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Quad Cities Nuclear Power Station
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RLB-93-021

February 5, 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 in a Licensee Event Report (LER) 91-003, Revision 1, 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). 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 4

Title (4)

Specific Points In ACAD/CAM Lines Exceed UFSAR Allowable Stresses

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 1	1 8	9 1	9 1	0 0 3	0 1	0 2	1 1	9 3	Quad Cities Unit Two	0 5 0 0 0 2 6 5
										0 5 0 0 0

OPERATING MODE (9)	1
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	
Mark Bridges, Technical Staff Engineer Ext. 2944	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		0 6	0 1	9 6

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

On January 18, 1991, at 1532 hours, Unit One was in the SHUTDOWN mode at 0 percent of rated core thermal power. At this time, the Boiling Water Reactor Systems Design department notified the station that twelve specific points on four Containment Atmospheric Monitoring (CAM)[IP] and Atmospheric Containment Atmosphere Dilution (ACAD) [BB] lines exceeded UFSAR allowable stresses. This determination was made during analyses for Modification 4-1-88-103. Further analyses showed that all piping was operable. The cause of the event was preservice error involving the design and construction of the affected lines. No immediate corrective action is required since all lines are operable. This condition will be resolved when Modification M04-1-88-103 is implemented. Subsequent walkdowns and analysis on the Unit 2 ACAD/CAM piping revealed that support modifications would be required to bring this piping within UFSAR/FSAR design margins. Operability for the Unit 2 piping was verified in January, 1991. These necessary support modifications/additions for Unit 2 will be included as part of modification No. M04-2-88-103 (Partial "B"). This report is being submitted in accordance with the requirements of the Code of Federal Regulations Title 10 Part 50.73 (a)(2)(ii)(B).

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

PLANT AND SYSTEM IDENTIFICATION:

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

EVENT IDENTIFICATION: Specific Points In ACAD/CAM Lines Exceed UFSAR Allowable Stresses.

A. CONDITIONS PRIOR TO EVENT:

Unit: One	Event Date: 01-18-91	Event Time: 1532
Reactor Mode: 1	Mode Name: SHUTDOWN	Power Level: 00%

This report was initiated by Deviation Report D-4-1-91-012

SHUTDOWN Mode (1) - Shutdown - In this position, a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection trip systems have been deenergized for 10 seconds prior to permissive for manual reset.

B. DESCRIPTION OF EVENT:

On January 18, 1991 at 1532 hours, Unit One was in the SHUTDOWN mode at 0 percent of rated core thermal power. At this time, the station was notified by the Boiling Water Reactor Systems Design Department (BWRSDD) that stresses at twelve points on four Atmospheric Containment Atmosphere Dilution (ACAD) [BB] and Containment Atmospheric Monitoring (CAM)[IP] lines exceeded UFSAR allowable stresses. This determination was made while performing analyses for Modification M04-1-88-103. The scope of this modification includes resolution between the "as built" and "as analyzed" configuration discrepancies. The points affected are on line 1-2502B-1" (ACAD piping from the air receiver line to drywell), on line 1-2503B-1" (ACAD piping from air receiver line to torus), on line 1-2401B-1/2" (CAM sample piping from monitor to drywell), and on line 1-2402B-1/2" (CAM sample piping from monitor to torus). Although these lines are attached to the primary containment, all points identified in this event are outside the scope of NUREG 0661 (Safety Evaluation Report MARK I Containment Long Term Program). Subsequent walkdowns and analysis on the Unit 2 ACAD/CAM piping revealed that support modifications would be required to bring this piping within UFSAR/FSAR design margins. These necessary support modifications/additions for Unit 2 will be included as part of modification No. M04-2-88-103 (Partial "B"). There is no component failure associated with these events and further evaluations determined that all lines affected by these events are operable.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73 (a)(2)(ii)(B), which requires the licensee to report any event or condition that resulted in the condition of the nuclear power plant, including its principle safety barriers, being seriously degraded, or that resulted in the nuclear power plant being in an unanalyzed condition that significantly compromised plant safety.

Per Technical Specifications the CAM system is required for post accident hydrogen monitoring. There are no Technical Specification requirements for the ACAD system. The apparent cause of this event is preservice error involving the design and the construction of the affected lines. There are differences between the "as built" and designed configuration which apparently were not reconciled during construction. BWRSDD has been investigating this type of discrepancy under the Small Bore Piping Verification Program. Modification M04-1(2)-88-103 was initiated as a result of walkdowns performed under this program.

D. SAFETY ANALYSIS OF EVENT:

The safety consequences of this event were minimal. The code allowable stresses are conservative and provide for an adequate safety factor by limiting primary membrane and bending stresses. As shown by analyses, both piping systems are operable. When CAM is required by Technical Specifications, one out of two post accident hydrogen monitors has to be operable per Table 3.2-4. There are no Technical Specification requirements for the ACAD system. If the ACAD and CAM system are inoperable alternate methods of post LOCA containment atmospheric dilution and monitoring are available. If both CAM system hydrogen monitors are inoperable continued reactor operation is permissible up to 30 days provided that during this time the HRSS hydrogen monitor capability for the drywell is operable. Instrument air can be used for dilution, instead of the ACAD system per procedure QOP 1600-26, "Post LOCA Drywell Purge With Air for Hydrogen Control". QOP 1600-25, "Post LOCA Drywell Purge With Nitrogen for Hydrogen Control" provides for containment atmospheric dilution with nitrogen.

E. CORRECTIVE ACTIONS:

The immediate corrective action was to determine operability of the affected piping system. Since operability had already been determined by BWRSDD, no further immediate action was necessary. Long term corrective action is to implement Modification M04-1(2)-88-103, Partial "B". This modification reconciles differences between as built and as analyzed small bore ACAD/CAM piping systems. The walkdown for Unit Two ACAD/CAM piping system as well as the Unit One walkdown has been completed and piping stress analyses for these piping systems are in progress.

These partial modifications' design and issue have temporarily been placed "on hold," pending the completion of the Combustible Gas Control (CGC) modification which installs a new NCAD line and holding tank. The CGC mod has been approved by the Station Modification Review Committee (modification request #MR4-1(2)-92-028) and is currently scheduled for implementation during Q1R14 (1995) and Q2R14 (1996). After the completion of testing and Op authorization for the CGC mod, a disposition of the ACAD piping will be determined (either removal of line or modified/new support installation). A supplemental report will then be issued detailing the planned corrective actions (NTS# 2542009101202). The ACAD/CAM stress analysis and supports design will then be completed. The modification will be installed which will result in all affected lines becoming seismically qualified based on the UFSAR/FSAR design margins.

F. PREVIOUS EVENTS:

The station records do not identify any similar events involving UFSAR allowable stresses for the ACAD and CAM piping identified in this event. One event involving "Mark I" CAM piping which did not meet UFSAR allowable code stresses is documented as follows.

LER NUMBERS	TITLE
254/86-022	Containment Atmospheric Monitoring does not meet code allowable limits.

Other similar events are documented as follows:

LER NUMBERS	TITLE
254/86-025	Torus [BO] Piping Supports Exceed Code Stress Allowable Limits.

LER NUMBERS	TITLE
254/87-030	Anticipated Transient Without Scram [JC] Instrument Sensing Lines Inadequately Supported due to Personnel Error and Inadequate Design.
265/87-019	Piping Supports Outside Compliance With Safety Analysis Report due to Design Error.
254/88-004	Reactor Head Vent Line Outside Safety Analysis Criteria for Allowable Stress due to Design Error.
265/88-017	MSIV Air Line Hanger Not Meeting FSAR Requirements.

These events do not indicate any unfavorable trends since none of the piping discrepancies identified are attributable to recent piping installations.

G. COMPONENT FAILURE DATA:

No component failure was involved in this event.