Commonwealth Edison
Dresden Nuclear Power Station
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March 30, 1993

CWS PMLTR 93-0132

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

Licensee Event Report 93-010, Docket 050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(i)(B).

for 3/30/93

Charles W. Schroeder
Station Manager
Dresden Station

CWS/slb

cc: A. Bert Davis, Regional Administrator, Region III
NRC Resident Inspector's Office
File/NRC
File/Numerical

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IE22

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Dresden Nuclear Power Station, Unit 2	0 5 0 0 0 2 3 7	1 of 0 4

Title (4) Main Steam Safety Valve Outside Technical Specification Limit Due to Setpoint Drift

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	0 3	9 3	9 3	---	0 1 0	--- 0 0	0 3	3 0	9 3		0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
N		20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	0 0 0	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
Neil Spooner, Technical Staff System Engineer Ext. 2789	8 1 5 9 4 2 - 2 9 2 0

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	S B	R V	D 2 4 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

On March 3, 1993 while performing Dresden Maintenance Procedure (DMP) 200-03, Main Steam Safety Valve Pre-Maintenance Test, the as-found lift setpoint for main steam safety valve 2-203-4C was found outside of Technical Specification (Tech. Spec.) limits. The apparent cause of the event was attributed to setpoint drift. The safety significance was determined to be minimal since the setpoint drift did not exceed the 3% ASME band which is also used in the reactor vessel overpressurization reload analysis. As corrective action, the affected valve will be rebuilt and calibrated in accordance with DMP 200-30, Reactor Main Steam Safety Valve Repair and Post Maintenance Testing, and DMP 040-30, Safety/Relief Valve Lapping. A rebuilt, calibrated safety valve was installed in its place. Similar previous events are listed under Section F of this report.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor -2527 MWt rated core thermal power.

Nuclear Tracking System (NTS) tracking code numbers are identified in the text as (XXX-XXX-XX-XXXXX).

EVENT IDENTIFICATION:

Main Steam Safety Valve Outside Technical Specification Limit Due to Setpoint Drift

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: March 3, 1993 Event Time: 0800 hrs.

Reactor Mode: N Mode Name: Refuel Power Level: 0

Reactor Coolant System (RCS) Pressure: 0 psig

B. DESCRIPTION OF EVENT:

On March 3, 1993 at 0800 hours with Unit 2 in the Refuel Mode (D2R13), while performing Dresden Maintenance Procedure (DMP) 200-03, Main Steam Safety Valve Pre-Maintenance Test, the as-found lift setpoint for main steam [SB] safety valve 2-203-4C (Serial Number BK 6277) was found outside of Technical Specification (Tech. Spec.) limits. Tech. Spec. 4.6.E requires the lift setpoint for each main steam safety valve to be within +/-1% of the designed lift setpoint for the valve. The as-found lift setpoint for safety valve 2-203-4C was 1265 psig; this was outside Tech. Spec limits, since the designed lift setpoint for the valve (BK 6277) is 1250 psig. Although outside of the 1% band required by Tech. Spec., the valve was well within the 3% band required by American Society of Mechanical Engineers (ASME) Code Section XI, as referenced in Operations and Maintenance (OM-1) guidelines (permission had been previously granted by the NRC to reference a later ASME Code for the 3% acceptance band). It therefore was not necessary to increase the number of safety valves being tested above the amount required by Tech. Spec. 4.6.E.

C. Apparent Cause of Event:

This event is reported in accordance with 10CFR50.73(a)(2)(i)(B) which requires the reporting of any operation or condition prohibited by the Technical Specifications.

The apparent cause of the event has been attributed to setpoint drift. Safety valve drift can be caused by a change in position of the valve compression screw, or any contact between the shaft and the internal adjustment guide. Visual inspection of the valve, and checks for seat leakage are made in accordance with DMP 200-03 prior to 'popping' the safety valve to determine its as-found setpoint. No physical abnormalities or seat leakage were observed. A maintenance history review revealed that safety valve (BK 6277) was rebuilt and calibrated under Work Request D65460 in accordance with the following procedures:

DMP 200-30, rev.3	Reactor Main Steam Safety Valve Repair and Post Maintenance Inspection
DMP 040-30, rev.0	Safety/Relief Valve Lapping
DMP 200-33, rev.1	Six Inch Safety Valve Post Maintenance Testing

D. Safety Analysis of Event:

A reactor vessel overpressurization reload analysis is performed for each operating cycle of the Unit. The most limiting overpressurization transient analysis is closure of the Main Steam Isolation Valves (MSIVs) at full power in conjunction with a postulated failure of the MSIV 10% closure scram. No credit is taken for actuation of the electromatic relief valves or the relief function of the Target Rock valve. In addition, further conservatism is used by increasing the lift setpoints on all safety valves by 3% and decreasing the rated flow of each valve by 4%. The analysis confirms that, for the conditions stated above, the reactor vessel design pressure (1250 psig) is not exceeded by more than 10% (1375 psig), in accordance with ASME Code requirements. It also assures that the Reactor Coolant System Safety Limit (1345 psig steam dome pressure) is not exceeded.

As a result, safety valves which are found with lift setpoints under the +3% setpoint band pose no safety concern since the event falls within the conditions used in the overpressurization analysis. Any safety valve found with a lift setpoint outside the +3% setpoint band is reviewed on a case-by-case basis.

For the above reasons, the safety significance of this event (lift setpoint under the +3% setpoint band) was considered minimal.

E. Corrective Actions:

Revisions to DMP 200-30 and DMP 040-30 have been made to significantly improve the repair/rebuild and calibration process in an effort to reduce safety valve setpoint drift. DMP 200-33 has been incorporated into the revised DMP 200-30 (which has been renamed 'Reactor Main Steam Safety Valve Repair and Post Maintenance Testing') and has been subsequently deleted.

Main steam safety valve 2-203-4C was replaced with a rebuilt, calibrated safety valve (Serial Number BK 6271) under Work Request D07121, having a designed lift setpoint of 1250 psig. The rebuilt valve (BK 6271) was calibrated in accordance with DMP 200-30 under Work Request D04040, with an as-left lift setpoint of 1246 psig. The removed safety valve (BK 6277) will be rebuilt and calibrated in accordance with the latest revisions of DMP 200-30 and DMP 040-30. It will then be stored for future use.

F. Previous Occurances:

LER/Docket Numbers	Title
90-021/050237	Main Steam Safety Valves 2-203-4E Thru 4H Setpoints Found Outside Technical Specification Limits Due to Setpoint Drift

While performing DMP 200-03, safety valves 2-203-4E thru -4H were discovered to have as-found lift setpoints above Tech. Spec. limits. The apparent cause of the event was setpoint drift. The valves had not been previously rebuilt under revised procedures, which had been upgraded to reduce setpoint drift. The affected valves were rebuilt and calibrated under the revised procedures and then stored for future use. Rebuilt and calibrated safety valves were installed in their place.

89-002/050237 Setpoints on Main Steam Safety Valves Found Outside Technical Specification Limits Due to Setpoint Drift

While performing DMP 200-03, safety valves 2-203-4A and -4C were discovered to have as-found lift setpoints above Tech. Spec. limits. The apparent cause of the event was setpoint drift. The valves had not been previously rebuilt under revised procedures, which had been upgraded to reduce setpoint drift. The affected valves were rebuilt and calibrated under the revised procedures and then stored for future use. Rebuilt and calibrated safety valves were installed in their place.

G. Component Failure Data:

Manufacturer	Nomenclature	Model #	MFG part #
Dresser	safety valve	3777QA	BK 6277

An industry wide Nuclear Plant Reliability Data System (NPRDS) data base search was performed for setpoint drift of the Dresser model 3777QA valve. Of the events listed, 9 setpoint drift events occurred at Dresden Station.