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RLB 93-039

February 24, 1993

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

Reference: Quad Cities Nuclear Power Station
Docket Number 50-265, DPR-30, Unit Two

Enclosed is Licensee Event Report (LER) 93-005, Revision 00, for Quad Citie 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

H.J. Hentschel for
R.L. Bax
Station Manager

RLB/TB/as

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)
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Title (4)

ESF Actuation from Spurious High Reactor Pressure due to Personnel Error.

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 1	2 9	9 3	9 3	0 0 5	0 0	0 2	2 6	9 3		0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 1 0 0

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	
James N. Dinh, T.S. Engineer Ext. 2065	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)

___ 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:

At 1340 hours on January 29, 1993, Unit Two was in the RUN Mode when a full SCRAM occurred. The SCRAM was initiated by a spurious high reactor pressure signal. The indicated reactor pressure was normal for plant operations.

The apparent cause is due to a contractor accidentally bumping the reactor high pressure switches' common sensing line while performing the Reactor Vessel Water Level Indicating System modification work.

This event will be tailgated with groups performing work in the plant to enhance personnel awareness of the sensitivity of the reactor high pressure switches and their associated sensing lines.

PLANT AND SYSTEM IDENTIFICATION:

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

EVENT IDENTIFICATION: ESF Actuation from Spurious High Reactor Pressure due to Personnel Error.

A. CONDITIONS PRIOR TO EVENT:

Unit: Two	Event Date: January 29, 1993	Event Time: 1340
Reactor Mode: 4	Mode Name: Run	Power Level: 100%

This report was initiated by Deviation Report D-4-02-93-009

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 1340 hours on January 29, 1993, Unit Two was in the RUN Mode at 100 percent of rated core thermal power when a full SCRAM occurred. The first alarm was Reactor High Pressure, from panel 902-5, window C-13, and was noted by the Unit Two Nuclear Station Operator (NSO). Reactor pressure was indicating 1005 psig, which is normal for full power operation

The Shift Control Room Engineer (SCRE) ordered the NSO to enter QGA 100, Reactor Pressure Vessel Control procedure, to control the reactor pressure and water level. The reactor water level dropped to +8 inches due to steam voids collapsing. Primary Containment Isolation (PCI) [JM] Group II, and III isolations, along with Reactor Building and Control Room Vent isolations and Standby Gas Treatment (SBGT) [BH] auto-start, occurred as designed due to the low reactor water level. The Unit Two NSO then performed routine SCRAM recovery per QCGP 2-3, Reactor SCRAM procedure.

The Control Room event recorder logged a trip of Reactor high pressure switch (PS) [PS] 2-263-5C, which makes up half of the required logic for the high reactor pressure SCRAM. The second half of the logic is supplied by Reactor High PS 2-263-55B or 2-263-55D. Neither was identified by the event recorder to have tripped.

The Reactor Protection System (RPS) [JC] high reactor pressure monitoring consists of four pressure switches 2-263-55A, (B,C,D) which are set up in a one out-of-two taken twice logic. PS 2-263-55A, (C) and PS 2-263-55B, (D) make up channel A and channel B, respectively.

The Shift Engineer (SE) suspected the instrument racks 2202-5 and 2202-6, location of the reactor high pressure switches 2-263-55A(B,C,D), might have been disturbed. This had created spurious signals to the reactor high pressure switches and had caused the tripping of these switches in the past. The SE sent an Equipment Attendant (EA) to examine the instrument racks.

The EA found three contractors working on scaffolding above the 2202-6 instrument rack, installing pipe hangers for the Reactor Vessel Water Level Indication System (RVWLIS) modification. They were in the vicinity of the reactor vessel instrumentation sensing lines when the SCRAM occurred. No one else was found working at or near the instrument racks.

The RVWLIS modification work was put on hold pending further investigation.

At 1410 hours on January 29, 1993, the Reactor Building and the Control Room vents were reset and the SBGT train was turned off. The Reactor SCRAM recovery was satisfactory.

Instrument Maintenance technicians were sent out to test PS 2-263-55A(B,C,D). Reactor high pressure switches 2-263-55B(C,D) setpoints were found at 1046 psig, which is conservatively below the tolerance range of 1050 to 1060 psig. PS 2-263-55A setpoint was found at 1050 psig. All of the setpoints were readjusted to trip at the mid tolerance range.

At 0655 hours on January 31, 1993, Unit Two commenced startup.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with 10CFR50.73(a)(2)(iv), which requires that 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], be reported.

The apparent cause is due to a contractor accidentally bumping the reactor high pressure switches' common sensing line while performing the RVWLIS modification work. In the past, a great amount of bushhammering work, which caused an intense amount of vibration, was performed in installing piping clamps days earlier in the vicinity of the instrument sensing lines and no tripping of the reactor pressure switches resulted. Also, the same scope of work was performed for the Unit One RVWLIS modification and no tripping of the reactor pressure switches was reported.

The Unit Two NSO received the High Reactor Pressure alarm as the first hit. The NSO then verified the actual reactor pressure to be reading at 1005 psig which is normal for plant operations. This pressure indication was correct as verified by data from a control room recorder and computer points. This indicates that the SCRAM was caused by a spurious high reactor pressure signal.

The Control Room Event Recorder logged the tripping of Reactor High PS 2-263-55C at the time of the SCRAM. The tripping of Reactor High PS 2-263-55D, which is required to fulfill the full reactor SCRAM logic, was not identified by the event recorder to have tripped. It is believed that the spurious high pressure signal lasted long enough to trip the pressure switch 2-263-55D but not long enough to latch in the event recorder. All these indications signify the cause of the SCRAM to be from pressure switches 2-263-55C and D due to inadvertent bumping of their common sensing line.

The SCRAM occurred while the contractors performed a rough fitting of a prefabricated piping section. One of the contractors apparently bumped into PS 2-263-55C and D common sensing line. This vibration would have induced a spurious signal to the Reactor High PS's and caused the SCRAM.

These Barksdale PS's have displayed great sensitivity to vibration historically. It is maintained that during the installation of the protective cages, now in place around the 2202-5 and 2202-6 racks, that the vibration of drilling the wedge anchor holes in the concrete floor was enough to trip the switches and initiate a SCRAM.

Other plants have reported vibrational induced tripping of these Barksdale pressure switches when the associated pressurized sensing lines were lightly tapped.

The setpoints of the pressure switches were not a contributing factor. At the time of the event, the setpoints were above the indicated pressure. The IMD technicians tested the PS's after the SCRAM and found out that these PS setpoints were at 1046 psig which are conservatively lower than the tolerance range of 1050 - 1060 psig. The Unit Two reactor pressure was at 1005 psig during the SCRAM as verified by results from a Control Room Recorder and computer points.

D. SAFETY ANALYSIS OF EVENT:

The safety consequences of this event were minimal. The SCRAM, along with the PCI Group II and III isolations, the Control Room and Reactor Building ventilation isolations, and the SBGT autostart performed as designed. The Reactor Vessel pressure and water level were controlled, and the SCRAM recovery was satisfactory.

E. CORRECTIVE ACTIONS:

The immediate corrective action included controlling the Reactor Vessel pressure and water level per QGA 100 and performing the SCRAM recovery per QCGP 2-3.

The Shift Engineer dispatched an EA to check the instrument racks 2202-5 and 6, where the contractors were working on a modification.

The modification work was stopped pending further investigation.

The Station Engineering and Construction (ENC) work control program was reviewed. The work control program consists of four steps:

1) Prejob Walkdowns - The complete job is walked down prior to the start with Operations involvement. Operations approves the job start, and identifies any precautions as needed.

2) Scaffolding Requests - If scaffolding is required, Operations surveys the working area prior to approval of the scaffolding request. Precautions concerning the erection of the scaffolding and the work to be performed are identified and documented on the scaffolding authorization document.

3) Shift Briefing - Each Operating crew is briefed and work approval is requested whenever work is to be performed on that shift. Precautions are stressed by Operations at this stage.

4) Job Overview - Each contracting crew is supervised by its own foreman, who is situated near the working area at all times. The foreman may be responsible for more than one crew, depending on the working conditions. Each working crew is overviewed daily by its General Foreman and cognizant ENC field engineer. If a situation arises which requires additional verification or supervision, contractor Quality Control personnel, ENC field engineers, and Operating personnel are available.

Based on personal inverviews, the contractors were fully aware of all sensitive equipment and their associated sensing lines. The precautions were stressed daily during shift briefing by Operations. There were no work control program deficiencies found.

This event will be tailgated with groups performing work in the plant to enhance personnel awareness of the sensitivity of the reactor high pressure switches and their associated sensing lines (NTS #2652009300901 thru 08).

F. PREVIOUS EVENTS:

There have been four Licensee Event Reports (LER) in the past five years due to contractors personnel errors.

1) LER 88-007 A Stand-by Gas Treatment suction from the Reactor Building blocked from plastic draped over the intake bell.

2) LER 88-010 Drywell atmosphere thermocouple splices found not environmentally qualified.

3) LER 90-035 "A" Core Spray room floor drain plug removed.

4) LER 92-010 Load drop from loss of 1A recirculation pump after losing Unit One 125 Vdc Turbine Building buses 1A and 1A-2.

G. <u>COMPONENT FAILURE DATA</u>:

There was no component failure involved in this event.