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Quad Cities Nuclear Power Station
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DcD
JE23A

RLB-92-099

June 4, 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) 88-006, Revision 02, for Quad-Cities Nuclear Power Station. This revision provides additional information regarding flued head anchors.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(ii)(B), which requires the reporting of any event or condition that resulted in the nuclear power 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

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
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Title (4) Exceeding Technical Specification Leakage Limit for Containment Isolation Valves and Penetrations Due to Excessive Leakage from the HPCI Steam Exhaust Check Valve And Other Isolation Valves.

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 2	0 5	9 0	9 0	0 0 3	0 1			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)	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 David Kunzmann, Technical Staff Engineer, Ext. 2162

TELEPHONE NUMBER: AREA CODE 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
X	B J	I S V	M 3 6 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

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

Expected Submission Date (15): Month 0 6 Day 3 1 Year 9 0

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

ABSTRACT:

On February 4, 1990, Quad Cities Unit Two was shutdown for the end of cycle 10 refueling and maintenance outage. On February 5, 1990, at 1730 hours, while local leak rate testing (LLRT) the High Pressure Coolant Injection (HPCI) system steam exhaust check valve, it was determined that the measured leakage rate of 528.6 standard cubic feet per hour (SCFH) exceeded the Technical Specification 3.7.A.2.a.2 limit of 293.75 SCFH (0.60La) for all valves and penetrations excluding the main steam isolation valves (MSIV).

The apparent cause for exceeding the Technical Specification 0.6 La limit was due to the excessive leakage of the HPCI steam exhaust check valve. Corrective action included replacing the valve and modifying the sparger to prevent seat erosion. This report is being submitted to comply with 10CFR50.73(a)(2)(i)(B).

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].

EVENT IDENTIFICATION:

Eleven Unit Two flued head anchors do not meet the design requirements due to analysis deficiency.

A. CONDITIONS PRIOR TO EVENT:

Unit: Two	Event Date: April 4, 1988	Event Time: 1410
Reactor Mode: 4	Mode Name: RUN	Power Level: 93%

This report was initiated by Deviation Report D-4-2-88-017

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 April 4, 1988, Quad-Cities Unit Two was in the RUN mode at 93 per cent of rated core thermal power. At 1410 hours, the Station was notified by the Boiling Water Reactor Engineering Department (BWRED) that eleven (11) flued head anchors [SPT] did not meet the design requirements specified in the Quad-Cities Final Safety Analysis Report (FSAR). All of the Unit Two flued head anchors were reviewed and the ones in question are located at the following penetrations:

Penetration Number	Associated System
1. X-11	High Pressure Coolant Injection [BJ] Steam Supply
2. X-13A	Residual Heat Removal (RHR)/Low Pressure Coolant Injection (LPCI) injection [BO]
3. X-13B	RHR/LPCI Injection
4. X-16A	Core Spray [BM] Injection
5. X-16B	Core Spray Injection
6. X-23	Reactor Building Closed Cooling Water (RBCCW) [CC] Supply
7. X-24	RBCCW Return
8. X-36	Control Rod Drive (CRD) [AA] Return
9. X-47	Standby Liquid Control [BR] Injection
10. X-8	Main Steamline Drains [SB]
* 11. X-7A, B, C, D	Main Steam
X-9A, B	Feedwater [SJ]
X-10	Reactor Core Isolation Cooling [BN] Steam Supply
X-12	Shutdown Cooling Suction [BO]
X-17	RHR Head Spray [BO]

*Gang Anchor

The NRC, via the Emergency Notification System (ENS), was notified of this condition at 1423 hours, to satisfy the requirements of 10 CFR 50.72. The design concern for the eleven (11) flued head assemblies was the result of a concern identified at Dresden Station. It was identified during the Dresden review that the flued head anchor structures at Dresden and Quad-Cities were not included under the I.E. Bulletins No. 79-14 and 79-02 scope of work.

The flued head anchors in question were assessed for consideration of continued operability. The results of the assessment concluded that the flued head anchors will perform their intended functions, thereby, establishing an acceptable operability basis.

Subsequent to the initial submittal of this LER, it was discovered that one shear pin was missing from the Flued Head Anchor structure at penetration X-16A. This specific concern was assessed by Boiling Water Reactor Engineering Department and the anchor was considered operable without the pin. It was recommended, however, to install a pin in order to fulfill the FSAR requirements.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION								
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		Year	///	Sequential Number	///	Revision Number		
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TEXT								

C. APPARENT CAUSE OF EVENT:

This report is submitted to comply with the requirements of 10 CFR 50.73(a)(2)(ii)(B), which requires the reporting of any event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant.

The exclusion of the structures in the 79-02 and 79-14 programs was due to misinterpretation of the scope requirements. Therefore, analysis for these structures was not reassessed for design base requirements. Although the eleven (11) flued head anchors did not meet FSAR design requirements, these flued head anchors were considered operable as determined by an analysis in January, 1988.

The cause of the missing pin for the structure at penetration X-16A could not be determined.

D. SAFETY ANALYSIS OF EVENT:

The health and safety of the public and of plant personnel was not adversely affected by this event. Since the anchor assemblies were analyzed and considered operable, the associated safety significance is minimal.

The concerns identified are to be resolved to comply with the necessary FSAR requirements.

E. CORRECTIVE ACTIONS:

When the flued head anchor assembly concerns were initiated, Commonwealth Edison (CECo) reviewed the basis for the exclusion of the assemblies under the IE Bulletin No. 79-14 and 79-02 scope of work. Subsequently, CECo presented justification for the operability criteria as well as the basis for the exclusion in relation to IEB 79-14 and 79-02. The NRC has disagreed with this exclusion basis, hence, CECo has initiated a comprehensive program to demonstrate the adequacy of the aforementioned flued head anchor structures. An engineering walkdown was conducted during the ongoing Quad-Cities Unit Two refueling outage. The eleven (11) Unit Two structures were reviewed and Engineering Change Notices were issued. Modification (M-4-2-88-017) has been initiated to revise the structures to a condition that complies with FSAR design requirements. The eleven (11) flued head anchors are complete and the anchors meet the required FSAR design requirements. In general, the modification involved the addition of structure baseplates and anchors, the welding of some minor support additions, and changeout of certain concrete expansion anchors (Nuclear Tracking System 2652008801701).

In regard to the missing pin, a new pin was installed and a keeper tab was welded to hold the pin in place. This should prevent recurrence.

Further analysis of Unit One flued head anchor assemblies showed that assemblies located at the X-11, X-13A, X-13B, X-16A, X-16B, X-23, X-24, X-36, X-47, X-7A, X-7B, X-7C, X-7D, X-8, X-9A, X-9B, X-10, X-12 and X-17 penetrations did not meet FSAR design requirements. The station implemented modification M-4-1-88-017 to resolve any FSAR design requirement deficiencies. Modification M-4-1-88-017 was completed during the Q1R10 refueling outage and authorized for operation on 11-22-89. Under this modification piping to the X-36 penetration was cut and capped at both ends. This reduced the pipe load to zero and eliminated the need to modify the anchor.

F. PREVIOUS EVENTS:

LER NUMBER	TITLE
254/86-022	Containment Atmospheric Monitoring [IL] Line does not meet code allowable stress limits.
254/86-024	U-1 and U-2 Residual Heat Removal Service Water [BO] Piping Supports exceeded code stress allowable limits.
254/86-025	Torus attached Small Bore Piping does not meet Allowable Limits
254/87-008	1C Residual Heat Removal Service Water Pump [P] 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.
254/87-026	Piping Supports Outside Compliance with Safety Analysis Report due to Design/Construction Error.
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.

TEXT

G. COMPONENT FAILURE DATA:

Since the structures are not considered [illegible]operable, no component failure is identified in this event.