Commonwealth Edison
Quad Cities Nuclear Power Station
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RLB-92-107

May 9, 1992

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

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

Enclosed is Licensee Event Report (LER) 92-012, 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). 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
Station Manager

RLB/TB/plm

Enclosure

cc: J. Schrage
T. Taylor
INPO Records Center
NRC Region III

STMGR 365

JE22

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
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Title (4)

U-2 RHR Support M-1026D-310 Outside FSAR Allowables

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 4	0 9	9 2	9 2	0 1 2	0 0	0 5	0 8	9 2		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)	
Gary E. Knapp Ext. 2153	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
__ 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 April 9, 1992 at 1631 hours, Unit Two was in the SHUTDOWN mode at 0 percent rated core thermal power when the station was notified by the Boiling Water Reactor Engineering Department (BWRE) that the expansion anchors for Unit 2 Residual Heat Removal (RHR) [BO] support M-1026D-310 did not meet Final Safety Analysis Report (FSAR) design allowables. The safety consequences of the expansion anchor's condition are minimal. An operability analysis showed that the supports failure would not affect the RHR line's ability to function. The apparent cause of the event was a reconciliation between the current method of performing the thermal seismic analysis with the original seismic analysis. The immediate corrective action was to issue Modification M04-2-92-009 to permanently remove support M-1026D-310. Previous walkdowns on the Unit One RHR system has revealed no similar problems. No further corrective action is necessary. This event is being reported in accordance with 10CFR50.73(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: U-2 RHR Support M-1026D-310 Outside FSAR Allowables.

A. CONDITIONS PRIOR TO EVENT:

Unit: Two	Event Date: April 09, 1992	Event Time: 1631
Reactor Mode: 1	Mode Name: SHUTDOWN	Power Level: 00%

This report was initiated by Deviation Report D-4-2-92-049.

SHUTDOWN Mode (1) - 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 April 09, 1992 at 1631 hours, Unit Two was in the SHUTDOWN mode at 0 percent rated core thermal power. At this time, the station was notified by Boiling Water Reactor Engineering Department (BWRE) that the expansion anchors for Unit 2 Residual Heat Removal (RHR) [BO] support M-1026D-310 did not meet Final Safety Analysis Report (FSAR) design allowables. The support was in a configuration which resulted in an overstress condition in the expansion anchor plate. Support M-1026D-310 is a Pacific Scientific [PC29) model PSA-10 mechanical snubber [SNB] located on RHR line 2-1012A-16" which is the Unit 2A Loop Low Pressure Coolant Injection line. A telecopy of operability assessment was received by the station on the same day and Modification M04-2-92-009 was initiated on 04/10/92 to permanently remove the support.

C. APPARENT CAUSE OF EVENT:

This event is being reported according to 10CFR50.73(a)(2)(ii)(B), which requires the reporting of 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 apparent cause of the event was a reconciliation between the current method of performing the thermal and seismic analysis with the original seismic analysis. The original seismic analysis for the Reactor Recirculation (RR) [AD] system considered the RR pumps to be six-way seismic anchor points. However, torsional movement was not restrained. Impell Corporation reanalyzed the system seismic and thermal movements and a modification was performed to the system's supports. The new updated seismic and thermal movement data was then sent to Sargent & Lundy Engineers (S&L). S&L applied the new seismic and thermal movement data to line 2-1012A-16" and subsequently identified and subsequently identified the overstress condition in the expansion anchor plate of support M-1026D-310.

D. SAFETY ANALYSIS OF EVENT:

The safety consequences of the expansion anchor's condition is minimal. An operability assessment performed by BWRE showed that failure of the support would not affect the RHR line's ability to perform its function, i.e. the piping and remaining supports would meet FSAR allowables. However, the analysis also identified that the expansion anchors would not meet FSAR allowables and recommended that the support be permanently removed.

E. CORRECTIVE ACTIONS:

The immediate corrective action was to issue Modification M04-2-92-009 to permanently remove support M-1026D-310. This modification was completed prior to startup following the Unit 2 Refueling outage (Q2R11). Since modifications and minor design changes to the piping and supports may affect existing seismic and thermal analysis, it is routine during the implementation of these changes to update these analyses. Previous walkdowns on the Unit One RHR system have revealed no similar problems. No further corrective action is necessary.

F. PREVIOUS EVENTS:

The following are previous Licensee Event Reports (LER) written for similar events:

Docket #/LER #	Title
254/87-008	1C Residual Heat Removal Service Water Pump Piping In Excess Of Allowable Stress Due To Sheared Anchor Bolts.
254/87-011	Residual Heat Removal Support Embedment Plate In Excess Of Allowable Stress Due To Improper Anchor Strap Spacing.
254/87-024	Drywell Structural Steel Connections.
254/87/026	Piping Support Outside FSAR.

254/87-030	ATWS Hangers Do Not Meet Design Specifications.
254/888-004	Rx Head Vent Line Outside Of Safety Analysis Criteria For Allowable Stress Due To Design Error.
254/88-017	MSIV Air Line Hanger Does Not Meet FSAR Requirements.
254/90-022	Piping System Outside FSAR Compliance Caused By Computer User Input Error.
254/90-030	Rx Recirc Piping Outside Seismic Design Basis Due To A Design Discrepancy.
254/91-003	Specific Points In ACAD/CAM Lines Exceed UFSAR Allowable Stress Due To Preservice Error Involving The Design And Construction Of The Affected Lines.
254/91-016	Support Embedment Plates Outside Of Design Basis Due To A Preservice Error Involving Contractors & Engineering Personnel.
254/92-003	HPCI Suction Pipe Hanger Outside Of Design Basis Due To Unknown Cause.
254/87-019	RHR & HPCI Lines Outside Of FSAR Compliance Due To Design Error.
265/88-006	11 Flued Head Anchors Do Not Meet FSAR Design Requirements Due To An Analysis Deficiency.
265/88-010	Primary Cont. Structural Steel Connections Outside The FSAR Design Criteria Due To An Original Construction Oversight.
265/88-012	Piping Supports For Line 2-1265-2" Do Not Meet FSAR Design Requirements Due To An Improper Analysis During A Mod.
265/88-017	Air Supply To MSIV 2-203-1C Failed to Be Within Design Basis Of The FSAR Due To An Improperly Selected Hanger.
265/90-015	HPCI Drain Pot Lines Outside Of FSAR Due To An Inadequate Original Seismic Evaluation.
265/91-014	2A RHR Heat Exchanger Support Have Stresses Which Are Outside Of Design Bases Due To Notches On Beams Flanges From Unknown Causes.

G. COMPONENT FAILURE DATA:

There was no component failure identified in this event.