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RLB-90-154

June 18, 1990

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

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

Enclosed is Licensee Event Report (LER) 90-009, 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. L. Bax
Station Manager

RLB/MJB/jlg

Enclosure

cc: R. Stols
T. Taylor
INPO Reco ds Center
NRC Regio 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 6

Title (4)

Various Containment Volumes not Leak Rate Tested Due to Recent 10CFR50, Appendix J Interpretation.

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 5	1 8	9 0	9 0	--- 0 0 9	--- 0 0	0 6	1 8	9 0		0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 0 9 5

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)	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	
M. Brown, Regulatory Assurance Ext. 3102	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

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)

ABSTRACT:

On May 18, 1990 at 1150 hours, Unit One was operating in the RUN mode at 95 percent of rated core thermal power.

At this time, the operability of the Unit One primary containment was concluded to be indeterminate which placed the Unit into Technical Specification section 3.0.A.

A temporary Waiver of Compliance from Technical Specifications was requested from the NRR and verbal approval was granted by the NRC on May 18, 1990 at 1510 hours.

As part of the corrective action, local leak rate testing (LLRT) was completed on two of the systems involved. Previously, a modification had been initiated to install the necessary equipment to perform the LLRTs. LLRTs will be performed on the remaining systems the next unit refuel outage. An emergency Technical Specification change has been submitted.

This report is being submitted in accordance with 10CFR50.73(a)(2)(i)(B).

2782H

PLANT AND SYSTEM IDENTIFICATION:

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

EVENT IDENTIFICATION: Various Containment Volumes not Leak Rate Tested due to Recent 10CFR50, Appendix J Interpretation.

A. CONDITIONS PRIOR TO EVENT:

Unit: One	Event Date: May 18, 1990	Event Time: 1150
Reactor Mode: 4	Mode Name: RUN	Power Level: 95%

This report was initiated by Deviation Report D-4-1-90-039

RUN Mode (4) - In this position the reactor system pressure is at or above 825 psig, and the reactor protection system is energized, with APRM protection and RBM interlocks in service (excluding the 15% high flux scram).

B. DESCRIPTION OF EVENT:

On May 18, 1990 at 1150 hours, Unit one was operating in the RUN mode at 95 percent of rated core thermal power. At this time, the operability of the Unit One primary containment [NH] was concluded to be indeterminate which placed the unit into Technical Specification section 3.0.A.

In December, 1989, a Commonwealth Edison Company (CECO) self assessment/improvement audit of the station's local leak rate testing (LLRT) program noted 29 containment pathways, 7 different systems, that had not been tested. However, these pathways were not required to be tested in the Final Safety Analysis Report (FSAR) or Technical Specification. Due to a recent interpretation of 10 CFR 50, Appendix J with respect to licensing design criteria, the station decided to add these pathways to the type C LLRT program. Further information was reported in voluntary Licensee Event Report (LER), 90-001 and Revision 1.

In April, 1990, during an inspection by the NRC, the NRC expressed concerns about the operability of the Unit One primary containment. The station was requested to show that there was no significant additional risk due to the untested pathways which was to include a combination of physical justification as well as a probability risk assessment (PRA)-based assessment.

CECo staff personnel met with the NRR and NRC Region III personnel on May 17, 1990, to present and discuss the operability aspect of the containment. On May 18, a management meeting between CECo and the NRC was weld at the NRC Region III headquarters. At this time, it was concluded that Unit One primary contanment was indeterminate.

The indeterminate condition of the Unit One primary containment resulted in a Technical Specification 3.0.A. limiting condition for operation (LCO). On-site review (OSR) 90-20 was initiated to request a Temporary Waiver of Compliance from the Technical Specification. The OSR was approved on May 18, 1990 and NRC verbal approval of the Temporary Waiver request was granted at 1510 hours. It was concluded that the added risk of plant operation until October 1990 without performing the Type C tests was insignificant and did not warrant an earlier plant shutdown merely to perform the tests.

On May 18, LLRT was completed on one of the systems involved. The Drywell Air Sampling System [IL] valves [SMV], 21 total, were successfully tested with no leakage observed.

On May 19, OSR 90-21 was initiated to submit an emergency Technical Specification change to sections 3.7.A.2, 4.7.A.2, and Table 4.7-1. Section 3.7.A.2 added statements to temporarily exclude the new pathways specified in section 4.7.A.2. Section 4.7.A.2 added a statement which identifies the pathways in Table 4.7-1 and excludes their LLRT testing until the end of cycle 11 refueling outage. Table 4.7-1 lists the temporarily untested pathways which involve the Instrument Air [LD], Reactor Building Closed Cooling Water (RBCCW) [CC], Core Spray [BM], Standby Liquid Control [BR] and Clean Demineralizer Water [KC] Systems. OSR 90-21 was approved and submitted to the NRC on May 19.

On May 22, 1990, the NRC reaffirmed the verbal approval for a Temporary Waiver of Compliance from Technical Specification 3.0.A. The Waiver of Compliance remains in effect until the emergency Technical Specification change is approved.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with 10CFR 50.73 (a)(2)(i)(B): The licensee shall report any operation or condition prohibited by the plants' Technical Specifications.

The cause of this event is due to a recent interpretation of 10 CFR 50, Appendix J with respect to licensing design criteria. Quad Cities was licensed prior to publication of 10 CFR 50, Appendix J and during the initial interpretation of Appendix J, these pathways were considered exempt from Type C LLRT requirements. During the company's self-assessment audit to improve the Type B and C LLRT program for the station, 29 pathways were discovered which should be included in the program. These pathways were not local leak rate tested previously since the isolation valves did not appear to meet the four criteria specified in 10 CFR 50, Appendix J as requiring LLRT, and since they are not specified in either the Technical Specifications or FSAR as Type C primary containment isolation valves.

The pathways for Unit Two have been tested. Unit One primary containment was concluded to be indeterminate as 5 of these pathways had not been tested because a unit shutdown was required to install the modification needed to complete the leak rate testing.

This condition placed the unit into a Technical Specification 3.0.A. limiting condition for operation (LCO). Technical Specification 3.0.A. LCO states that in the event an LCO cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours unless corrective measures are completed that satisfy the LCO.

D. SAFETY ANALYSIS OF EVENT:

The safety of the plant and personnel was not affected by this event. An evaluation of the safety significance and potential consequences was performed. The following discussion demonstrates that this event did not create an unsafe condition nor an increase in the potential consequences for reasonably postulated events during the period of interest:

A. No open pathways from primary containment to the reactor building, or other ancillary structures or the environment exists.

1) Clean Demineralized Water, Penetration X-20:

This pathway is a single three inch line that penetrates the primary containment. Normal isolation is achieved by a check valve and locked closed manual valve outside of containment. In addition to these two containment isolation valves, there exists a closed piping system. The entire system is pressurized with water at about 100 psig during unit operation. This water serves both to seal any potential leakage through the valves and to continuously demonstrate the integrity of the piping system. Any leakage of water from the closed piping inside of containment would be detected due to an increase in drywell sump level. The system is supplied by multiple pumps feeding a common header taking suction from a 100,000 gallon storage tank.

2) Core Spray System, Penetration X-16 A and B:

The Core Spray System is a low pressure emergency core cooling system which provides reactor coolant in the event of a Loss of Coolant Accident (LOCA). The system is pressurized with high pressure water, relative to Pa, during post accident conditions which acts as a seal water system for the containment isolation valves. The injection lines are equipped with remote testable check valves inside primary containment and two remotely operated gate valves outside containment. The check valve is subject to reactor pressure during normal operation. The system is also equipped with a pressure switch between the outboard isolation valves, 1402-24 A/B, which are normally open and the inboard isolation valves, 1402-25 A/B, which are normally closed. If valve 1402-25 A/B were to leak, the pressure switch would sense a higher than normal keep-fill pressure during normal operation.

3) Standby Liquid Control (SBLC) System, Penetration X-47

The one and one-half inch SBLC line which penetrates primary containment contains closed valves in addition to the containment isolation valves. These closed valves are squib valves which consist of solid metal caps which block the pathway unless actuated. The potential of a seat or packing leak, therefore, does not exist. The SBLC system is an engineered safety feature [ESF] and the squib valves are only actuated in the event that the control rod scram function fails and reactor power cannot be reduced using normal methods. The valves, therefore, would not be actuated during the design basis LOCA.

4) Instrument Air to the Drywell and Torus, Penetration X-216 and X-22

The instrument air system penetrates primary containment by two lines. The line which penetrates the drywell is a one inch line and that which penetrates the torus is a one-half inch. Containment isolation is achieved by one check valve inside containment and one check valve outside of containment. The penetrating lines are connected inside of containment to a closed piping system that does not interface with the drywell atmosphere. Outside of containment, the lines are connected to a closed piping system that does not interface with the Reactor Building Atmosphere. During normal operation, the primary containment lines are pressurized with nitrogen at a pressure of approximately 2 times Pa. This pressurization may serve as a valve sealing system in the event of a leak.

During the previous Integrated Leak Rate Test (ILRT), these lines were properly depressurized and vented outside of containment. The closed piping inside of containment, however, was not vented to the containment; therefore, the containment isolation valves were not adequately challenged. The ILRT was successfully completed which provides assurance that leaks were not present through the inside piping systems and the containment isolation valves. The ILRT and the operating configurations are similar except that the line outside of containment is not vented and the entire system is pressurized during normal operation.

5) Reactor Building Closed Cooling System (RBCCW), Penetration X-23 and X-24

The RBCCW system consists of two eight inch lines that penetrate primary containment. The supply line is normally isolated using a check valve inside and a remotely operated manual gate valve outside of containment. The return line contains two remotely operated valves, one inside and one outside of the drywell.

In addition to the two containment isolation valves on each line, additional barriers exist. Inside of the containment, the piping forms a closed loop. Outside of containment, the piping is configured such that loop water seals are created. The system is filled with pressurized water during normal operation. The water serves as a seal for potentially leaky valves and as a system leakage detection system. Any through-wall water leaks would be easily detected either inside or outside of the drywell through operational indicators (sump levels, system pressures, tank levels, etc.).

The piping outside of containment is connected to a vented surge tank. This tank receives makeup water supply by multiple pumps connected to a common header which provides suction from a 100,000 gallon storage tank. This configuration provides substantial assurance that the system would remain water-filled in post accident conditions.

B. The fission product barrier, i.e., the containment functions, would be maintained except for an extreme combination of improbable added failures.

A Risk Assessment was performed to further demonstrate that the probability of an event during the remainder of Unit 1 Cycle 11 which would result in a loss of containment functions coincident with a LOCA is insignificant. Through this evaluation, fission product barriers remained intact provided that an extreme combination of coincident failures (which is highly improbable) does not occur. The probabilities calculated for the event in which containment function failure would occur under LOCA conditions were, therefore found to be insignificant, well below 1E-7. For example, in the case of RBCCW, in order to experience a containment function failure, a recirculation piping failure, RBCCW pipe failure inside containment and a failure of the loop seal would have to occur. The probability of the failure of RBCCW system containment function and LOCA is 2E-10 and is therefore considered to be insignificant.

E. CORRECTIVE ACTIONS:

A Temporary Waiver of Compliance from Technical Specifications was initiated by the station and granted by the NRC on May 18, 1990.

Unit Two LLRT for the pathways involved has been completed. On Unit One, the Service Air System [LF] was successfully tested on November 17 and 19, 1989 and the Drywell Sample System [IL] was successfully tested on May 18, 1990.

Modification M4-1(2)-89-167 was initiated to install the necessary test taps for Unit One, refer to NTS 2542009000202. The station's Type B and C LLRT program was revised to include these seven pathways. Prior to Unit One start-up following the refuelng outage a Type C LLRT will be performed on all volumes including these pathways, refer to NTS 2542009000203. The Type A test procedure for Unit One will be revised to drain and vent these pathways where practical, refer to NTS 2542009000204.

In the interim, Operating Orders have been issued to give the operators guidance to ensure containment integrity remains intact. The operators are instructed to close the remotely operated valves on the RBCCW system when the Recirc pumps trip during a LOCA. THe RBCCW pumps will be kept on if possible to ensure the system is filled with water and pressurized above containment pressure. During a LOCA event if the RBCCW Expansion Tank HI/LO level alarm is received the GSEP Station Director will send field teams, as conditions permit, to check RBCCW piping outside containment to ensure integrity. The GSEP Station Director will take the necessary action to further isolate the system.

F. PREVIOUS EVENTS:

LER 90-001, Revision 1 (voluntary) was written to document the same condition for Unit Two. All the required testing has been completed.

G. COMPONENT FAILURE DATA:

There was no component failure associated with this event.