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Dresden Nuclear Power Station
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Licensee Event Report #90-021, Docket #050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(i)(B).

E.D. Eenigenburg
Station Manager
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EDE/dal

Enclosure

cc: A. Bert Davis, Regional Administrator, Region III
File/NRC
File/Numerical

ZDVR/109

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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 5

Title (4) Main Steam Safety Valves 2-203-4E Thru 4H Setpoints Found Outside Technical Specification Limits 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)
									N/A	0 5 0 0 0
1 1	0 4	9 0	9 0	0 2 1	0 0	0 1	0 7	9 1	N/A	0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
		___ 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	
Frank J. Dundek, Technical Staff Engineer Ext. 2523	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)

Starting on November 4, 1990 at 0800 with the Unit 2 in refuel mode, during the performance of Dresden Maintenance Procedure (DMP) 200-03, Unit 2/3 Six Inch Safety Valve Pre-Maintenance Testing, the Main Steam Safety Valves 2-203-4E thru 4H (Serial numbers BK 6532, BK 6263, BK 6299, BK 6526, respectively) opened at a pressure in excess of the Technical Specification 4.6.E required setpoints for these valves of +/- 1% of their design setpoint. The cause was attributed to setpoint drift. These valves will be replaced with rebuilt valves that have been overhauled, set, and retested satisfactorily prior to reinstallation. The safety significance of this event is minimal based on an evaluation which shows that with the valves in the as-found conditions, the reactor pressure safety limit would not be exceeded under any design basis event. The last event of this type was reported by LER No. 90-010 on Docket No. 050249, involving a Main Steam Safety Valve setpoint found outside Technical Specification limits due to setpoint drift.

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

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2527 MWt rated core thermal power
Energy Industry Identification System (EIIS) codes are identified in the text as XX.
Nuclear Tracking System (NTS) tracking code numbers are identified in the text as (XXX-XXX-XX-XXXXX)

EVENT IDENTIFICATION:

Main Steam Safety Valves [SB] 2-203-4E thru 4H were found outside Technical Specification 4.6.E limits due to setpoint drift.

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: November 4, 1990 Event Time: 0800 Hours

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

Reactor Coolant System (RCS) Pressure: 0 psig

B. DESCRIPTION OF EVENT:

Starting at 0800 hours on November 4, 1990 during Unit 2 refuel outage, DMP 200-03 was performed on Main Steam Safety Valves 2-203-4E thru 4H. All of the as-found setpoints of these valves were out of the +/- 1% design setpoints as listed in the Technical Specifications 4.6.E. The quantitative results are listed in the following table:

Valve EPN	Valve Serial No.	Design Setpoint	"As Found" Setpoint
2-203-4E	BK 6532	1260 psig	1158 psig
2-203-4F	BK 6263	1260 psig	1283 psig
2-203-4G	BK 6299	1240 psig	1263 psig
2-203-4H	BK 6526	1250 psig	1272 psig

The safety valves are tested in accordance with the American Society of Mechanical Engineer (ASME) Boiler and Pressure Vessel Code, 1977 Edition through the Summer 1979 Addenda, Section XI Subsection IWV. The Summer 1979 Addenda requires additional safety valves be tested when any safety valve being tested fails to function properly during a regular test. The ASME Section XI Summer 1979 Addenda does not provide acceptance criteria by which it can be determined when additional safety valves need to be tested.

Since the 1979 Addenda does provide acceptance criteria, the Commonwealth Edison Production Services Department In-Service Testing Coordinator recommended on November 7, 1990, that later ASME Codes be consulted for acceptance criteria. Later editions of the ASME Code reference Operations and Maintenance (OM-1) for acceptance criteria. This acceptance criteria states that additional testing will be performed if the as-found setpoint is 3% greater than the design setpoint. The Main Steam Safety Valves tested did not exceed this criteria and therefore no additional safety valves needed to be tested during the refuel outage. As part of the Technical Specification requirements, the valves were replaced with rebuilt safety valves, which were bench tested to Technical Specification setpoints.

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

The Technical Staff In-Service Testing personnel reviewed the first safety valve test data sheet on November 6, 1990 and the remaining data sheets on November 15, 1990; further action concerning reportability was not initially taken since the ASME limit of 3% above the design limit was not exceeded. Subsequently on December 28, 1990, upon further review of the setpoint data with an Assistant Technical Staff Supervisor, this LER was initiated since the Technical Specification limits of +/- 1% were exceeded.

C. APPARENT CAUSE OF EVENT:

The cause of the Main Steam Safety Valves 2-203-4E thru 4H setpoint discrepancy has been attributed to setpoint drift. This can be caused by a change in the position of the compression screw or any contact between the shaft and the internal adjustment guide. These valves were installed during the 1987 Unit 2 Refuel Outage under Work Request D55681.

This report is being submitted in accordance with 10CFR50.73(a)(2)(i)(B), which requires the reporting of any operation or condition prohibited by the plant's Technical Specifications.

D. SAFETY ANALYSIS OF EVENT:

The safety valves are designed to relieve steam from the reactor vessel. The four Electromatic Relief Valves and the Target Rock Safety/Relief Valve are designed to automatically open prior to reactor pressure reaching the Safety Valve opening setpoints and relieve steam from the reactor vessel to the Torus, thus preventing automatic lifting of the Safety Valves. These Electromatic Relief Valves and the Target Rock Relief Valve were operable and would have automatically opened if required; these valves may also be opened via control switches from the Control Room.

Based on a safety review by Advanced Nuclear Fuels Corporation, the as-found Main Steam Safety Valve setpoints would not have allowed reactor vessel pressure to exceed the ASME maximum vessel pressure limit. The most limiting over-pressurization transient reload analysis is Main Steam Isolation Valve (MSIV) closure at full power in conjunction with a postulated failure of the MSIV 10% closure scram. This transient analysis was redone using the actual test pressures with an additional 3 psi margin for each group of valves as shown in Table 1, and the maximum setpoint tolerance for the target rock valve safety function. Further conservatism was included in both analyses by disallowing the operation of all relief valves. The Main Steam Safety Valve setpoints used are shown in Table 1, attached.

For the ASME overpressurization analysis, the pressure in the reactor vessel is not allowed to exceed the design pressure by more than 10%. Therefore, with a design pressure 1250 psig, the maximum allowable pressure is 1375 psig. Technical Specification 1.2, Reactor Coolant System Safety Limit of 1345 psig (as measured by the vessel steam space pressure indicator) insures margin to 1375 psig at the lowest elevation of the reactor vessel. The result of the transient analysis shows that the maximum vessel pressure in the worst case would have been 1331 psig at the lower plenum region of the reactor vessel. For these reasons, the safety significance of this event was minimal.

E. CORRECTIVE ACTIONS:

All removed Main Steam Safety Valves are overhauled and tested, so that setpoints are verified to be within one percent of the design value prior to installation. Following a previous occurrence of safety valve setpoint drift, the procedure for overhauling safety valves was substantially improved. The centering of the valve spindle has received significant attention. The tolerances for centering have been decreased and the methods for centering have been improved. Safety Valves 2-203-4E thru 4H had not been overhauled using the new procedure. It is believed that the new procedure will significantly reduce the possibility for setpoint drift of the safety valves in the future. Also, DMP 200-03 will be revised to clarify the step requiring the initiation of a LER if the setpoints exceed the Technical Specification limits of +/- 1% of the design setpoint. The step containing the ASME Section XI expansion requirements will also be clarified. These two clarifications will ensure proper action is taken when a safety valves setpoint is outside the Technical Specification Limits (237-200-90-15701).

F. PREVIOUS OCCURRENCES:

LER/Docket Numbers	Title
88-010-01/050249	Main Steam Safety Valve Setpoints Found Outside Technical Specifications Limits Due to Setpoint Drift
	This last previous occurrence of a Main Steam Safety Valve to be found outside the Technical Specification setpoint failed by setpoint drift. As corrective action, this event had referred to a revised procedure to overhaul these safety valves that had been written before the valves failed. The valves in the system at that time were not rebuilt to the improved procedure. Also, Main Steam Safety Valve 3-203-4H was overhauled and tested to verify its setpoint prior to installation.
87-030/050237	Main Steam Safety Valve Setpoint Found Outside Technical Specification Limits Due To Mishandling and Setpoint Drift
	This occurrence of a Main Steam Safety Valve found outside the Technical Specification tolerance limit caused by mishandling the valves during transport from the drywell to the test boiler. As a corrective action, the procedure for overhauling safety valves was substantially improved. Also, to prevent future reoccurrence of damage in transit from the drywell to the test stand, a protective guard has been fabricated to protect the stem assembly.

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Dresser Valves	Main Steam Safety Valve	3777QA	BK 6532
			BK 6263
			BK 6526
			BK 6299

An industry wide NPRDS data base search revealed 18 instances of Main Steam Safety Valves manufactured by Dresser Industries in excess of setpoint tolerances. All occurrences involved the setpoint to be off, and the valves were readjusted to the proper setting and returned to service.

TABLE 1

SAFETY VALVE ANALYSIS DATA

Valve Type	Quantity	Ideal Setpoints	As-Found Setpoints	Original Reload Analysis Setpoints	Reanalysis Setpoints
Safety Valve	2	1240	1. 1263	1252.6	1266
			2. Not tested	1252.6	1266
Safety Valve	2	1250	1. 1272	1262.7	1275
			2. Not tested	1262.7	1275
Safety Valve	4	1260	1. 1158	1272.8	1286
			2. 1283	1272.8	1286
			3. Not tested	1272.8	1286
			4. Not tested	1272.8	1286
Target Rock (Safety Mode)	1	1135	Previously replaced, bench test of replacement valve was 1135 psig	1146.5	1146.5