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RLB-92-061

March 6, 1992

U. S. Nuclear Regulatory Commission
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Reference: Quad Cities Nuclear Power Station
Docket Number 50-254, DPR-29, Unit One

Enclosed is Licensee Event Report (LER) 92-004, 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 shall 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

Gary Spedl for RLB

R. L. Bax
Station Manager

RLB/TB/rjb

Enclosure

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 9

Title (4) Unit One Reactor Scram Due to A Group I Isolation Believed To Be Caused By A Spurious Main Steam Line High Flow Trip Due to An Unknown Cause

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)
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0 2	0 7	9 2	9 2	— 0 0 4 —	0 0	0 3	0 7	9 2		0 5 0 0 0

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

OPERATING MODE (9)	4	20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	1 0 0	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
David Harmon, Technical Staff Engineer Ext. 2116	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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
IM	J B	L I T S	I 2 0 4	Y	PE	S B	J X	G 0 8 0	Y
IE	S B	C N V	F 1 8 0	Y	CM	S B	R V	D 2 4 3	Y

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:

At 0201 hours on February 7, 1992, Unit One was in the RUN mode at 100% power. A Channel "B" Main Steam Line (MSL) high flow annunciator was received in the Control Room. Immediately thereafter, a full Primary Containment Isolation Group I isolation occurred and a subsequent reactor scram.

All automatic actuations occurred as designed with the exception that Reactor Feed Pumps (RFP) did not trip on +48 inches reactor high level. Additionally, the "C" Electromatic Relief Valve (ERV) failed to open upon manual initiation. Reactor shutdown was accomplished by 1100 hours.

The root cause of the Group I isolation could not be determined. It is believed to be due to spurious initiation of MSL high flow instrumentation. Monitoring instrumentation was installed to evaluate future similar events.

The RFP high level trip did not occur due to setpoint drift. The applicable instruments were calibrated and functionally verified. The "C" ERV did not actuate due to loss of continuity between solenoid electrical contacts. The ERV's were inspected and all worn parts were repaired or replaced.

This report is being submitted to comply with 10 CFR 50.73 (a)(2)(iv).

PLANT AND SYSTEM IDENTIFICATION:

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

EVENT IDENTIFICATION: Unit One Reactor Scram Due To a Group I Isolation Believed To Be Caused By A Spurious Main Steam Line High Flow Trip Due To An Unknown Cause.

A. CONDITIONS PRIOR TO EVENT:

Unit: One Event Date: February 7, 1992 Event Time: 0201
Reactor Mode: 4 Mode Name: Run Power Level: 100%

This report was initiated by Deviation Report D-4-1-92-012

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:

At 0201 hours on February 7, 1992, Unit One was in the RUN mode at 100% of rated core thermal power. The High Pressure Coolant Injection [BJ] (HPCI) system was out-of-service in day one of a fourteen day Limiting Condition for Operation (LCO) due to a stop valve [VLV] failure. At this time, Annunciator [ANN,IB] C-16, "CHANNEL B MAIN STM LINE HIGH FLOW", was received on the 901-5 panel in the Control Room. Immediately thereafter, a full Primary Containment Isolation [JM] (PCI) Group I isolation occurred, and a subsequent reactor [RCT] scram due to Main Steam Line Isolation Valve [SB,JM] (MSIV) closure. The Reactor Mode Switch was moved to SHUTDOWN as per procedure QCGP 2-3, REACTOR SCRAM. The "B" Main Steam Line [SB] (MSL) flow indicator [FI] (FI) on the 901-5 panel in the Control Room, FI-1-640-23B, was observed to be spiking erratically during the event FI-1-640-23C was observed to be indicating off-normal.

Reactor water level dipped to approximately -20 inches due to rapid power decrease and steam void collapse and then immediately began to recover. As designed, upon reactor water level reaching the low level trip of +8 inches, PCI Group II and III isolations occurred, Standby Gas Treatment [BH] (SBGT) autostarted, and Reactor Building Vents [VA] (RBV) isolated. The "B" Reactor Feedwater Pump [P,SJ] (RFP) was taken off to help control rising reactor water level. At 0202 hours, reactor water level reached +48 inches and continued to rise. The Main generator [TB,GEN] protective relaying [RLY] scheme sensed a reverse power condition, caused by no steam flow to the Main Turbine [TA,TRB]. This reverse power condition energized the back-up lock-out relay and tripped the Main Generator and Main Turbine. Auxiliary power was transferred automatically upon the generator trip. The Shift control Room Engineer (SCRE) initiated entry into General Abnormal Procedure QGA 100, REACTOR PRESSURE VESSEL CONTROL.

At 0203 hours the Shift Engineer entered the Control Room and assumed command and control following a short briefing from the SCRE. Reactor water level continued to increase to greater than +52 inches. The Unit One NSO recognized that the +48 inches feedwater pump trip did not occur, and manually tripped the "A" RFP.

At 0205 hours, reactor pressure reached 1041 pounds per square inch (PSI). Under Shift engineer direction, the extra NSO opened the "B" Electromatic Relief Valve [RV,20,SB] (ERV) to control reactor pressure between 800 and 1000 psi. The acoustic monitor [MON,SB,JE] for the "B" ERV gave erratic indication during this event. At 0206 hours, reactor pressure peaked at 1052 psi. In anticipation of a rise in suppression pool [NH] water temperature due to ERV actuation, the Residual Heat Removal [BO] (RHR) system was placed in operation in the torus [NH] cooling mode at 0207 hours.

At 0208 hours, the Unit One NSO continued water level control and established Reactor Water Clean-up [CE] (RWCU) blowdown. Although the MSL high flow annunciation was the only condition present indicative of a MSL break, the Shift Engineer dispatched the Shift Foreman to inspect for evidence of a steam line break. Additionally, he directed Equipment Attendants (EA) to investigate the possibility of accidental damage or bumping of the differential pressure (DP) switches which initiate a MSL high flow signal. The Instrument Maintenance (IM) Foreman was also sent to check the dp switches for any abnormalities.

At 0212 hours the Unit One NSO started the "A" RFP to maintain level between +8 and +48 inches as per QGA 100. Reactor pressure had slowly decreased to 800 psi and the extra NSO closed the "B" ERV at 0214 hours. At 0219 hours, the extra NSO placed the Reactor Core Isolation Cooling [BN] (RCIC) system in service in the pressure control mode to assist in reactor pressure control. However, pressure continued to slowly increase to 1000 psi.

At 0228 hours, the extra NSO attempted to open the "C" ERV. The "B" and "C" ERV's are to be opened alternately as per operating procedure QCOP 203-1, REACTOR PRESSURE CONTROL USING MANUAL RELIEF VALVE ACTUATION. The "C" ERV did not open as indicated by the following: the open light [IL] did not illuminate, the acoustic monitor [MON,IJ] did not actuate, and reactor pressure continued to rise.

At 0229 hours, the "B" ERV was re-opened. Reactor pressure peaked at approximately 1018 psi. As pressure began to drop, reactor level took a sharp increase due to void swelling. AT 0230 hours, noticing this change, the Unit One NSO tripped the "A" RFP. Within three minutes , water level was approximately 30 inches and decreasing. The "A" RFP was restarted. However, the pump did not achieve the necessary pressure quick enough and, at 0233 hours, Group II and III isolations occurred at +12.7 inches indicated reactor water level. An additional reactor scram signal was received but no rod motion occurred because the initial scram had not been reset at this point.

At 0236 hours, the extra NSO closed the "B" ERV. RWCU blowdown was re-established for water level control at 0239 hours.

At approximately 0245 hours, the IM Foreman reported that the MSL high flow dp switches all appeared to be indicating normal. Investigations by the operating crew could find no evidence of a MSL break, nor any evidence that personnel were in the area of the MSL high flow switches at the time of this event. Therefore, the Group I isolation was reset and the MSIV's were opened. The "B" inboard MSIV was left closed because of erratic behavior of Flow Indicator [FI,SB] (FI) 1-640-23B as noticed by the Unit One NSO during the recovery operations. The reactor scram was reset at 0317 hours and Reactor Building Ventilation was reset at 0330 hours. At 0400 hours RCIC was taken off, as the Main Condenser [SG,COND] was being utilized as the heat sink for removing reactor heat. At 0402 hours, procedure QGA 100 was exited.

An Emergency Notification System (ENS) phone notification was completed at 0412 hours on February 7, 1992, as required under 10CFR50.72 (b) (2) (ii).

At approximately 1100 hours, the reactor was brought to cold shutdown with reactor water temperature less than 212 degrees. An investigation team was formed in accordance with QAP 1780-11. An investigation report was given to the station prior to start-up.

Procedures QIS 21-1, MSL HIGH FLOW CALIBRATION, and QIS 21-2, MSL HIGH FLOW FUNCTIONAL TEST, were completed for each of the 16 MSL high flow dp switches. As per Technical Specification Table 3.2-1, the trip setting for MSL high flow is ≤140% of rated steam flow, which is equivalent to 148 pounds per square inch differential (psid). The as found data showed that all 16 switches tripped within the Technical Specification limit.

Work Request #Q97927 was written to investigate the failure of the "C" ERV. Troubleshooting the actuator, EM personnel identified a resistance of 182 ohms across the shorting bar and contacts of the cut off switch. The switch was replaced and resistance measured to be less than 1 ohm. A reddish dust was observed within the actuator housing. The "B", "D", and "E" ERV's were actuated after cold shutdown and all were verified to operate properly.

Work Request #Q97935 was written to investigate the RFP reactor high level trip which should have occurred at +48 inches. Level Indicating Transmitter With Switches [LIT, LS, JB] (LITS) 1-263-59A and LITS 1-263-59B provide for this high level trip. Switch #4 from each LITS is arranged such that both switches must open to trip the RFP's and the turbine. LITS-1-263-59A Switch #4 was found to trip at a reactor level of +53.5 inches. LITS-1-263-59B Switch #4 was found to trip at a reactor level of +48.1 inches. The trip of the RFP's would have occurred at +53.5 inches. The A and B switches were recalibrated to trip at 47.6 and 48.7 inches reactor level, respectively.

The erratic MSL flow indication was investigated. There are four MSL flow indication loops, each composed of a dp transmitter [PDT] (DPT) (1-645A, B, C, & D), power supply [JX], square-root converter [CNV], and Control Room indicator (FI-1-640-23A, B, C, & D). These loops have no trip function. The "B" loop power supply was found to be faulty, creating a spurious spike. All four square-root converters were identified as having a non-linearity problem resulting in inaccurate readings at low flows. The 1-645B transmitter was replaced under Minor Design Change #P04-1-90-092 which was implemented by Work Request #Q97971. The 645A, C, & D transmitters were calibrated satisfactorily.

C. APPARENT CAUSE OF EVENT:

This report is being submitted to comply with 10CFR50.73 (a)(2)(iv); "The licensee is required to report any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature [JE] (ESF), including the Reactor Protection System [JC] (RPS), except an actuation which is part of a preplanned sequence during testing or reactor operation."

The apparent cause of this event is a Group One Isolation caused by MSL high flow signal due to an unknown cause. MSL high flow annunciation was received in the Control Room and no evidence of a MSL break could be found. The MSL high flow switches were calibrated and functionally tested and found to be within Technical Specification limits. A search of past history of these switches showed excellent accuracy and reliability. A walkdown inspection of the sensing line piping and all electrical connections was performed and no abnormalities were found. An extensive search of security data and radiation area access control revealed a very limited number of personnel could have been in the vicinity at the time of the reactor scram. Interviews concluded that no one was in the area near the racks at the time of the trip. Two flow check valves were removed from the sensing lines of the dp switches to inspect for possible blockage. No blockage was found. There are four dp switches on each MSL. They are arranged in PCI initiation logic in a (1 of 4) out of 2 taken twice logic, such that the right combination of 2 of the 4 switches connected to the same MSL can initiate a full Group I isolation.

The switches were pressurized to simulate normal operating dp, and vibration induced testing methods were used to test the sensitivity of the switch actuations. No actuations occurred during extensive testing. Since no evidence of an actual steam flow or pressure transient could be identified and all the associated equipment was found to be working properly, the root cause of the Group I isolation and subsequent reactor scram remains unknown.

There were two contributing causes which resulted in the failure of the "C" ERV. Vibration induced wear created enough brass and phenolic dust from components within the actuator to result in a loss of continuity between the solenoid electrical contacts. Also, a more thorough root cause analysis from a similar recent event could have prevented this failure. (Reference DVR# 4-1-91-131 described in the Previous Events section.)

The root cause of the failure of the RFP high level trip is setpoint drift of LITS-1-263-59A Switch #4. The switch is calibrated to trip at 48 ±1.7 inches reactor water level. Upon investigation, this switch was found to trip at +53.5 inches reactor water level. The switch was last calibrated on July 20, 1991, to trip at 46.7 inches.

The cause of erratic MSL flow indication on FI-1-640-23B was a faulty power supply. Also, the 1-645B transmitter was determined to be in need of replacement due to calibration adjustment problems and drift history. The cause of off-normal indication on FI-1-640-23C was non-linearity problems with the square root converter.

The "B" ERV Acoustic Monitor was inspected by Instrument Maintenance (IM) personnel. The erratic indication was determined to be due to the mounting clamp having become loose. The clamp was inspected and no wear or abnormalities were found. The clamp was tightened and the acoustic monitor was verified to be operating properly.

D. SAFETY ANALYSIS OF EVENT:

The safety of the public and plant personnel was not affected by this event and the safety significance of this event was minimal. Both vessel pressurization and consequences of a loss of coolant accident have been previously analyzed for a situation with the HPCI system and one relief valve out of service.

During this event, the main feedwater system [SJ] was available at all times to maintain reactor water level. Reactor water level was maintained at least 120 inches above the top of active fuel at all times, thereby assuring adequate core cooling.

Vessel overfill was not a concern during this event since the highest reactor water level reached during this event was approximately +60 inches, which is approximately 47 inches below the main steam lines. Although the reactor feed pumps were manually secured by the NSO, subsequent calibration and functional testing of the trip instruments showed that the automatic trip would have occurred at a reactor water level of +53.5 inches. Therefore, adequate margin was available for vessel overfill protection.

The highest reactor pressure achieved during the event was 1052 psi, which is 63 psi below the lowest relief valve setpoint and 293 psi below the reactor pressure safety limit. Reactor pressure control during the time that the MSIV's were closed was accomplished according to the Emergency Operating Procedure within a band of 780 to 1015 psi using the "B" ERV. The failure of the "C" ERV did not hinder reactor pressure control during the time that the MSIV's were closed. The "B" ERV operation was sufficient to control pressure with the amount of decay heat present at the time of the event. The "A", "D", and "E" relief valves would have been available to control reactor pressure if they had been needed during the event. In addition, the failure of the "C" ERV would not have degraded the performance of the Automatic Depressurization System [SB,JE] (ADS) below that assumed in the transient and accident analysis previously performed for Quad Cities Station.

All automatic actions, except for the RFP high level trip described above, functioned as expected during the event.

E. CORRECTIVE ACTIONS:

The immediate corrective actions taken were to use procedures QCGP 2-3, REACTOR SCRAM, and QGA 100, REACTOR PRESSURE VESSEL CONTROL, to safely control reactor pressure, level, and other parameters following the scram. The reactor was depressurized and brought to cold shutdown conditions.

A pressure transducer [TD] was installed in each of the eight sensing lines for the MSL dp switches. Recorders [PR,PDR] were installed prior to start-up to continuously monitor MSL pressure and dp, and to monitor the MSL high flow switches and the MSL low pressure switches at the 901-15 and 901-17 panels [pnl] in the Control room. Minor Design Change (MDC) P04-1-92-021 was installed to log all PCI Group I relay actuations on the Sequential Event Recorder [IQ]. The recorders and MDC P04-1-92-021 will enhance the Station's ability to evaluate any future events involving the MSL dp switches and PCI Group I relays.

The investigation of the ERV's identified that some vibration is inherent to the MSL's and that complete mitigation of the vibration is not likely. Therefore, the following corrective actions have been completed or are in progress. The "B", "D", & "E" ERV actuators were also inspected. The resistance of the shorting bar and contact of these valve actuators varied from 0.2 to 8 ohms. All were cleaned reducing the resistance to less than 0.5 ohms. The reddish dust was observed in the "D" and "E" ERV's as well. All worn parts were repaired or replaced on each ERV actuator. The Station will enhance its maintenance procedures to include acceptance criteria for resistance across the shorting bar, a periodic inspection of the actuator parts, and lubrication of actuator parts which could exhibit wear (NTS #254200?201201). The applicable parts were lubricated prior to starting up Unit One. The Station will evaluate the actuator brass parts for possible material replacement (NTS #2542009201202). To prevent repeat failures, the Station will evaluate its failure analysis process to assure critical equipment failures are sufficiently investigated prior to start-up from outages and re-start from scrams. (NTS #2542009201203).

The instruments that caused the RFP high reactor level trip, LITS-1-263-59A&B, were recalibrated and functionally tested. The calibration interval of these transmitters on both units will be changed from once per refuel cycle to quarterly to minimize the possibility of excessive drift (NTS #2542009201204). The system engineer has provided Operations personnel with a discussion of the RFP high level trip tolerances and inaccuracy of the indications. This information has been covered during shift turnover meeting. An evaluation on the replacement of the transmitters with appropriate state-of-the-art technology determined that the transmitters are functioning as designed, but the equipment is obsolete. The Station will determine the need for upgrading the transmitters with new models (NTS #2542009201205).

The "B" MSL flow transmitter, FT-1-645B, was replaced with a new model as per MDC P04-1-90-092. The "B" MSL flow indication loop power supply was replaced. The square-root converters for all four indication loops were replaced with calibrated units having no non-linearity problems.

The "B" ERV Acoustic Monitor was repaired by tightening its clamp and verifying proper operation. The other ERV acoustic monitors functioned properly during startup testing.

F. PREVIOUS EVENTS:

The following previous similar events are summarized below:

Date	DVR#	Description
CAUSE ASSOCIATED WITH MSL HIGH FLOW SWITCHES:		
6/23/89	4-2-89-032	1/2 Group I due to Instrument Maintenance Tech bumping MSL high flow switch after performing functional procedure.
1/30/90	4-2-90-003	1/2 Group I due to Contractors bumping the MSL high flow switches.
CAUSE UNKNOWN:		
11/3/90	4-2-90-064	Spurious Group I Alarm
3/18/91	4-1-91-045	Spurious MSL high flow alarm and 1/2 Group I
4/26/91	4-1-91-070	Spurious 1/2 Group I

ASSOCIATED WITH ERV'S:

4-1-90-073	Failure of "C" ERV to open due to worn bushing in solenoid valve.
4-1-91-131	Failure of "B" ERV to open due to a defective cutout switch and binding of the actuator. The shorting bar exhibited high resistance and the guide assembly of the actuator was found to be bent slightly.

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

A nationwide Nuclear Plant Reliability Data System (NPRDS) search was conducted for the 59A&B LITS, the 645B transmitter, the square-root converters, and the loop power supply. The results are as follows:

	Total Failures Nationwide	Failures at Quad Cities Station
59A & B LITS	8	3
645B transmitter	21	0
Square-Root Converter	2	0
Power Supply	22	0

An NPRDS search was recently conducted for the ERV's as per Deviation Report 4-1-91-031. Eighteen failures nationwide were reported. The ERV Acoustic Monitor is not an NPRDS reportable item.

G. COMPONENT FAILURE DATA:

The MSL high flow switches are manufactured by Barton, model 288. LITS 1-263-59A&B, which provide for reactor high level RFP and turbine trips, are manufactured by Yarway, model 4418CE. The "C" ERV, 1-203-3C, is a 6-inch automatic relief valve manufactured by Dresser Industries Inc., model 1525-VX. The failed MSL flow loop components are as follows:

Flow Transmitter FT-1-645B	Barton, model 296
Square-Root Converters 1-640-39A,B,C&D	Foxboro, model 66AT-OH
Loop Power Supply 1-640-10	General Electric, model 50-570062FAAC1

The "B" ERV Acoustic Monitor is manufactured by NDT International, model 104D.