NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: ① (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

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LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

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0 1 REPORT SOURCE L ⑥ 0 5 0 0 0 2 6 5 ⑦ 1 0 2 8 8 3 ⑧ 0 2 2 8 8 4 ⑨
DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES ⑩

0 2 During the Quad-Cities Unit Two Refuel Outage, ultrasonic examinations of large bore
0 3 stainless steel pipe welds performed to comply with the requirements of the NRC
0 4 IGSCC Inspection Order identified 11 welds as having crack indications. All of the
0 5 welds were located on the Reactor Recirculation System piping. The slow growth rate,
0 6 typical with this type of indications, combined with the reduced allowable
0 7 containment leakage rate would have been sufficient to readily identify any possible
0 8 leakage and preclude a gross failure.

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP SUBCODE	VALVE SUBCODE
0 9	C B ⑪	E ⑫	C ⑬	P I P E X X ⑭	E ⑮	Z ⑯

⑰ LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
8 3	—	0 2 1	/	0 1	T	—	5

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS ㉒	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP SUPPLIER	COMPONENT MANUFACTURER
B ⑱	Z ⑲	Z ⑳	Z ㉑	0 0 0 0	Y ㉓	N ㉔	N ㉕	D 2 4 0 ㉖

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS ㉗

1 0 The cause of this occurrence is postulated as being intergranular stress corrosion
1 1 cracking. The weld repair program involved either performing a weld overlay or
1 2 leaving the weld in its present condition. Induction Heat Stress Improvement (IHSI)
1 3 was also employed in order to arrest further crack growth. Additional inspection
1 4 techniques were used on several welds, and no crack indications could be identified using these advanced techniques.

	FACILITY STATUS	% POWER	OTHER STATUS ㉚	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION ㉜
1 5	H ㉘	0 0 0 ㉙	NA	C ㉛	IGSCC Inspection Order

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY ㉟	LOCATION OF RELEASE ㊱
1 6	Z ㉝	Z ㉞	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION ㊴
1 7	0 0 0 ㊲	Z ㊳	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION ㊶
1 8	0 0 0 ㊵	NA

8403200106 840228
PDR ADOCK 05000265
S PDR

LOSS OF OR DAMAGE TO FACILITY ㊸

	TYPE	DESCRIPTION
1 9	Z ㊷	NA

PUBLICITY

	ISSUED	DESCRIPTION ㊺	NRC USE ONLY
2 0	Y ㊹	Numerous Local Media Reports	

NAME OF PREPARER D Clark PHONE: 309-654-2241, ext 244

IE22

GPO 917-926

I. LER NUMBER: LER/RO 83-21/01T-5

II. LICENSEE NAME: Commonwealth Edison Company
Quad Cities Nuclear Power Station

III. FACILITY NAME: Unit T.o

IV. DOCKET NUMBER: 050-265

V. EVENT DESCRIPTION:

On September 4, 1983, Quad Cities Unit Two was shut down in order to begin the Cycle 6 refueling outage. During the outage, numerous inspections of the Primary Coolant System were performed as required by the N.R.C. IGSCC Inspection Order. The ultrasonic testing was performed by Lambert, McGill and Thomas, Incorporated personnel and the results were reviewed by the Commonwealth Edison Level III examiner. The results indicated that eleven welds on large bore stainless steel pipe were identified to contain linear indications in the heat-affected zone. All of the welds were located on the Reactor Recirculation System. A complete list of affected welds, including a description of the indications and the final disposition, can be found on Attachment 1.

An additional eleven welds in the Reactor Recirculation System and the Residual Heat Removal (RHR) Shutdown Cooling Suction piping were also found to contain linear indications. These welds were inspected as part of the N.R.C. I.E. Bulletin 83-02 and are reported under LER/RO 83-20/01T.

VI. PROBABLE CONSEQUENCES OF THE OCCURRENCE

The prob ble consequences of this occurrence were minimal. Crack indications his type tend to propagate at a slow rate. Therefore, a 100 percen ugh-wall crack could be easily detected using existing Primary Cont leakage monitoring systems before a complete failure would occur. the Operating Cycle, the allowable containment leakage rate h n reduced in order to expedite the investigation of potential leakage stainless steel piping. None of the indications discovered extended completely through the weld. Safe operation of the Reactor was not jeopardized as a result of this occurrence.

VII. CAUSE

The exact cause of the crack indications has not been determined; but it is postulated that intergranular stress corrosion cracking is the probable mode of failure. The normal heat generated by welding causes a heat-affected zone at the weld to piping interface. This, combined with coolant impurities, high operating temperatures, and stresses experienced in the weld area, are factors encountered in the Reactor Recirculation System which are mechanisms necessary for intergranular stress corrosion cracking to occur.

The stainless steel piping was fabricated by the Dravo Corporation, Type A358, Grade TP 304. The pipe fittings are Type A 403 Grade WP 304. The stainless steel used in all the original Recirculation System pipe and fittings contained carbon contents between 0.05 and 0.08 percent.

VIII. CORRECTIVE ACTION

The crack indication evaluation and repair criteria determination was performed by NUTECH Engineers, Inc. Indications were evaluated based upon indication depth, length, direction, and applied stresses. Induction Heat Stress Improvement (IHSI) was performed on many welds, both with and without crack indications in order to reduce weld residual stress. As a general rule, circumferential indications with a length greater than 120 degrees of the pipe circumference and/or a depth of greater than 25% of the pipe wall thickness were repaired by applying a weld overlay. All axial indications were repaired by weld overlay. 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." See Attachment 2 for a typical evaluation sequence.

Additional inspections were performed on welds 02F-F6, 02J-F6, 02AS-F14, 02BS-F7, and 02BS-S12 (ref. LER/RO 83-20/01T) to determine the actual size of the indications. Second and third party inspections by Independent Testing Laboratory and Universal Testing Lab. using the Shear Wave technique confirmed the original evaluation. A newer ultrasonic test technique, identified as ID Creeping Wave, was also used by UTL. The results of this inspection, contrary to the other inspection results, indicated that no cracks existed in these welds.

A sample plug was then cut out of weld 02BS-S12 to perform further analyses. The plug was examined visually and by dye penetrant testing but no evidence of a crack could be found. The plug was then sectioned, polished, etched and examined microscopically. No cracks were found on the sample. The remainder of the weld was examined by radiography and the pipe I.D. was visually inspected with a borescope. Results of the sample plug and weld inspections provided conclusive evidence that no IGSCC cracks existed in this weld.

The repair program consisted of either performing a weld overlay or leaving the weld as-is. All welds containing indications that were left as-is had IHSI performed on them. Four welds were repaired using weld overlay. The length and thickness of each overlay differed, depending upon the indication size, analysed stresses and pipe geometry. A more detailed description of the indication evaluation and repair program can be found in a Commonwealth Edison letter from B. Rybak to Mr. Harold R. Denton, "Quad Cities Station Unit 2 Weld Inspection Results, NRC Docket No. 50-265", dated January 27, 1984.

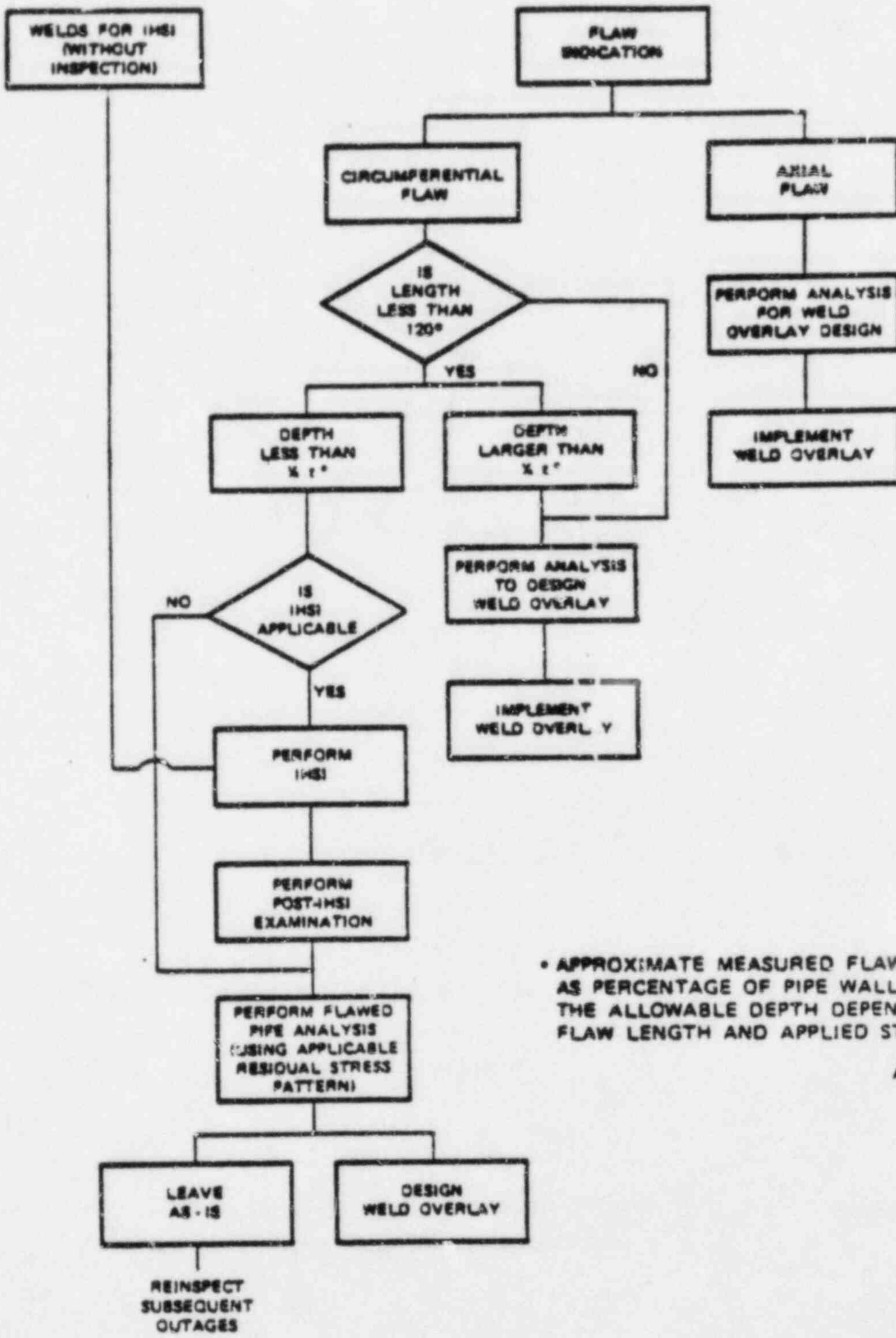
Each weld overlay was dye penetrant tested; and an ultrasonic examination was performed to verify bonding between the base metal and weld material. A post-examination of each weld treated by IHSI was performed by ultrasonic testing. Prior to the reactor startup, the

entire Recirculation System was hydrostatically tested in conjunction with the reactor vessel hydrostatic test at 1.1 times the system nominal operating pressure.

LER/RO 83-21/01T
Quad-Cities Station
Indication Description & Repair

Weld ID	Location	Pipe Diameter	Crack Type	Flaw Characterization(1) Max Depth	Length	Location	Disposition	Weld IHSI'd
02F-F6	F Recirc Riser Pipe to Pipe	12"	Circ (1)	15%	360° Int	Upstream Pipe	Overlay	Yes
02J-F6	J Recirc Riser Saddle to Pipe	12"	Circ (1)	15%	15.0"	Pipe Side	Overlay	Yes
02M-S3	M Recirc Riser Elbow to Pipe	12"	Circ	13%	1.0"	Pipe Side	Leave As-Is	Yes
				30%	1.5"	Elbow Side		
02M-S4	M Recirc Riser Pipe to Elbow	12"	Circ	9%	0.5"	Elbow Side	Leave As-Is	Yes
02AS-S4	A Loop Suction Elbow to Pipe	28"	Circ	13%	21.0" (total)	Elbow Side	Leave As-Is	Yes
				11%	4.0"	Pipe Side		
02AS-S6	A Loop Suction Pipe to Pipe	28"	Circ	21%	7.0"	Upstream Pipe	Leave As-Is	Yes
02AS-S9	A Loop Suction Valve to Elbow	28"	Axial	10%	0.5"	Elbow Side	Overlay	Yes
			Circ	22%	24.0" (total)	Elbow Side		
02AS-S12	A Loop Suction Elbow to Pipe	28"	Circ	9%	3.0" (total)	Elbow Side	Leave As-Is	Yes
			Circ	14%	8.0"	Pipe Side		
02AS-F14	A Loop Suction Pipe to Elbow	28"	Circ (1)(2)	30%	43.0"	Pipe Side	Leave As-Is	Yes
02BS-F7	B Loop Suction Valve to Pipe	28"	Circ (1)(2)	16%	360° Int	Pipe Side	Overlay	Yes
02A-S10	Ring Header Pipe to Cap	22"	Circ	26%	4.0"	Cap Side	Leave As-Is	Yes

NOTE: (1) Flaw characterization based on composite of LMT/ITL results.
(2) Additional inspection by UTL and plug sample of 02BS-S12 confirms that no flaw actually exists.



* APPROXIMATE MEASURED FLAW DEPTH AS PERCENTAGE OF PIPE WALL THICKNESS. THE ALLOWABLE DEPTH DEPENDS ON THE FLAW LENGTH AND APPLIED STRESS.

ACOM84.01

TYPICAL FLAW DISPOSITION SEQUENCE

DmB

Commonwealth Edison
Quad Cities Nuclear Power Station
22710 206 Avenue North
Cordova, Illinois 61242
Telephone 309/654-2241

NJK-84-68

February 28, 1984

J. Keppler, Regional Administrator
Office of Inspection and Enforcement
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Reference: Quad-Cities Nuclear Power Station
Docket Number 50-265, DPR-30, Unit Two
Appendix A, Section 6.6.B.1.c

Enclosed please find Reportable Occurrence Number (RO) 83-21/01T-5 for Quad-Cities Nuclear Power Station. Previous revisions to this Reportable Occurrence have identified welds containing linear indications found during the Inservice Inspection required by the inspection order of all large bore stainless steel piping. This revision identifies the final disposition of these indications.

This report is submitted to you in accordance with the requirements of Technical Specification 6.6.B.1.c; an abnormal degradation discovered in the Reactor Coolant Pressure Boundary.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

N. J. Kalivianakis
Station Superintendent

NJK:DGC/bb

Enclosure

cc B. Rybak
A. Morrongiello
INPO Records Center

MAR 16 1984

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