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June 18, 2004

SVP-04-061

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 265/04-004, "Main Steam Safety/Relief Valve As-Found Setpoint Outside of Technical Specification Allowed Value Due to Vibration"

Enclosed is Licensee Event Report (LER) 265/04-004, "Main Steam Safety/Relief Valve As-Found Setpoint Outside of Technical Specification Allowed Value Due to Vibration," for Quad Cities Nuclear Power Station, Unit 2.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(i)(B), which requires reporting of any operation or condition that was prohibited by the plant's Technical Specifications, and Part 50.73(a)(2)(v)(A), which requires the reporting of any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shut down the reactor and maintain it in a safe shutdown condition.

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

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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 3

4. TITLE Main Steam Safety/Relief Valve As-Found Setpoint Outside of Technical Specification Allowed Value Due to Vibration

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	DOCKET NUMBER
									N/A	N/A
04	19	2004	2004	– 004 –	00	06	18	2004	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE	1
10. POWER LEVEL	088

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)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)	X	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)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	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
XM4	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 19, 2004, during as-found testing of the Unit 2 Main Steam Safety Relief Valve (S/RV), it was identified that the S/RV had an as-found pressure setpoint 6.8% above nameplate, which exceeded the Technical Specification allowed value (nameplate +/- 1%) and the ASME Code requirements (nameplate +/- 3%). During disassembly of the pilot stages of the valve, a 0.008" groove in the bellows cap assembly was identified.

The root cause of the S/RV degraded condition is vibration that caused material to be removed from the bellows cap assembly, causing the partial capture of a coil of the pressure adjusting spring, resulting in extra force (pressure) being required to actuate the safety portion of the S/RV. This vibration is related to operation at Extended Power Uprate (EPU) levels.

Corrective Actions include limiting operation of the units to pre-EPU levels, and modification of the S/RV based on the results of further vibration testing.

The Anticipated Transient Without Scram (ATWS), ASME overpressure and fuel thermal analyses applicable to Unit 2 Cycle 17 were re-evaluated as satisfactory using as-found safety valve data. It was determined that the event duration and torus early heat load for an Appendix R event may be increased due to operation of the S/RV at a higher reactor pressure.

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

Main Steam Safety/Relief Valve As-Found Setpoint Outside of Technical Specification Allowed Value Due to Vibration

A. CONDITION PRIOR TO EVENT

Unit: 2	Event Date: April 19, 2004	Event Time: 1500 hours
Reactor Mode: 1	Mode Name: Power Operation	Power Level: 088%

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

B. DESCRIPTION OF EVENT

During a Unit 2 refueling outage in the Spring of 2004 (Q2R17), the Main Steam Safety Relief Valve (S/RV) [SB] [RV] was removed and shipped to Wyle Laboratories Nuclear Services S/RV Facility in Huntsville, AL, for as-found testing. On April 19, 2004, this testing identified that the S/RV had an as-found pressure setpoint that exceeded the Technical Specification allowed value (nameplate +/- 1%) and the ASME Code requirements (nameplate +/- 3%). The testing identified an as-found lift pressure of 1213 psig against a nameplate setpoint of 1135 psig; i.e., a lift pressure 6.8% higher than nameplate.

Quad Cities Nuclear Power Station personnel were present at Wyle Laboratories to supervise and document disassembly and inspection of the S/RV, which took place on May 11, 2004, and May 12, 2004. During disassembly of the pilot stages of the valve, subcomponent degradation (0.008" groove in bellows cap assembly) was identified. This groove would have partially captured the pressure adjusting spring, resulting in the need for additional force (pressure) to open the first stage pilot.

C. CAUSE OF EVENT

The root cause of the S/RV degraded condition is vibration that caused material to be removed from the bellows cap assembly, causing the partial capture of a coil of the pressure adjusting spring, resulting in extra force (pressure) being required to actuate the safety portion of the S/RV. This vibration is related to operation at Extended Power Uprate (EPU) levels, based on the fact that this failure has not been experienced in the past and does not appear to be experienced on a sample of similar valves and a valve from pre-EPU operation at Quad Cities Nuclear Power Station, and on the fact that there is a known increase in vibration magnitudes at EPU operating levels.

D. SAFETY ANALYSIS

The safety significance of this event was minimal. The Anticipated Transient Without Scram (ATWS), ASME overpressure and fuel thermal analyses applicable to Unit 2 Cycle 17 were re-evaluated using revised inputs that bound the as-found Main Steam Safety Valve (MSSV) data. The as-found values were used for the S/RV previously discussed and for one other safety valve that had also exceeded the Technical Specification allowed value. For the other MSSVs, the Technical Specification limit (nameplate +1%) was used. The acceptance criteria for both the ATWS and ASME overpressure analyses were met and the applicability of the established Cycle 17 fuel thermal limits was confirmed. Additionally, the impact of the S/RV lifting at a higher pressure on the Appendix R analysis was examined. It was determined that the duration of the event may have increased and the heat load added to the torus early in the event may be slightly higher as a result of the S/RV cycling at a higher reactor pressure. Therefore, this event is considered a Safety System Functional Failure (SSFF) because of the possibility that the Appendix R analysis criteria would not be met. In addition to 10 CFR 50.73(a)(2)(v)(A) associated with SSFFs, this Licensee Event Report is also being submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) as operation of the plant in a condition prohibited by Technical Specifications.

E. CORRECTIVE ACTIONS

Immediate Actions

Controls to limit Unit 1 and Unit 2 reactor power to pre-EPU levels for any significant duration have been established. Short duration power increases above pre-EPU power levels for data acquisition purposes are allowed.

Corrective Actions to be Completed:

Further vibration testing and evaluation of the S/RV will be performed and any appropriate modifications made.

F. PREVIOUS OCCURRENCES

No previous examples of this type of wear failure of an S/RV were identified. Previous examples of an S/RV as-found setpoint being outside the Technical Specification allowed value at Quad Cities Nuclear Power Station were identified in 1997 and 1999. Both examples were attributed to setpoint drift.

G. COMPONENT FAILURE DATA

The S/RV is a Model 7467F Safety/Relief Valve manufactured by Target Rock.