LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
QUAD-CITIES, NUCLEAR POWER STATION, UNIT One	0 5 0 0 0 2 5 4	1 of 0 5

Title (4) Isolation Of Instrument Root Valves 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 3	1 8	8 6	8 6	--- 0 1 7	--- 0 0	0 4	1 0	8 6		0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 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)	x 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	
Mark Kooi, Compliance Coord. Ext. 2242	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)

On March 18, 1986, Unit One was in the REFUEL mode during a refueling and maintenance outage. At 1349 hours an Anticipated Transient Without Scram (ATWS) trip was received from a Division I reactor low water signal. The trip was reset at 1404 hours. At 1425 hours a Channel "B" Reactor Protection System (RPS) trip occurred from reactor low water level signal. It was then observed that the reactor water level indicator 1-263-100A did not agree with other control room level indication. While backfilling the level instrument lines trying to correct the problem, a second ATWS trip was received at 1715 hours. At 1900 hours it was discovered that instrument root valves 1-263-2-12A and 14A at drywell penetration X-49 were isolated. These root valves isolated reactor variable leg and reference leg instrument lines that feed instrumentation on the 2201-5 rack. It was not known immediately when the valves were closed. An investigation determined that the valves were closed between 0300 and 0330 hours on 3-18-86. A valve checklist was in progress at that time and it is believed the root valves were inadvertently closed during performance of the checklist. The cause of the initial ATWS trip and Channel "B" RPS trip is believed to be the result of the isolated instrument lines. Corrective action was to inform operators of importance of proper positioning of root valves and to reverify valve positions of all critical instrumentation prior to startup. This report is submitted in accordance with the requirements of 10 CFR 50.73(a)(2)(ii) and (a)(2)(iv).

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PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2511 MWt rated core thermal power. Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

IDENTIFICATION OF OCCURRENCE:

Isolation of instrument line root valves due to personnel error in performance of valve checklist.

Discovery Date: 3-18-86

Report Date: 04-10-86

This report was initiated by Deviation Report D-4-1-86-36

CONDITIONS PRIOR TO OCCURRENCE:

REFUEL Mode(2) - Rx Power 00% - Unit Load 000 MWe

REFUEL Mode(2) - Refuel - In this position interlocks are established so that one control rod only may be withdrawn when flux amplifiers are set at the proper sensitivity level and the refueling crane is not over the reactor. Also, the trip from the turbine control valves, turbine stop valves, main steam isolation valves, and condenser vacuum are bypassed. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

DESCRIPTION OF OCCURRENCE:

On March 18, 1986, Unit One was in the REFUEL mode during a refueling and maintenance outage. At 1349 hours an Anticipated Transient Without Scram (ATWS) [JC] trip was received which initiated a reactor scram from Alternate Rod Insertion (ARI). The reactor recirculation pumps [AD] were not operating and thus did not trip. THe ATWS trip was initiated by a Division I reactor low water level signal. Indicated reactor water level appeared normal and the ATWS trip was reset at 1404 hours.

At 1425 hours a channel "B" Reactor Protection System [JC] trip was received from a reactor low water level signal. When the reactor water level was raised to clear the half scram, it was observed that there was a discrepancy between the reactor level indications in the control room. Level Indication 1-263-100A was indicating approximately +20 inches. Level Indicator 1-263-100B and the GE/MAC Level Indicators 1-640-29A and B were indicating approximately +55 inches. An investigation was begun to determine the cause for the level indication discrepancy.

At 1715 hours another ATWS trip was received which initiated ARI and tripped both reactor recirculation pumps. The cause of this trip was due to Instrument Maintenance personnel backfilling the instrumentation on the 2201-5 rack while in the process of troubleshooting the reactor level indication. The trip occurred when the reactor level transmitter for the ATWS system was valved in after backfilling. The ATWS trip was reset at 1730 hours.

At 1900 hours it was discovered that the instrument root isolation valves 1-263-2-12A and 1-263-2-14A at drywell penetration X-49 were closed. These valves isolate reactor reference leg and variable leg sensing lines. The sensing lines feed instrumentation on the 2201-5 instrument rack, among which is the level transmitter for Level Indicator 1-263-100A. The root valves were opened and the "A" loop level indication was restored. The other instrumentation that was affected by the root valve closure included instruments that provide trips and interlocks for various Engineered Safety Features.

This report is being submitted in accordance with the requirements of 10 CFR 50.73(a)(2)(ii) and (a)(2)(iv), which require the reporting of any event or condition that resulted in a nuclear power plant being in an unanalyzed condition or that resulted in the actuation of any Engineered Safety Feature.

APPARENT CAUSE OF OCCURRENCE:

An investigation was initiated to determine the cause of the root valve closure and the length of time the valves were closed. Traces of the reactor vessel level instrument readings were obtained from the stored process computer data for the period prior to the discovery of the closed valves. The level data from isolated level instrument 1-263-100A was compared to that from level instruments 1-263-100B and 1-640-29A and B. Examination of the level traces revealed that all of the level instruments tracked reactor level together until between 0300 and 0330 hours on 3-18-86. At this time the 1-263-100A level indicator stopped following reactor level with the other instruments. It is believed that the root valves were closed during this time period.

A check of station activities in progress at that time disclosed that the only related activity was the performance of the Integrated Primary Containment Leak Rate Test Valve Lineup Checklist, QTS 150-S2. The valves in question were on the checklist and had been verified as being in the open position on 3-18-86. The non-licensed operator who performed that portion of the checklist was interviewed. The operator stated that he had checked the valves in question at approximately 0300 hours on 3-18-86. The checklist required the valves to be in the open position. The operator indicated that he believed he had properly checked the valves to be in the open position.

Root valves 1-263-2-12A and 14A are located on drywell penetration X-49, which is approximately 17 feet above the second floor of the reactor building. A ladder is required to reach the penetration. The penetration contains six instrument lines arranged in a circular pattern. The stems for root valves 1-263-2-12A and 14A point downward. It is believed that the operator possibly became confused about the direction of rotation of the valves and inadvertently closed them while checking the stem rotation in an attempt to verify that they were open.

The initial ATWS trip and the Channel B RPS trip on reactor low water level are believed to have resulted from the instrument lines being isolated. Instrument maintenance personnel were performing QIS 5, "High Reactor Pressure Scram Surveillances" at the time the trips, occurred. This surveillance involves calibration of pressure switches that are connected to the sensing line isolated by the 1-263-2-12A valve. This sensing line also serves as the reference leg for two of the reactor low level scram switches and the Division I ATWS level transmitters. The manipulations of the pressure switch isolation valves on the isolated sensing lines are believed to have caused the ATWS and scram switches to trip.

ANALYSIS OF OCCURRENCE:

The instruments affected by the isolated sensing lines are listed below.

Instrument Equipment Piece Number	Function
Level Indicating Switch 1-263-72A & C	ECCS Initiation; Diesel Generator Start; HPCI and RCIC Turbine Trips
Level Transmitter 1-263-57A & B	Input Signal to Level Indicating Switch 1-263-140A and B and Level Switch 1-263-143A and B
Level Switch 1-263-143A & B 1-263-140A & B	Standby Gas Treatment System Initiation; Reactor Recirc Pump Trip Reactor Low Level Scram; Group I, II, III Isolations
Pressure Switch 1-263-55A & B	Reactor High Pressure Scram
Pressure Indicator 1-263-60A	Local Indication
Pressure Controller 1-203-3A, 3B, 3C, 3D, 3E	Main Steam Relief Valve Pressure Signals
Level Indicator 1-263-101	Control Room Upper 400 Level Indication
Level Indicator 1-1360-28	RCIC Local Reactor Level Indicator
Level Indicating Transmitter Switch 1-263-59A	Main Turbine Trip; Reactor Feed Pump Trip; Feedwater Pump Runout Reset; Control Room Indication (1-263-100A)
Pressure Transmitter 1-263-20A & C	ATWS Reactor Recirc Pump Trip and ARI
Level Transmitter 1-263-23A & C	ATWS Reactor Recirc Pump Trip and ARI
Pressure Switch 1-263-52A	ECCS Pump Reactor Low Pressure Permissive

The root valves were closed for approximately 16 hours. Unit One was shutdown during this time period. The Emergency Core Cooling Systems were not required to be operable since no activity was in progress that had the potential to drain the vessel. Primary containment integrity was not required. Valve checklist QTS 150-S2 was being performed to verify that all valves were in the proper positions following the normal refueling and maintenance outage activities. A second verification of this checklist was to have been performed prior to the Integrated Primary Containment Leak Rate Test. This verification would have detected the valving error. Additional operating valve checklists are performed prior to startup that would have also discovered the error.

This activity is only performed during outages and thus would not have occurred during reactor operation. Thus the safety consequences of this event are minimal.

CORRECTIVE ACTION:

All operating shifts were informed about the incident and the importance of correct positioning of instrument root valves was emphasized. For the remainder of the Unit One outage, all valve checklists were reviewed by licensed management personnel at the end of the shift on which the checklists were performed and any discrepancies were resolved. After all outage work was completed, the root valves, instrument rack stop valves and instrument isolation valves on all critical instrumentation were verified to be in the correct positions prior to unit startup.

FAILURE DATA:

None.

DEVIATION REPORT

Commonwealth Edison

DVR NO. 4 - 1 - 86 - 36 (STA UNIT YEAR NO)

PART 1 TITLE OF DEVIATION: Rx VESSEL LEVEL INSTRUMENTS ISOLATED

OCCURRED: 1425 (DATE) 3-18-86 (TIME)

SYSTEM AFFECTED: 263

PLANT STATUS AT TIME OF EVENT: MODE S/D, POWER(%) --

WORK REQUEST NO.

TESTING: [ ] YES [X] NO

DESCRIPTION OF EVENT

AT 1425 A 1/2 SCRAM WAS RECEIVED FROM LOW Rx VESSEL WATER LEVEL. IT WAS NOTICED THAT THE A LOOP ROSEMOUNT LEVEL INDICATOR WAS AT APPROX. +10". THE B LOOP ROSEMOUNT AND BOTH LOOPS OF THE GEMAC INDICATORS WERE AT APPROX. 50". WATER LEVEL WAS RAISED, AND THE 1/2 SCRAM RESET. AT 1900 THE DW PENETRATION ISOLATION VALVES FOR THE A LOOP YARWAY/ROSEMOUNT LEVEL INSTRUMENTS REFERENCE LEG AND VARIABLE LEG (1-263-2-12A & 14A) WERE FOUND CLOSED. THEY WERE OPENED AND THE LINES FILLED.

POTENTIALLY SIGNIFICANT EVENT PER NSD DIRECTIVE A-07 [X] YES [ ] NO

10CFR50.72 NRC RED PHONE NOTIFICATION MADE [ ] 1 HOUR [X] 4 HOUR 2220 (TIME) [ ] NO

ANTHONY FUHS — RESPONSIBLE SUPERVISOR — 3-18-86 (DATE)

PART 2 OPERATING ENGINEER'S COMMENTS

[ ] NON REPORTABLE EVENT

[X] 30 DAY REPORTABLE/10CFR 50.73 (a) (2) (ii)

[ ] 5 DAY REPORT PER 10CFR21

[ ] ANNUAL/SPECIAL REPORT REQUIRED

A.L.R. #

L.E.R. # 86-17

NOTIFICATION: REGION III — DATE — TIME

NSD — DATE — TIME

[ ] CECO CORPORATE NOTIFICATION MADE IF ABOVE NOTIFICATION IS PER 10CFR21

TELECOPY: CECO CORPORATE OFFICER — DATE — TIME

PRELIMINARY REPORT COMPLETED AND REVIEWED: J. SWALES — OPERATING ENGINEER — 3-19-86 (DATE)

INVESTIGATION REPORT & RESOLUTION ACCEPTED BY STATION REVIEW: M Kroi; R.A. Robey; J. Swales

RESOLUTION APPROVED AND AUTHORIZED FOR DISTRIBUTION: Richard F. Bax — STATION MANAGER — 4-15-86 (DATE)

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86-5176 (Form 15-52-1) 11/85

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

RLB-86-13

April 10, 1986

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

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

Enclosed please find Licensee Event Report (LER) 86-017, Revision 00, for Quad-Cities Nuclear Power Station.

This report is submitted to you in accordance with the requirements of the Code of Federal Regulations, Title 10, Part(s) 50.73(a)(2)(ii) and (a)(2)(iv), which require the reporting of any event or condition that resulted in a nuclear power plant being in an unanalyzed condition or that resulted in actuation of any Engineered Safety Feature.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

R. L. Bax
Station Manager

RLB/MSK/dak

Enclosure

cc: J. Wojnarowski
A. Madison
INPO Records Center
NRC Region III