NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Quad-Cities Nuclear Power Station, Unit 1	0 5 0 0 0 2 5 4	1 OF 0 3

TITLE (4)

Linear Indications on Reactor Recirculation System Welds

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 4	1 4	8 4	8 4	0 0 5	0 1	0 8	2 3	8 4	NA	0 5 0 0 0
									NA	0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (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, NRC Form 366A)
[ ] 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)
Hien Q. Do	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
X	A D	P S X	D 2 4 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On March 6, 1984, Quad-Cities Unit One was shutdown in order to begin the Cycle 7 Refueling Outage. On April 13, Induction Heat Stress Improvement (IHSI) procedures were performed on the Reactor Recirculation System welds and on some Residual Heat Removal (RHR) System welds as part of the Inservice Inspection Program for large bore stainless steel piping. Following the IHSI, ultrasonic inspections revealed several welds with circumferential and/or axial linear indications. Visual inspections also detected several weld areas with small cracks. This report provides the summation of those findings, the final evaluation, and the corrective actions taken as a result of these indications.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Quad-Cities Nuclear Power Station, Unit 1	0 5 0 0 0 2 5 4	8 4	— 0 0 5	— 0 1	0 2 OF 0 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

Event Description

On March 6, 1984, Quad-Cities Unit One was shutdown to begin the Cycle 7 Refueling Outage. On April 13, Induction Heat Stress Improvement (IHSI) procedures were initiated on 88 welds in the Reactor Recirculation System (AD) and on a portion of the Residual Heat Removal System (BO). Following IHSI, the welds were visually and ultrasonically tested by Lambert, McGill and Thomas, Incorporated and/or by Universal Testing Laboratories, Incorporated, personnel and the results were reviewed by the Commonwealth Edison Level III examiner. The results indicated that 18 welds in the Reactor Recirculation System were identified as having linear indications in the heat affected zone of the welds.

Visual examinations performed on the 18 welds revealed 10 welds with water seeping from small pinholes or from small axial cracks.

A complete list of affected welds including a description of the indications and the final disposition can be found in Attachment 1.

Probable Consequences of the Occurrence

Because of the inherent high material toughness of austenitic stainless steel piping, intergranular stress corrosion cracking is unlikely to cause a rapidly propagating failure. If this did occur, a 100 percent through-wall crack could be easily detected using existing Primary Containment leakage monitoring systems and therefore, the safety consequences of this event were minimal. During the current operating cycle, the allowable containment leakage rate has been reduced in order to expedite the investigation of potential leakage from stainless steel piping. No leakage was detected prior to the current refueling outage.

Cause

The cause of the crack indications is postulated to be Intergranular Stress Corrosion Cracking (IGSCC). The materials used in the original Recirculation piping and fittings are regular Grade Type 304 stainless steel which is known to be susceptible to stress corrosion cracking based on plant operating history.

The normal heat generated by the welding of Type 304 stainless steel causes a sensitized heat affected zone at the weld-to-pipe interface. This, combined with coolant impurities, tensile stresses experienced in the weld area, and high operating temperatures are factors encountered in the Reactor Recirculation System. These factors are the mechanism necessary for IGSCC to occur.

Cause (continued)

All welds with water seeping from a pinhole or axial crack, with the exception of weld 02J-F6, were found after IHSI had been performed on them. The reversing of stresses within the weld during IHSI combined with an axial crack of greater than 50 percent through-wall depth caused the crack to propagate 100 percent through-wall. The result was a small pinhole and/or a short axial crack large enough to allow water to seep through. Weld 02J-F6 was found to be leaking after surface preparation, prior tq IHSI.

Corrective Action

The crack indication evaluation and repair criteria was performed by NUTECH Engineers, Incorporated. Indications were evaluated based on indication depth, length, direction, and applied stresses. Induction Heat Stress improvement was performed on many welds in order to bring about a compressive residual stress pattern which prevents the initiation of IGSCC and retards the propagation of small existing IGSCC flaws.

The repair program consisted of either performing a weld overlay, or leaving the weld as-is or removing the flaws and then performing weld repair. All welds containing indications that were left as-is had IHSI performed on them.

Circumferential indications detected ultrasonically by the regular shear wave technique required additional testing using a newer ultrasonic testing technique, known as ID creeping wave. This was necessary to determine the actual size of the indications.

As a general rule, circumferential indications with a length greater than 120 degrees of the pipe circumference and/or a depth greater than 25 percent of the pipe wall thickness were repaired by applying a weld overlay. All axial indications and welds seeping water required either a weld overlay or flaw removal. All analyses were performed to the guidelines specified in the ASME Boiler and Pressure Vessel Code, Section XI, Paragraph IWB-3640, "Acceptance Criteria for Austenitic Steel Piping".

Sixteen welds were repaired using weld overlay. The length and thickness of each overlay differed, depending upon the flaw size, analyzed stresses, and pipe geometry. One weld had flaws completely removed and the weld repaired. Another weld with small circumferential indications was left as is.

Prior to the Reactor startup, the entire Recirculation System was hydrostatically tested in conjunction with the Reactor Vessel Hydrostatic Test at 1.1 times the system nominal operating pressure.

QUAD CITIES - UNIT ONE
Attachment 1 - Indication Description and Repair

WELD I.D.	LOCATION	PIPE DIAM.	CRACK TYPE	FLAW CHARACTERIZATION MAX DEPTH	LENGTH	LOCATION	DISPOSITION	WELD IHSI'd	WELD LEAKS
02C-S4	'C' Riser Pipe to Elbow	12"	Circ	44%	4"	Pipe side	Overlay	Yes	No
			3 Axials	51%	1" Max	Pipe side	Overlay	Yes	
02D-S4	'D' Riser Pipe to Elbow	12"	2 Axials	77%	7/8" Max	Pipe side	Overlay	No	No
02E-S4	'E' Riser Pipe to Elbow	12"	Circ	65%	4/5"	Pipe side	Overlay	Yes	Yes
			8 Axials	100%	1 1/8" Max	Pipe side	Overlay	Yes	
02F-S4	'F' Riser Pipe to Elbow	12"	3 Axials	25%	4/5" Max	Pipe side	Overlay	No	No
02G-S3	'G' Riser Elbow to Pipe	12"	Circ	50%	3/4"	Pipe side	Overlay	Yes	Yes
			7 Axials	100%	1 1/8" Max	Pipe side	Overlay	Yes	
			1 Axial	100%	7/8" Max	Elbow side	Overlay		
02G-S4	'G' Riser to Elbow	12"	Circ	18%	1"	Elbow side	Overlay	No	No
			1 Axial	18%	1/8" Max	Elbow side	Overlay		
02H-S3	'H'Riser to Pipe	12"	Circ.	21%	3"	Pipe side	Overlay	Yes	Yes
			3 Axial	100%	3/4" Max	Pipe side	Overlay		
02H-S4	'H' Rise Pipe to Elbow	12"	2 Axials	100%	3/4" Max	Pipe side	Overlay	Yes	Yes
			2 Axials	26%	1/2" Max	Elbow side	Overlay		
02J-S3	'J' Riser Elbow to Pipe	12"	Circ	12%	3/4"	Pipeside	Overlay	Yes	Yes
			1 Axial	100%	1/2"	Pipeside	Overlay		
02J-S4	'J' Riser Pipe to Elbow	12"	6 Circ(s)	55%	13 1/4" Total	Pipeside	Overlay	Yes	Yes
			9 Axials	100%	1 1/10" Max	Pipe side	Overlay		

QUAD CITIES - UNIT ONE

ATTACHMENT 1 - Indication Description and Repair (continued)

WELD I.D.	LOCATION	PIPE DIAM.	CRACK TYPE	FLAW CHARACTERIZATION MAX DEPTH	LENGTH	LOCATION	DISPOSITION	WELD IHSI'd	WELD LEAKS
02J-F6	'J' Riser Sweepolet to pipe	12"	7 Axials	100%	1 1/4" Max	Pipe side	Overlay	No	Yes
02K-S3	'K' Riser Elbow to Pipe	12"	4 Circs 5 Axials	25% 100%	10 2/5" Total 5/8" Max	Pipe side Pipe side	Overlay Overlay	Yes	Yes
02K-S4	'K' Riser Pipe to Elbow	12"	2 Axials	100%	1/4" Max	Pipe side	Overlay	Yes	Yes
02M-S3	'M' Riser Elbow to Pipe	12"	3 Axials	64%	1" Max	Pipe side	Overlay	Yes	No
02B-S7	Ring Header Cross to Pipe	22"	1 Axial	100%	Can't be determined	Pipe side	Overlay	Yes	Yes
02B-S10	Ring Header Pipe to Cap	22"	Circ 3 Axials	10% 100%	2" 1/2 " Max	Cap Side Cap Side	Overlay Overlay	Yes	No
02BS-S9	'B' Loop Suction Pipe to Elbow	28"	2 Circ(s)	18%	2" Total	Pipe side	Leave as is	Yes	No
02BS-S12	'B' Loop Suction Elbow to Pipe	28"	Circ 2 Axials	18% 18%	4/5" 3/8" Max	Pipe side Pipe side	Flaws removed and weld repaired	Yes	No

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NJK-84-248

August 23, 1984

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) 84-005, Revision 1, for Quad-Cities Nuclear Power Station.

This supplemental report is submitted to you in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73-(a)(2)(ii), to inform you of the corrective actions performed on cracks found during the Quad-Cities Unit One Primary Coolant Circuit NDE UT Inspections.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

N. J. Kalivianakis
Station Superintendent

NJK:DBC/bb

Enclosure

cc B. Rybak
A. Morrongiello
INPO Records Center
NRC Region III

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