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GGC-94-101

July 11, 1994

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) 94-005, Revision 00, for Quad Cities Nuclear Power Plant Station.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(iv). The licensee shall report any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature.

The following commitments are being made by this letter:

The piece of 1 inch tubing which was removed from the FASTC line will be sent to the System Material and Analysis Department (SMAD) for analysis to verify that this is the same failure mode as previously identified in a similar EHC 1/2 inch tubing crack.

The EHC System Engineer will evaluate replacing the one inch and two inch flared fittings at the turbine control valves.

A liquid penetrant examination will be performed on the one inch and two inch EHC tubing end connections for all four Unit One turbine control valves during the current (Q1R13) refuel outage.

If there are any questions or comments concerning this letter, please refer them to Nick Chrissotimos, Regulatory Assurance Administrator at 309-654-2241, ext. 3100.

Respectively,

COMMONWEALTH EDISON
QUAD CITIES NUCLEAR POWER STATION

L G Tucker

for G.G. Campbell
Station Manager

GGC/TB/plm

Enclosure

cc: J. Schrage
C. Miller
INPO Records Center
NRC Region III

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LICENSEE EVENT REPORT (LER) Form Rev. 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Quad Cities Unit Two	0 5 0 0 0 2 6 5	1 of 0 5

Title (4)

Unit Two Electro Hydraulic Control (ECH) Fluid Leak And Subsequent Manual Scram.

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 6	2 1	9 4	9 4	--	0 0 5	--	0 0	0 7	1 1	9 4		0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)				
04		20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)	
POWER LEVEL (10)	9 5	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)	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
Joe Manemann, Regulatory Assurance Ext. 3190	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 June 21, 1994, Unit 2 was in the RUN mode at approximately 95% power. An Equipment Operator reported that the Electro Hydraulic Control (EHC) fluid reservoir level was rapidly decreasing. At 1405 the Reactor was manually scrammed and the main turbine was manually tripped due to anticipation of low EHC control oil pressure.

The source of the EHC fluid leak was discovered on a one inch EHC line to turbine control valve #2. It appears that the EHC line failure was due to high cycle fatigue crack propagation at the radius of the tube flare. The exact source of the cycle stresses responsible for the fatigue crack initiation and propagation could not be determined. A likely cause is that the stresses were induced by vibration on the EHC line caused by oscillations of the #2 turbine control valve.

The Causal Factors (C/F) for this event are attributed to Equipment Specification, Manufacture, Construction and Plant/System Operation.

Corrective actions that have been completed include: replacement of the EHC lines to the #2 control valve, liquid penetrant examination of the EHC lines to the #1, 3 & 4 control valves, replacement of the #2 control valve servo valve and circuit boards and recalibration of the #2 control valve controller loop.

Corrective actions to be completed include: analysis of the cracked tubing, evaluating the replacement of the flared fittings, and liquid penetrant examination of the EHC lines to the Unit 1 control valves.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2511 MWt rated core thermal power.

EVENT IDENTIFICATION: Unit Two Electro Hydraulic Control (EHC) Fluid Leak and Subsequent Manual Scram.

A. CONDITIONS PRIOR TO EVENT:

Unit: Two Event Date: June 21, 1994 Event Time: 1249
Reactor Mode: 04 Mode Name: RUN Power Level: 95%

This report was initiated by Licensee Event Report

RUN (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 June 21, 1994, Unit 2 was in the RUN mode at approximately 95% of rated core thermal power. At 1249 hours, the control room annunciator came up for Electro Hydraulic Control (EHC) fluid [TG] high/low level. An Equipment Operator (EO) was immediately dispatched to the EHC unit and he reported that the EHC fluid reservoir level was rapidly decreasing. The EO began adding a barrel of EHC fluid to the reservoir but the level did not increase.

Visual inspection identified that EHC fluid was spraying from an EHC line at the #2 turbine control valve [PCV] and at 1320, an emergency load drop was started using the recirculation pumps [AD]. At 1405, with the Reactor at 53% power, the standby EHC pump automatically started on low EHC oil pressure. The Reactor was manually scrammed and the main turbine was manually tripped in anticipation of an automatic turbine trip from low EHC control oil pressure.

The expected reactor water level transient, due to the collapse of voids following the scram, caused reactor water level to drop below the low level setpoint of +8 inches. Group II and III Primary Containment Isolations (PCI) [JM] were received along with Reactor Building Ventilation Isolation [VA], Control Room Ventilation Isolation [VI] and Standby Gas Treatment (SBGT) [BH] initiation.

At 1435 the EHC fluid leak at the #2 turbine control valve was verified to have stopped as a result of tripping the turbine. At 1445, a reactor cooldown was started and at 0345 on June 22, the reactor reached cold shutdown.

An Emergency Notification System (ENS) notification of this event was completed at 1613 hours on June 21, 1994 to comply with the requirements of 10CFR50.72(b)(2)(ii).

There were no other systems or components inoperable at the beginning of this event which could have contributed to this event.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with 10CFR50.73(a)(2)(iv), which requires the reporting of any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF) [JE], including the Reactor Protection System (RPS) [JC].

The source of the EHC fluid leak was discovered on the one inch Fluid Actuator Supply Trip Control (FASTC) line to turbine control valve #2. The leak was located at the valve where the 1 inch stainless steel tubing mates with a straight thread adapter which threads into the bottom head of the turbine control valve actuator. A 180 degree circumferential crack was found in the radius of the 37 degree flared end of the 1 inch tubing. The flared end of the tubing mates with the flared male end of the straight thread adapter.

The following is a summary of Conclusions and Causal Factors (C/F) contributing to equipment malfunctions.

C/F: Equipment Specification, Manufacture and Construction

It appears that the EHC line failure was due to high cycle fatigue crack propagation. Fatigue crack initiation occurred at the radius of the tube flare because this radius acted as a stress concentrator. The hardness of the tube flare was greatly increased due to work hardening of the stainless tube metal when the flare was formed. General Electric (GE) Technical Information Letter (TIL) 841-3a recommends replacement of control valve flared fittings with connecter fittings that have a machined flare on one end and are socket welded to the stainless steel tubing.

C/F: Plant/System Operation

The exact source of the cycle stresses responsible for the fatigue crack initiation and propagation could not be determined. A likely cause is that the stresses were induced by vibration on the EHC line caused by small repetitive oscillations of the #2 turbine control valve.

D. SAFETY ANALYSIS OF EVENT:

The safety significance of this event was minimal. All manual and automatic Engineered Safety Features (ESF) occurred as designed to bring the reactor to a safe shutdown condition. If the leak had gone undetected, then level in the EHC reservoir would have decreased below the pump suction lines. This would have caused EHC system pressure to decrease and a turbine trip would have occurred at 1100 psig. An automatic reactor scram would have been initiated upon 10 percent closure of the turbine stop valves. The turbine bypass valves would have opened for approximately one minute due to the accumulator and check valve arrangement and then closed on loss of EHC pressure. This event is discussed in section 15.2.3.1 of the UFSAR and concludes that the Minimum Critical Power Ratio (MCPR) limit would not exceed the fuel cladding integrity safety limit.

E. CORRECTIVE ACTIONS:

Mechanical Maintenance (MM) and System Engineering personnel disassembled and inspected the one inch FASTC connection to turbine control valve #2. The one inch and two inch EHC lines for the #2 turbine control valve were replaced with new lines that had their 37 degree flares installed using a hydraulic flaring tool. The two inch line was replaced as a preventive maintenance measure.

A liquid penetrant examination was performed on the one inch and two inch EHC tubing end connections for the remaining three Unit 2 turbine control valves. Both lines on turbine control valves #1, #3 and #4 were found acceptable.

A walkdown of the Unit Two EHC system tubing was performed by the System Engineer after the system was repressurized. No additional leaks were identified.

The following work was performed to correct the small repetitive oscillations of the #2 turbine control valve:

- Replaced the signal cable from the linear variable differential transformer (LVDT) to the local junction box
- Inspected the junction box and found no problems
- Replaced the electrical ground strap bolt that was found loose
- Replaced the #2 turbine control valve servo valve and circuit boards
- Recalibrated the #2 turbine control valve controller loop

The piece of 1 inch tubing which was removed from the FASTC line will be sent to the System Material and Analysis Department (SMAD) for analysis to verify that this is the same failure mode as previously identified in a similar EHC 1/2 inch tubing crack (NTS #2651809400501).

After General Electric Technical Information Letter (TIL) 841-3a was issued, the 1/2 inch lines were removed from both units turbine control valves. The decision was made at that time not to convert the remaining portion of the piping to a welded system. The EHC System Engineer is currently evaluating replacement of the one inch and two inch flared fittings at the turbine control valves (NTS #2651809400502).

A liquid penetrant examination will be performed on the one inch and two inch EHC tubing end connections for all four Unit One turbine control valves during the current (Q1R13) refuel outage (NTS#2651809400503).

F. PREVIOUS EVENTS:

A review of previous Licensee Event Reports (LER) at Quad Cities Station Units One and Two, since January 1, 1988 revealed one (Unit 1 LER 92-019) previous EHC leak that resulted in a ESF actuation. The direct cause of the 1992 reactor scram was attributed to insufficient training by the operating crew on cooling down the reactor using turbine bypass valves. The initiating event was a 1/2 inch EHC line flare fitting crack similar to this event. At that time the failure mode was believed to be limited to 1/2 inch tubing.

G. COMPONENT FAILURE DATA:

The EHC tubing which failed was annealed, thick walled, AISI Type 304 Stainless Steel. The fittings for the EHC system are straight thread o-ring type, and 37 degree flare type fittings made of 316 stainless steel. The tubing and fittings are furnished by the GE factory.