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July 25, 2005

SVP-05-050

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

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

Subject: Licensee Event Report 265/04-004, Revision 1, "Main Steam Safety/Relief Valve As-Found Setpoint Outside of Technical Specification Allowed Value Due to Pressure Adjusting Spring Tolerances, Pressure Cap Material and Vibration"

Enclosed is Licensee Event Report (LER) 265/04-004, Revision 1, "Main Steam Safety/Relief Valve As-Found Setpoint Outside of Technical Specification Allowed Value Due to Pressure Adjusting Spring Tolerances, Pressure Cap Material and Vibration," for Quad Cities Nuclear Power Station, Unit 2.

This revised 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. This report was revised to expand the description of previous events, revise the root cause, and remove the reference to a condition that could have prevented the fulfillment of a safety function.

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

Respectfully,

for

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

LICENSEE EVENT REPORT (LER)

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

4. TITLE Main Steam Safety/Relief Valve As-Found Setpoint Outside of Technical Specification Allowed Value Due to Pressure Adjusting Spring Tolerances, Pressure Cap Material and 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 N/A	DOCKET NUMBER N/A
04	19	2004	2004	– 004 –	01	07	25	2005	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)		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 insufficient pressure adjusting spring tolerance specifications in combination with the use of a bellows cap material that did not have sufficient hardness.

Original corrective actions included further vibration testing. As a result of the vibration testing, the S/RVs were replaced with S/RVs with improved subcomponents.

The Appendix R event, 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.

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 Pressure Adjusting Spring Tolerances, Pressure Cap Material and 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.

Subsequent shaker table testing was performed at Wyle Labs to identify changes to S/RV subcomponents to improve performance. Original testing scope focused on caps manufactured from more wear-resistant materials. Early test results, as well as data points on Dresden valves having similar wear in pre-EPU conditions, caused focus to include the pressure adjusting spring tolerances. The original failure condition was simulated in testing utilizing the in-plant spring and a new stock bellows assembly cap. A successful 24-hour aging run was performed utilizing a pressure adjusting spring with improved straightness and perpendicularity tolerances, and a cap with an electrolyzed coating at a thickness of 0.0015". This final configuration resolved the original wear issue.

C. CAUSE OF EVENT

The root cause of the S/RV degraded condition is insufficient pressure adjusting spring tolerance specifications in combination with the use of a bellows cap material that did not have sufficient hardness, which resulted in higher contact forces on the bellows assembly cap. A required contributing factor included the typical vibration of the main steam system piping. This resulted in fretting wear to the bellows assembly cap, which led to set pressure drift. Although operation of the plant at EPU power levels has resulted in an increase in main steam system piping vibration, it was not the root cause of the fretting wear to the bellows assembly cap.

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 increase in S/RV lift pressure would not cause a significant change in the time to reach top of active fuel or in torus water temperature response, and that the Appendix R analysis remains valid. Therefore, this event is not considered a Safety System Functional Failure (SSFF). This Licensee Event Report is 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

Original corrective actions included limiting operation of the units to pre-EPU levels and further vibration testing. Based on the results of the vibration testing, the limitation on operation of the units was removed.

The S/RVs in both units have been replaced utilizing S/RVs with improved subcomponents.

Controls have been put in place such that future refurbishment of spare S/RVs will be accomplished utilizing the improved S/RV subcomponents.

F. PREVIOUS OCCURRENCES

No previous examples of this type of wear failure of an S/RV were identified.

There have been previous instances of S/RVs and Main Steam Safety Valves (MSSVs) being outside of the Technical Specification allowed value (+/- 1%). Following the Unit 1 refuel outage in October of 2000 (Q1R16), the S/RV setpoint was 2.203% lower

than nameplate, one MSSV setpoint was 2.0643% greater than nameplate, and one MSSV setpoint was 1.20% greater than nameplate. Following the Unit 2 refuel outage in February of 2002 (Q2R16), the S/RV setpoint was 2.026% greater than nameplate, one MSSV setpoint was 2.8% less than nameplate, one MSSV setpoint was 1.8% less than nameplate, and one MSSV setpoint was 1.5% less than nameplate. Following the Unit 1 refuel outage in November of 2002 (Q1R17), the S/RV setpoint was 2.203% greater than nameplate and one MSSV setpoint was 1.2% lower than nameplate.

In every case except the Q1R17 S/RV, the effect of the setpoint on functionality was assessed. For every case, including the Q1R17 S/RV, the setpoint was within the code allowable of +/- 3%, and therefore there was no effect on functionality.

In addition to the corrective actions for the Q2R17 S/RV results, and based on the history described above, Quad Cities Nuclear Power Station is pursuing a revision to the Technical Specification allowable value for the MSSVs and S/RVs to reflect the code allowable, as well as any further modifications to the MSSVs and S/RVs to support the change in the allowable value.

G. COMPONENT FAILURE DATA

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