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AUTH.NAME AUTHOR AFFILIATION
WEBER,G.A. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
BLIND,A.A. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 90-016-01:on 901221,determined that two of three safety valves had lift settings outside TS 4.4.3 acceptance criteria.Possibly caused by steam cutting in disc insert & nozzle.Valves satisfactorily tested.W/910607 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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Bridgman, MI 49106
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June 7, 1991

United States Nuclear Regulatory Commission
Document Control Desk
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Operating Licenses DPR-58
Docket No. 50-315

Document Control Manager:

In accordance with the criteria established by 10 CFR 50.73 entitled Licensee Event Report System, the following report is being submitted:

90-016-01

Sincerely,

A. Alan Blind

A.A. Blind
Plant Manager

AAB:sb

Attachment

c: D.H. Williams, Jr.
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NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
D. C. Cook Nuclear Plant - Unit 1	0 5 0 0 0 3 1 5	1 OF 0 3

TITLE (4) Failure of two pressurizer safety valves to meet Technical Specification required surveillance test criteria

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
1 2	2 1	9 0	9 0	0 1 6	0 1	0 6	0 7	9 1		0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
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, NRC Form 366A)*
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
G. A. Weber - Plant Engineering Superintendent	AREA CODE 6 1 6 — 4 6 5 - 5 9 0 1

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	A B	R V	C 7 1 0	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)

This revision is being submitted to provide additional information regarding the Cause Description and Corrective Actions Statement.

On December 21, 1990 it was determined that two of three pressurizer safety valves sent to a test laboratory off site for testing required by Technical Specification 4.4.3 had been found with lift settings outside of the acceptance criteria. Acceptable settings are between 2461 psig and 2509 psig. Valve 1-SV-45B lifted at 2451 psig and valve 1-SV-45C at 2548 psig.

An evaluation of the test data revealed that the valve with a lift setting of 10 psi below acceptance criteria would have had no impact on the operability of affected components, as it would have lifted prematurely. The valve with a lift setting exceeding the acceptable range did so by 39 psi. Calculations performed indicate that if the reactor coolant system pressure had reached 2548 psig, no damage would have been incurred by pipe, fittings, valves and/or the pressurizer. The UFSAR was reviewed for impact of the as-found setpoints. The high setpoint would have been offset by the additional capacity of power operated relief valves and full safety valve capacity is not required. Also reviewed were criteria associated with Departure from Nucleate Boiling. No adverse consequences applied to safety analyses.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
D. C. Cook Nuclear Plant - Unit 1	0 5 0 0 0 3 1 5	9 0	— 0 1 6	— 0 1	0 2 OF 0 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

This revision is being submitted to provide additional information regarding the Cause Description and corrective actions taken.

Conditions Prior to Occurrence

Unit One - Mode 5 (Cold Shutdown).
Unit Two - Mode 1 (Power Operation).

Description of Event

On December 21, 1990 it was determined that two of the three pressurizer safety valves (EIIS/AB-RV) had lift settings outside Technical Specification 3.4.3 acceptance criteria. The safety valves are tested at a test laboratory using steam at nominal temperature and pressure, as required by Technical Specification Surveillance 4.4.3. The valves are required to lift between 2461 and 2509 psig. Replacement valves tested by the same laboratory have been installed in place of the three valves removed for testing. Valve 1-SV-45B lifted at 2451 psig and 1-SV-45C lifted at 2548 psig. The third valve, 1-SV-45A, was acceptable. Technical Specification 4.4.3 requires that each pressurizer code safety valve be demonstrated operable per Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

Cause of Event

The vendor could not conclusively determine the cause for 1-SV-45B or 1-SV-45C to have unacceptable lift setpoints.

However, the vendor reported that inspection of 1-SV-45C revealed that the disc insert and nozzle showed signs of steam cutting. The steam cuts were evidenced by the removal of 0.002 inches of material during lapping. Steam cuts cause leakage, which will tend to increase the lift setpoint pressure.

The cause for 1-SV-45B to have a low lift setpoint could not be determined.

These failures are considered isolated events and not indicative of generic failures of the Pressurizer Safety Valves. The previous reportable event (LER 50-316/89-04) was attributed to setpoint drift and is also considered to be an isolated event.

Analysis of Event

Test data was assessed following receipt. The valve which was found with a lift setting of 10 psi below acceptance criteria would have had no impact on the operability of affected components. It would have lifted prematurely and the plant maintained in a safe condition. The valve with a lift setting which exceeded acceptance criteria did so by 39 psi. Calculations subsequently performed indicate that, had the reactor coolant system reached the setpoint lift pressure of 1-SV-45C, no damage would have been incurred by the pipe, fittings, valves, and/or the pressurizer.

The UFSAR Chapter 14 Safety analyses applicable to Unit 1 were reviewed for the impact of "as found" safety valve opening setpoints of 2451 psig for 1-SV-45B and 2548 psig for 1-SV-45C.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Analysis of Event Continued

For over-pressurization cases, the additional capacity of the PORVs would offset the higher final setpoint, and the full safety valve capacity is not required for any case. Therefore, the increased setpoint of the higher value is of no concern.

For Departure from Nucleate Boiling (DNB) cases, the lowered setpoint would have been reached for only one case, the loss of external load/turbine trip case, minimum feedback, with pressure control. The accident analysis minimum DNBR for this case is approximately 1.79, and the acceptable value is 1.45, so the slight pressure reduction is easily offset by the existing margin.

In conclusion, the specified "as found" pressurizer safety valve setpoint would have no consequence on the applicable safety analyses. The failure of the valves to meet the surveillance test criteria are being reported under 10CFR50.73 (a)(2)(i)(B), "Any operation or condition prohibited by the plant's Technical Specifications."

Corrective Action

Valves 1-SV-45B and 1-SV-45C required lapping of the disc and nozzle seats and were satisfactorily tested for steam set pressure and seat leakage.

Failed Component Identification

Pressurizer Safety Valve
Plant Designation: 1-SV-45B and 1-SV-45C
Manufacturer: Crosby Valve Company
Model: HB-86-BP
EIIS Code: AB-RV

Previous Similar Events

LER 50-316/89-04