Commonwealth Edison
Quad-Cities Nuclear Power Station
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NJK-76-420

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U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

November 12, 1976

J. Keppler, Regional Director
Office of Inspection and Enforcement
Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Reference: Quad-Cities Nuclear Power Station
Docket No. 50-254, DPR-29, Unit 1
Appendix A, Sections 3.5.D.3 and 6.6.B.1.e.

Enclosed please find Reportable Occurrence Report No. RO 50-254/76-34 for Quad-Cities Nuclear Power Station. This occurrence was previously reported to Region III, Office of Inspection and Enforcement by telephone on November 1, 1976 and by telecopy on November 2, 1976.

This report is submitted to you in accordance with the requirements of Technical Specification 6.6.B.1.

Very truly yours,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

R. E. Queno
for
N.J. Kalivianakis
Station Superintendent

NJK/LFG/lk

cc: G.A. Abrell

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PDR ADOCK 05000254
S PDR

NOV 18 1976

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1–6)

(PLEASE PRINT ALL REQUIRED INFORMATION)

	LICENSEE NAME (9–14)	LICENSE NUMBER (15–25)	LICENSE TYPE (26–30)	EVENT TYPE (31–32)
01	I L Q A D 1	0 0 – 0 0 0 0 0 – 0 0	4 1 1 1 1	0 1

	CATEGORY (57–58)	REPORT TYPE (59)	REPORT SOURCE (60)	DOCKET NUMBER (61–68)	EVENT DATE (69–74)	REPORT DATE (75–80)
01 CON'T		T	L	0 5 0 – 0 2 5 4	1 1 0 1 7 6	1 1 1 2 7 6

EVENT DESCRIPTION

02 During the startup for Unit Two, Cycle Three, the electromatic relief valves were
03 tested for operability. Two of the relief valves failed to open. Due to these
04 problems on Unit Two, it was decided to test the Unit One electromatic relief valves.
05 At 2:45 pm on November 1, 1976, the Unit One relief valves were tested. The 1-203-3C
06 and 1-203-3E relief valves failed to open when actuated from the control room. It was

	SYSTEM CODE (9–10)	CAUSE CODE (11)	COMPONENT CODE (12–17)	PRIME COMPONENT SUPPLIER (43)	COMPONENT MANUFACTURER (44–47)	VIOLATION (48)
07	S F	E	V A L V E X	N	D 2 4 3	N

CAUSE DESCRIPTION

08 (Proximate Cause-Equipment Failure) These valves are designed to operate by venting
09 the area below the valve disc by opening a pilot valve, and causing a differential
10 pressure across the valve disc. This results in the valve being forced (see attached)

	FACILITY STATUS (9)	% POWER (10–12)	OTHER STATUS (13–44)	METHOD OF DISCOVERY (45)	DISCOVERY DESCRIPTION (46–80)
11	E	0 7 7	NA	C	Test based on Unit 2 valve failures

	FORM OF ACTIVITY RELEASED (9)	CONTENT OF RELEASE (10)	AMOUNT OF ACTIVITY (11–44)	LOCATION OF RELEASE (45–80)
12	Z	Z	NA	NA

PERSONNEL EXPOSURES

	NUMBER (9–11)	TYPE (12)	DESCRIPTION (13–80)
13	0 0 0	Z	NA

PERSONNEL INJURIES

	NUMBER (9–11)	DESCRIPTION (12–80)
14	0 0 0	NA

OFFSITE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

	TYPE (9)	DESCRIPTION (10–80)
16	Z	NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 (Event Description contd) determined that the pilot valves were functioning as
19 evidenced by a temperature rise down stream of the pilot valve. However, (cont)

NAME: James Schrock PHONE: 309-654-2241 Ext. 246

CAUSE DESCRIPTION contd

open by reactor steam pressure. There is an orifice through the disc retainer that allows the area below the valve disc to re-pressurize and close the valve when the pilot valve is closed. The apparent cause of the valve failures was that there was excessive steam leakage into the area below the valve disc, in addition to the normal steam flow to this area through the orfice. As a result of this additional steam flow, the pilot valve could not adequately vent the area below the valve disc to allow the valve to open. There are two possible leakage paths that could have resulted in the valve failures. The steam could have leaked between the valve disc guide and the piston rings on the valve disc or there could be leakage past the threads on the disc retainer.

ADDITIONAL FACTORS

EVENT DESCRIPTION contd

as described below, there was no noticeable steam flow from the reactor through the valve discharge piping.

The presence of steam flow was checked by first lowering the setpoint on the turbine Load Set, in order to open a main steam bypass valve. Each relief valve was actuated from the control room for a one-second time interval. Relief valve opening and subsequent steam flow are then verified by means of observing closure of the bypass valve during the time the relief valve is open.

This response of the bypass system was satisfactorily verified for relief valves 1-203-3A, 1-203-3B, and 1-203-3D.

In accordance with Technical Specification 3.5.D.3, an orderly shutdown was commenced immediately and load was decreased at the rate of 50 MWe per hour. The unit was in the Cold Shutdown Condition by 8:00am on November 2, 1976. Work Requests 4094-76 and 4095-76 were issued to determine the source of the problem and perform repairs.

The failure of two electromatic relief valves renders the Auto-blowdown function of the Emergency Core Cooling System inoperable. However, the High Pressure Coolant Injection (HPCI) Sub-system was demonstrated to be fully operable, thereby providing a means available for introducing emergency core cooling water into the reactor vessel at operating pressure. Another function of the electromatic relief valves is to protect the vessel from over-pressurization. This function was provided for by virtue of the three operable relief valves providing a path for blowdown to the suppression chamber. The Target Rock Safety-Relief Valve 1-203-3A was demonstrated to be operable, and would have lifted at a reactor pressure of 1125 psig. Valves 1-203-3B and 1-203-3D would have opened at reactor pressures of 1130 psig and 1135 psig, respectively. Therefore, there was overpressure protection at all three pressure setpoints specified in Technical Specification 4.6.E. Also, the Electro-hydraulic Control (EHC) system was operable,

enabling the bypass valves to dump steam to the condenser in the event of a turbine trip, thereby controlling reactor pressure. Furthermore, all nine Main Steam Safety Valves were operable, and were fully capable of preventing the reactor vessel pressure from reaching the Safety Limit of 1325 psig at the vessel steam space. Therefore, the possible consequences of this occurrence were minimized by the redundant design of the safety systems and the fact that all other safety systems were operable. At no time was the public health and safety in jeopardy, nor was the ability to safely shutdown the reactor compromised. (RO 50-254/76-34)

Corrective Action to Prevent Recurrence

When Unit One was shut down and de-inerted, a drywell entry was made. The two elecromatic relief valves that failed were removed, brought to the shop, disassembled, and inspected. The disc retainer lock arm and lock screw was missing from the 1-203-3E relief valve. It was decided to replace the 1-203-3E valve, serial number 7069, with a spare relief valve, serial number 7062, and finish inspecting the 7069 valve at a later date. The spare relief valve had been overhauled and had new piston rings, valve disc, and pilot valve disc installed. The 1-203-3C relief valve, serial number 7063, was overhauled and the piston rings and valve disc guide were replaced. Both valves were then given a leak test and operability test in the shop and then were replaced on the main steam lines. Startup was then commenced on Unit One on November 6, 1976. The reactor was brought to operating pressure and the 1-203-3C and 1-203-3E relief valves were tested using Temporary Procedure No. 743. This procedure constitutes a revision to the existing procedure for manual operation of the electromatic relief valves, and calls for verification of a bypass valve closure response as well as temperature, to verify steam flow from a relief valve. This revision will be a permanent procedure, and shall be implemented accordingly. After the electromatic reliefs were replaced on the Unit One steam lines the 7069 relief inspection was completed and the threads on the disc retainer were found to be worn.

Investigations into this problem are being continued between the valve manufacturer and Commonwealth Edison Company to resolve the causes of these valve failures.

Failure Data

Unit One Main Steam Relief Valves had not experienced this type of failure in the past. Modification M-4-1-73-45 was installed on the relief valves in May, 1973. This modification installed a valve disc retainer locking device, to prevent the locking plate from falling off. This change had been performed based on an electromatic relief valve failure at Oyster Creek Station in December, 1972.

The 1-203-3C and 1-203-3E electromatic relief valves are manufactured by Dresser Corporation, serial numbers BK-7063 and BK-7069.