LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
QUAD-CITIES, NUCLEAR POWER STATION, UNIT One	0 5 0 0 0 2 5 4	1 of 0 5

Title (4) Torus Attached Small Bore Piping Does Not Meet Code Allowable Limits Due To Design Error

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 8	2 7	8 6	8 6	0 2 5	0 0	0 9	1 9	8 6	Quad Cities Unit Two	0 5 0 0 0 2 6 5
										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)	__ 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	
Bob Castro, Technical Staff Engineer X 2166	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
X Yes (If yes, complete EXPECTED SUBMISSION DATE)	__ NO		1 2	3 1	8 7

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

During a re-analysis of the IEB 79-141 Mark I program, it was discovered that certain small bore torus attached piping (four inches or less) did not meet FSAR requirements to meet code allowable stress limits for seismic and Mark I loading conditions. Piping on both Units One and Two is affected. The cause of this occurrence is attributed to inadequate design review. The AE firm only qualified added or modified supports on torus attached piping during the initial design review. Supports that did not require modification were not verified to be qualified and were subsequently found to exceed code stress allowables. An operability assessment has determined that the systems in question are functional and that the safety of the plants has not been jeopardized. Approximately 20 to 30 hangers per unit will require modification. Design of the modified supports is in progress. This report is submitted in accordance with the requirements of 10 CFR 50.73(a)(2)(ii).

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PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2511 MWt rated core thermal power. Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

IDENTIFICATION OF OCCURRENCE:

Certain lines (small bore less then 4") attached to the torus do not meet FSAR Code Stress allowable for Mark I Containment due to certain supports not being qualified.

Discovery Date: 8/27/86

Report Date: 9/19/86

This report was initiated by Deviation Report D-4-1-86-86

CONDITIONS PRIOR TO OCCURRENCE:

RUN Mode(4) - Rx Power 95% - Unit Load 790 MWe

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).

DESCRIPTION OF OCCURRENCE:

On August 27, 1986, Unit One and Two were operating at 95 and 97 percent of rated core thermal power when Quad Cities Station was notified by the Station Nuclear Engineering Department of an operability assessment concerning various torus attached small bore piping (four inches or less) on both units. The assessment was performed to address the small bore piping supports which were not qualified by detailed calculation during the Mark I/IEB 79-14 programs.

The Mark I program was intended to re-evaluate and subsequently qualify various containment piping systems for design basis accident hydrodynamic load conditions. For the small bore piping, an initial evaluation was undertaken to determine if each piping configuration was acceptable from a Mark I code stress allowable standpoint. For the piping systems' supports, the following simplifying assumption was made: If the pipe stresses were found to be less than allowable (Mark I criteria), the supports were assumed to be adequate without qualification. The hangers that the assumption applied to had been field verified during various system walkdowns. For the second alternative, if the piping configuration was stressed beyond allowable limits (Mark I criteria), supports were added, modified, or removed to bring piping stresses within Mark I criteria limits. Subsequently, only added or modified supports were qualified to Mark I code stress allowables. The Architect Engineer associated with the Mark I/IEB 79-14 programs has recently stated that the assumption concerning supports within a piping system that qualified to code stress allowables upon the initial evaluation may not be valid. Hence, any support not added or modified during the program may be unable to perform its desired function since it was not qualified. Therefore, greater stresses could be induced into the piping than originally designed for.

The Station Nuclear Engineering Department had anticipated that all existing supports would be qualified by detailed calculation regardless of the as-found initial evaluation for the Mark I program. When the discrepancy was discovered (during other system field walkdowns) the assessment on the affected piping systems was undertaken to justify continued operation of both units until discrepant supports could be modified and qualified by calculation.

The following piping systems contain supports not qualified by calculation:

<u>Unit One</u>

1-1079-3"-DX   RHR 1C & 1D Pump Discharge [BO]
1-1032-3"-DX   RHR 1A & 1B Pump Discharge [BO]
1-2363-2"-L   HPCI Turbine Exhaust Drain Line [BJ]
1-4712-1"-LX   Vacuum Relief [LD]
1-2308-1"-LX   Vent Line for PSH 1-2355 [BJ]
1-8803-1/2"-H   Air Sample [NH]
Instrument line for PT 1-2366 [BJ]
Instrument line for 1-2368A [BJ]
Instrument line for 1-2366B [BJ]

<u>Unit Two</u>

2-2340-4"-DR   HPCI Pump Discharge [BJ]
2-1079-3"-DX   RHR 2C & 2D Pump Discharge [BO]
2-1032-3"-DR   RHR 2A & 2B Pump Discharge [BO]
2-2309-2"-LX   HPCI Drain Pot Drain [BJ]
2-1303-2"-DX   RCIC Min. Flow [BN]
2-8702-1"-LX   Vacuum Relief [NH]
2-8803W-1/2"-H   Air Sample [NH]
2-1634-1/2"-LA   Level Transmitter [NH]

The operability assessment performed by the Architect Engineer is a detailed step by step analysis which examines piping and supports for compliance by a progressively detailed evaluation. Thus, the systems with the highest design margins are eliminated from consideration first, with increasing levels of analysis being applied to the systems with lower design margins.

To accomplish the operability assessment, a representative sample of the affected piping was chosen based on the availability of information and the knowledge that higher design margins generally exist in the smaller piping supports. Only the three (3) and four (4) inch lines on both units were included in the analysis for a total of eleven (11) lines.

As a result of the assessment, the supports on four (4) of the eleven (11) lines sampled passed the operability criteria upon an initial screening detailed in the assessment procedure. Of the remaining seven lines, the line with the most severe apparent overstress condition was selected, and the operability procedure was continued. Essentially this line was shown to pass the operability criteria with all unqualified supports removed from the analysis. The acceptance criteria used for the analysis is such that the total combined piping stresses due to dead weight, pressure, Mark I loading, and a seismic event are less than two times the yield stress of pipe material. FSAR damping values (0.5 percent) were included for the analysis as well. The operability assessment has determined that all lines in question are operable.

Hence, it has been determined that the systems in question are functional and that the safety of the plant has not been jeopardized.

This report is submitted to you in accordance with the requirements of 10 CFR 50.73(a)(2)(ii), which requires the reporting of 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.

APPARENT CAUSE OF OCCURRENCE:

The cause of this event can be attributed to inadequate design/modification review for the Mark I/IEB 79-14 program implementations. The Station Nuclear Engineerng Department had assumed that all piping supports attached to Mark I affected piping would be qualified by calculation. The Architect Engineer involved has stated that an analysis assumption resulting in calculated qualification of only added or modified supports was inadequate for the Mark I/IEB 79-14 program.

ANALYSIS OF OCCURRENCE:

Some lines in question form primary containment boundaries. If a design basis accident were to occur in conjunction with a seismic event, the concern is that the lines could achieve an overstressed condition and subsequently rupture. If the rupture were to occur at a non-isolatable point, primary containment could be jeopardized. Various minimum flow lines from Emergency Core Cooling Systems are also mentioned as affected piping. The Unit One and Two Residual Heat Removal test return lines, and the Unit Two High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems test return lines are included. Depending on where a postulated line break were to occur in relation to isolation locations, these systems' design flow to the reactor vessel could be impaired. The operability assessment performed to evaluate the functionality of the affected lines provides assurance that an overstress condition will not occur.

CORRECTIVE ACTION:

At the present time the Station Nuclear Engineering Department is in the process of developing a cost estimate and a schedule for performing modification work where necessary to affected piping supports. The modifications will result in qualified piping supports to Mark I Code Stress allowables. It appears that there will be approximately twenty (20) to thirty (30) modified hangers per unit. A preliminary date for having all drawings issued for modifications is February 1, 1987. A supplemental report is expected to be submitted in December 1987 which will detail repairs accomplished and describe the status of the affected lines.

FAILURE DATA:

A similar occurrence affecting the Containment Atmospheric Monitoring System piping is documented in Deviation Report 4-1-86-51.

Commonwealth Edison
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RLB-86-175

September 22, 1986

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 please find Licensee Event Report (LER) 86-025, Revision 00, for Quad-Cities Nuclear Power Station.

This report is submitted to you in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(ii), which requires the reporting of any event or condition that resulted in a nuclear power plant being in an unanalyzed condtion.

Respectfully,

COMMONWEALTH EDISON CO
QUAD-CITIES NUCLEAR POWER STATION

R L Bax

R. L. Bax
Station Manager

RLB/MSK/dak

Enclosure

cc: J. Wojnarowski
A. Morrongiello
INPO Records Center
NRC Region III

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