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RLB-92-199

September 14, 1992

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Reference: Quad Cities Nuclear Powcr Station
Docket Number 50-254 DPR-29, Unit One

Enclosed is Licensee Event Report (LER) 92-019, Revision 00, for Quad Cities Nuclear Power 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 s'all report any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD CITIES NUCLEAR POWER STATION

C. Tietz for
R. L. Bax
Station Manager

RLB/TB/plm

Enclc ure

cc: J. Schrage
T. Taylor
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 One	0 5 0 0 0 2 5 4	1 of 0 6

Title (4)

Reactor Water Low Level Scram Due To Insufficient Operator Training

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
										0 5 0 0 0
0 8	1 5	9 2	9 2	0 1 9	0 0	0 9	1 4	9 2		0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 0 6 4

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	X 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)	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	
Rob Korneta, Technical Staff Engineer, Ext. 2158	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)

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)

ABSTRACT:

On August 15, 1992, Quad Cities Unit One was in the RUN mode at 64 percent of rated core thermal power. At 0157 hours, an emergency load drop was initiated due to an unisolable Electro-Hydraulic Control (EHC) fluid leak. At 0248 hours, the Main Turbine was manually tripped in an attempt to isolate the EHC leak. The reactor was then manually scrammed in preparation for securing the EHC system. At 0303 hours, an automatic scram occurred when reactor water level dropped below the low level setpoint of +8 inches. All safety feature actuations occurred as designed. An Emergency Notification System (ENS) notification was completed at 0548 hours on August 15, 1992.

The cause of the low level scram was attributed to insufficient training by the operating crew on cooling down the reactor using Turbine Bypass Valves. Corrective actions will include training on this event during license requalification. The simulator response to this transient will also be evaluated and the computer code adjusted, as necessary, to ensure that the simulator more closely emulates actual plant response to this scenario.

This report is being submitted to comply with 10CFR50.73(a)(2)(iv).

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

PLANT AND SYSTEM IDENTIFICATION:

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

EVENT IDENTIFICATION: Reactor Water Low Level Sram Due To Insufficient Operator Training.

A. CONDITIONS PRIOR TO EVENT:

Unit: One	Event Date: August 15, 1992	Event Time: 0253
Reactor Mode: 4	Mode Name: RUN	Power Level: 64%

This report was initiated by Deviation Report D-4-01-92-087.

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 August 15, 1992, Unit One was in the RUN mode at 64 percent of rated core thermal power. At 0155 hours, Annunciator B6, EHC FLUID HI/LO LEVEL, illuminated on Control Room Panel [PL] 901-7. An Equipment Operator (EO) and both Shift Foremen (SF) were immediately dispatched to the Unit One Electro-Hydraulic Control (EHC) [TG] fluid reservoir [RVR] to investigate. The EO reported that there were no visible EHC leaks at the fluid reservoir, but fluid level was observed to be rapidly decreasing.

At 0157 hours, an emergency load drop was initiated to below 40 percent power by reducing Reactor Recirculating Pump [P] speeds and inserting control rods according to the approved sequence. A SF entered the Low Pressure Heater Bay and determined that the fluid was leaking from a 1/2 inch EHC line at the #3 Turbine Control Valve (TCV) [PCV]. At that time it was determined that the leak was unisolable. Reactor [RCT] power was then further reduced and Turbine Bypass Valves [PCV] were opened to reduce Main Generator [GEN] load.

At 0248 hours, the Main Turbine [TRB] was manually tripped from Control Room Panel 901-7. This action was taken in an effort to reduce or isolate the EHC leak. The EHC leak did not isolate; therefore, in preparation for a reactor scram, reactor water level was raised to approximately +40 inches in accordance with QCGP 2-3, Reactor Scram. A manual scram was then inserted at 0253 hours.

The expected reactor water level transient due to the collapse of voids following the scram caused reactor water level to drop to approximately +12 inches. Group II and III Primary Containment Isolations (PCI) [JCI] were received along with Reactor Building [NG] Ventilation Isolations [VA], and Standby Gas Treatment [SBGT) [BH] initiation.

Reactor water level recovery was accomplished by automatic operation of the Low Flow Feedwater Regulating Valve (FWRV) [LCV] and stabilized at approximately +40 inches. At 0257 hours, a reactor cooldown was commenced to reduce reactor pressure in preparation for securing the EHC system. Initially, Turbine Bypass Valve (BPV) #1 was throttled to the 1/2 open position, but after discussions with the Assistant Superintendent of Operations (ASO) it was decided to increase the cooldown rate by opening additional bypass valves. The Nuclear Station Operator (NSO) then slowly opened BPV #1 to the full open position. BPV #2 was then slowly throttled to the 1/2 open position. Reactor water level was initially stable but then began an increasing trend which was attributed to swell.

At 0300 hours, the 1A Reactor Feed Pump (RFP) tripped due to high reactor water level (+48 inches). As reactor water slowly decreased, the BPV's were throttled closed until only 1 BPV remained opened.

At 0303 hours, with reactor water level approximately +28 inches, the 1A RFP was restarted. The NSO then throttled closed the BPV until only 1/2 BPV remained open. The throttling down of the Turbine Bypass Valves and the addition of colder feedwater to the reactor vessel caused water level to rapidly decrease to approximately -1 inch. A second Group II and III PCI, Reactor Building Ventilation Isolation and SBGT initiation occurred along with an automatic reactor scram signal from reactor low water level (+8 inches). No control rod movement occurred since the control rods were already fully inserted.

Reactor water level recovery was accomplished by automatic operation of the Low Flow FWRV. The NSO completely closed BPV #1 which was 1/2 open. Reactor water level was stabilized at approximately +28 inches.

At 0310 hours, the Reactor Core Isolation Cooling (RCIC) [BN] system was placed in operation to control reactor pressure. At 0319 hours, the EHC system was shutdown to stop the leak at the #3 TCV.

An Emergency Notification System (ENS) notification of this event was completed at 0548 hours on August 15, 1992 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), including the Reactor Protection System (RPS) [JC].

The source of the EHC fluid leak was discovered on the 1/2 inch Fluid Jet Supply (FJS) line to TCV #3. The leak was located at the valve actuator where the 1/2 inch stainless steel tubing mates with a straight thread adapter which threads into the bottom head of the TCV actuator. A 180 degree circumferential crack was found in the radius of the 37 degree flared end of the 1/2 inch tubing. The flared end of the tubing mates with the flared male end of the straight thread adapter.

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.

The exact source of the cycle stresses responsible for the fatigue crack initiation and propagation could not be determined. However, it is postulated that the stresses were induced by normal system vibration on the EHC line.

The cause of the reactor low water level scram was attributed to insufficient training for the operating crew on cooling down the reactor vessel using Turbine Bypass Valves. Both actions taken by the Nuclear Station Operator (NSO), the addition of colder feedwater (approximately 160 degrees F) to the reactor vessel and the throttling down of the BPVs, caused collapsing of the voids normally present inside the reactor vessel. This caused reactor water level to decrease. Performing the above actions simultaneously caused reactor water level to decrease at a rate greater than could be compensated for by the Low Flow FWRV. This allowed reactor vessel level to drop below the low water level scram setpoint of +8 inches.

Discussions with the operating crew involved indicated that the simulator response to this type of event seemed different than what actually occurred during this event. Reactor water level perturbations in response to BPV movement appear to be much less severe on the simulator than what actually occurred on Unit One during this event.

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. The reactor vessel low water level scram occurs when level inside the reactor drops below +8 inches. This scram is intended to prevent fuel damage following abnormal operational transients by single equipment malfunctions or single operator errors that result in a decreasing reactor vessel water level. The lowest water level reached during this event was approximately -1 inch. This was still 142 inches above the top of active fuel. If reactor vessel level had continued to decrease, Emergency Core Cooling Systems were available to provide makeup water to the vessel once level dropped to -59 inches. In this event, all control rods were already inserted into the core which further minimized any transients.

E. CORRECTIVE ACTIONS:

Mechanical Maintenance (MM) and Technical Staff personnel disassembled and inspected the 1/2 inch FJS connection to TCV #3 according to Nuclear Work Request (NWR) Q02512. Approximately one inch of tubing was removed from the end the FJS line which completely encapsulated the circumferential crack. A new 37 degree flare was mechanically installed in the end of the FJS line. A new straight thread adapter was obtained and the 1/2 inch FJS line was reconnected to TCV #3 actuator.

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

The piece of 1/2 inch tubing which was removed from the FJS line will be sent to the System Material and Analysis Department for analysis in an attempt to discover the root cause of the circumferential crack. Further corrective actions will be based on the results of this analysis (NTS #2542009208701).

The Technical Staff System Engineer is currently evaluating the implementation of General Electric Technical Information Letter (TIL) 841 which recommends eliminating the 1/2 inch FJS lines to the turbine steam valves. FJS to the servo-valves can then be taken from the Fluid Actuator Supply (FAS) lines by addition of a special manifold mounted under each servo-valve. This TIL was recommended due to industry failures experienced in the 1/2 inch tubing (NTS #2542009208702).

This event will be incorporated into both the "Modifications/Lesson Learned" portion of operator retraining class, as well as required reading. This training will stress enhanced operator awareness to reactor water level changes when cooling down the reactor vessel using Turbine Bypass Valves (NTS #2542009208703).

The simulator response to this transient will be evaluated and the computer code adjusted as required to ensure that the simulator more closely emulates actual plant response to this type of scenario (NTS #2542009208704).

F. PREVIOUS EVENTS:

A review of previous Deviation Reports (DVR)/Licensee Event Reports (LER) at Quad Cities Station, back to 1988 revealed two events in which unexpected feedwater level transients were attributed to reactor scrams following Turbine BPV manipulations during reactor cooldowns.

LER 04-02-91-007 involved a reactor low water level scram following a RFP trip on high level. A BPV was initially opened following the RFP trip but soon after was throttled closed. Reactor water level began to drop rapidly from approximately +39 inches. A low level scram was received three minutes later before a RFP could be started. Corrective action for this low level scram involved the addition of the following Precaution to QCGP 2-1, Normal Unit Shutdown and QCGP 2-3 Reactor Scram: Prior to starting a Reactor Cooldown, verify a high pressure source of make-up water (i.e., Feedwter system, HPCI, RCIC) is available to maintain Reactor Water Level.

LER 04-02-90-010 involved decreasing the pressure setpoint following a reactor scram from Moisture Separator high level. The pressure setpoint had been increased above reactor pressure in accordance with QOS 250-1, Pressurizing The Main Steam Lines Following A Group I Isolation. As the NSO decreased pressure set to establish a cooldown, two BPVs opened. The increased steam flow through the BPVs, coupled with no RFP in operation caused reactor water level to decrease rapidly to below the low level scram setpoint. There were no corrective actions initiated as a result of the low level scram.

A search of maintenance history files revealed a similar 1/2 inch EHC tubing failure on April 8, 1988. The tubing was repaired under NWR Q65698.

G. COMPONENT FAILURE DATA:

The EHC tubing which failed was AISI Type 304 Stainless Steel. The reactor low water level scram was not associated with any component failure.