Exelon Generation
Quad Cities Generating Station
22710 206th Avenue North
Cordova, IL 61242-9740
Tel 309-654-2241

www.exeloncorp.com

Exelon Nuclear

June 12, 2003

SVP-03-075

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

Quad Cities Nuclear Power Station, Unit 2
Facility Operating License No. DPR-30
NRC Docket No. 50-265

Subject: Licensee Event Report 265/03-002, "Self-Actuation of Main Steam Relief Valve due to Excessive Leakage Through Pilot Valve Seat."

Enclosed is Licensee Event Report (LER) 265/03-002, "Self-Actuation of Main Steam Relief Valve due to Excessive Leakage Through Pilot Valve Seat," 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)(i)(A), which requires reporting of the completion of any nuclear plant shutdown required by the plant's Technical Specifications, and Part 50.73 (a)(2)(iv)(A), which requires the reporting of any event or condition that resulted in manual or automatic actuation of a general containment isolation signal.

Should you have any questions concerning this report, please contact Mr. W. J. Beck at (309) 227-2800.

Respectfully,

Timothy J. Tulon
Site Vice President
Quad Cities Nuclear Power Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

IE22

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER)		Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Quad Cities Nuclear Power Station Unit 2	05000265	1 of 4

4. TITLE Self-Actuation of Main Steam Relief Valve due to Excessive Leakage Through Pilot Valve Seat

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER N/A
04	16	03	03	- 002 -	00	06	12	03	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)	X	50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Wally Beck, Regulatory Assurance Manager	(309) 227-2800

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	SB	RV	T020	Y					

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On April 16, 2003, at 1322 hours, the Unit 2 3B Main Steam Relief Valve (RV) self-actuated. The operators initiated the appropriate procedures for an open RV, including initiation of suppression pool cooling, a manual reactor scram, removal of the control power fuses to the 3B RV, and closure of the main steam isolation valves to slow the reactor cooldown rate. At 1359 hours, the suppression pool temperature reached 110F and an Alert was declared in accordance with the Exelon Emergency Plan. The Alert was exited at 2251 hours.

The safety significance of this event was minimal. The opening of the 3B RV did not affect the capability of the relief valves to protect against high reactor pressure. At the time of the event, the ½ Emergency Diesel Generator was out of service for planned maintenance. All other mitigating systems were available. Therefore, both emergency and non-emergency sources of injection to the vessel were available to make up for the coolant being relieved through the RV to the suppression pool.

The root cause was excessive leakage past the 3B RV pilot valve seat, which allowed the RV closure force to diminish and the RV to open. Corrective actions included replacement of the 3B RV, improvements in RV tailpipe temperature monitoring, and removal of the requirement to perform on-line test actuations of the RV.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor, 2957 Megawatts Thermal Rated Core Power

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

EVENT IDENTIFICATION

Self-Actuation of Main Steam Relief Valve due to Excessive Leakage Through Pilot Valve Seat

A. CONDITION PRIOR TO EVENT

Unit: 2	Event Date: April 16, 2003	Event Time: 1322 hours
Reactor Mode: 1	Mode Name: Power Operation	Power Level: 100%

Power Operation (1) - Mode switch in the RUN position with average reactor coolant temperature at any temperature.

B. DESCRIPTION OF EVENT

On April 16, 2003, at 1322 hours and with Unit 2 at 100% power, alarms associated with the Unit 2 3B Main Steam Relief Valve (RV) [RV] being open were received in the main control room. No activities involving the RVs were in progress at that time. The operators initiated the appropriate procedures for an open RV. At 1330 hours, suppression pool cooling [BO] was initiated. At 1337 hours, when the suppression pool temperature exceeded 95F, the operators manually scrammed the reactor. At 1340 hours the operators removed the control power fuses to the 3B RV, with no observed effect. At 1344 hours, a Primary Containment isolation signal was received due to low reactor water level. At 1352, the operators closed the main steam isolation valves [ISV] to slow the reactor cooldown rate.

At 1359 hours, the suppression pool temperature reached 110F and an Alert was declared in accordance with the Exelon Emergency Plan. At 1411 hours, the reactor cooldown rate exceeded the Technical Specification limit of 100F/hour.

At 2106 hours, with reactor pressure at approximately 50 psig, the 3B RV indicated closed. At 2237 hours reactor coolant was below 212F and Mode 4, Cold Shutdown, was entered. At 2251 hours the Alert was exited.

C. CAUSE OF EVENT

The root cause of this event was excessive leakage past the 3B RV pilot valve seat, which allowed the RV closure force to diminish and the RV to fully open. The excessive leakage was due to erosion of the pilot valve seating surface as a result of an unknown pilot valve seat flaw. Possible causes of the seat flaw include on-line steam test actuations and very small initial leakage following refurbishment of the pilot valve seat.

Contributing causes included a system monitoring plan that contained performance limits that failed to preclude inadvertent operation of the valve, and a RV design that is susceptible to a self-actuation due to a small amount of pilot valve leakage.

D. SAFETY ANALYSIS

The safety significance of this event was minimal. The spurious opening of the 3B RV did not affect the capability of the relief valves to protect against high reactor pressure. At the time of the event, the ½ Emergency Diesel Generator was out of service for planned maintenance. All other mitigating systems were available. Therefore, both emergency and non-emergency sources of injection to the vessel were available to make up for the coolant being relieved through the RV to the suppression pool.

The Probabilistic Risk Assessment of this event performed by Exelon risk personnel determined that the calculated conditional core damage probability was approximately 3E-7, which is below the level for very low risk (i.e., 1E-6).

E. CORRECTIVE ACTIONS

Corrective Actions Completed:

The failed Unit 2 3B RV was replaced.

The Unit 2 3B RV tailpipe thermocouple was relocated to the optimum location.

An analysis was performed to determine that the reactor coolant system was acceptable for continued operation following the cooldown rate of greater than 100F/hr.

Approval of a Technical Specification amendment to delete the requirement to actuate relief valves on-line has been obtained.

Relief from the ASME/OM Code requirement to lift relief valves at reduced pressure following maintenance has been obtained.

Corrective Actions to be Completed:

The relief valve post-maintenance (refurbishment) test requirements will be revised to include testing at a steam test facility at a nominal 1000 psi, with an acceptance criteria of a nominal zero lb/hr, in accordance with the vendor technical manual.

The System Engineering Main Steam monitoring plan will be verified or revised to include the appropriate tailpipe temperature monitoring frequency and action thresholds for relief valve tailpipe high temperatures.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Quad Cities Nuclear Power Station Unit 2	05000265	2003	002	00	4 of 4

(If more space is required, use additional copies of NRC Form 366A)(17)

QCOS 0203-02, "Safety And Relief Valve Temperature Surveillance," will be revised to ensure a Condition Report is initiated at a conservative tailpipe temperature to determine continued operation of a degraded relief valve.

F. PREVIOUS OCCURRENCES

No events were identified at Quad Cities Station involving self-actuation of a main steam relief valve.

G. COMPONENT FAILURE DATA

The Unit 2 3B main steam relief valve is a Model 93V-001 Power Operated Relief Valve manufactured by Target Rock.