 373

PLANT AND SYSTEM IDENTIFICATION:

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

EVENT IDENTIFICATION: Tech Spec Containment Leakage Limit 0.6 La Exceeded.

A. CONDITIONS PRIOR TO EVENT:

Unit: One Event Date: September 21, 1992 Event Time: 0715
Reactor Mode: 2 Mode Name: REFUEL Power Level: 00%

This report was initiated by Deviation Report D-4-01-92-104.

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 September 20, 1992, Quad Cities Unit One was shutdown for refueling maintenance (Q1R12) On September 21, 1992 at 11:30 hours, while performing Local Leak Rate Testing (LLRT) of the Atmospheric Containment, Atmospheric Dilution (ACAD) to Standby Gas Treatment (SBGT) Containment Isolation Valve AO-1-2599-5B [IK] [BH] it was determined that the Technical Specification 3.7.A.2.a.2 limit of 293.75 SCFH (0.6 La) was exceeded. The failure mode for this and other valves [V] tested during the refuel outage are detailed in Section E.

C. APPARENT CAUSE OF EVENT:

This report is being submitted to comply with the requirements of 10CFR50.73(a)(2)(ii) which states that 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 cause of the excessive leakages have been determined where possible; the repairs have been completed and the valves have been retested.

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 (0.6 La), 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. All of the required as found Type B and C testing is complete, the total as found minimum pathway leakage of 215.82 scfh is still within the allowable limit of 367.2 scfh (0.75La).

E. CORRECTIVE ACTIONS:

Corrective actions have been taken at this time. The causes and repairs taken to bring the combined leakage below the required limits are as follows:

Feedwater Check Valves 1-220-58A/62A and 58B/62B

The 58B valve was found to have a bonnet leak. This leak also attributed to the leakage of the 62B valve. The leak path was repaired. The 58A and 62A valves both showed significant hinge pin wear and an insufficient o-ring seating surface. The o-ring seating surface was widened by welding and the hinge pins were replaced.

RHR Torus Spray 1-1001-37A

The 30 scfh leakage was attributed to a scored seat. The valve disc was machined to improve the seat.

Standby Liquid Control Checkvalve 1-1101-16

Crud buildup was a contributing factor for the 28 scfh leakage. The valve was flushed with clean demin water which reduced the leakage.

Service Air to Drywell 1-4699-47

The seat of this one inch check valve was cleaned and lapped.

Drywell/Torus Exhaust 1-1601-23, 24, 60, 61, 62 and 63

The as found test method could not identify the specific leak paths(s). However, due to past poor performance of the 18 inch butterfly valves, a modification had been initiated and was completed during this outage. The modification replaced the valves with serviceable Jamesbury valves. In addition a separate modification completed this outage now provides for more specific valve testing.

Reactor Building Closed Cooling Water MOV-1-3703 and CV 1-3799-31

Valve 3703 had some seat leakage in addition to leakage of a boundary valve. The valve was rebuilt and an alternate test method eliminated the boundary valve from the test. The boundary valve is scheduled for repair during the next refuel outage. Check valve 1-3799-31 which is the original valve was found with spots of corrosion and splatter (debris) on disc seating surface. The valve was replaced.

ACAD to Standby Bus Treatment 1-2599-5B

This valve is an air operated globe valve, spring to close, air to open. The valve was found partially open due to a weak spring. The spring was replaced.

F. PREVIOUS EVENTS:

265/92-002 Local Leak Rate Test Leakage Limits for Unit One exceeded the Technical Specification Limit.

254/90-029 Leak rate from all valves and penetrations including MSIVs on Unit 1 exceeded the Technical Specification Limit.

265/90-003 Leak Rate from all valves and penetrations including MSIVs on Unit 2 exceeded the Technical Specification Limit.

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:

The failed components with undetermined leakages are listed below . The components listed with quantified leakages are those that exceeded an administrative limit set for preventive maintenance purposes and as a group make a significant contribution to the overall leakage total and are not considered NPRDS failures.

Failed Components	Manufacturer/Model	As Found/ As Left (scfh)
Feedwater Check Valves		
1-220-58A	Crane 18" 900 lb model 973	undetermined/1.2
Rx Bldg. Closed Cooling Water Supply		
CV-3799-31	Mission 8" crane model 15SMF402	undetermined/0.1
ACAD to SBGT		
1-2599-5B *Combined leakage of mutiple valves	Hancock Co. model 5580w	undetermined/2
High Leakage Components		
Feedwater check valve 1-220-62A	Crane 18" 900 lb model 973	52/4.5
1-220-58B	Crane 18" 900 lb model 973	19/0.4
1-220-62B	Crane 18" 900 lb model 973	27.5/1.1
RHR Torus Spray		
1-1001-37A	Crane 6" gate	30/11
Standby Liquid Control		
1-1001-16	1 1/2" Crane model #3888-U check valve	28/16

Service Air to Drywell

1-4699-47	Hancock Co, model 5580w 1" check valve	12/2.2

Drywell/Torus Exhaust

1-1601-23,24,60,61, 62, and 63	Pratt, Henry 18" 175 lb model 2F11w/d1200GAD	50/6

Rx Bldg. Closed Cooling Water Return

MO-3703	8" Crane model 47 - 1/2XR	36/1.9