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REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8903230349 DOC.DATE: 89/03/10 NOTARIZED: NO DOCKET #
FACIL:50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
AUTH.NAME AUTHOR AFFILIATION
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SMITH,W.G. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 89-004-00:on 890210,pressurizer safety valve lift point due to setpoint drift.

W/8 ltr.

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

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	ACRS WYLIE	1	1	AEOD/DOA	1	1
	AEOD/DSP/TPAB	1	1	AEOD/ROAB/DSP	2	2
	ARM/DCTS/DAB	1	1	DEDRO	1	1
	NRR/DEST/ADE 8H	1	1	NRR/DEST/ADS 7E	1	0
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EXTERNAL:	EG&G WILLIAMS,S	4	4	FORD BLDG HOY,A	1	1
	H ST LOBBY WARD	1	1	LPDR	1	1
	NRC PDR	1	1	NSIC MAYS,G	1	1
	NSIC MURPHY,G.A	1	1			

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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 2	0 5 0 0 0 3 1 6	1 OF 0 4

TITLE (4) Pressurizer Safety Valve Lift Point Due To Setpoint Drift

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 2	1 0	8 9	8 9	0 0 4	0 0	0 3	1 0	8 9		0 5 0 0 0
										0 5 0 0 0

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

OPERATING MODE (9):
POWER LEVEL (10): 0 0 0

[ ] 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 AREA CODE	
T. K. Postlewait - Technical Engineering Superintendent	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)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

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

On November 24, 1989, during a hydrostatic test of the RCS (Reactor Coolant System), leakage through the pressurizer safety valves (EIIS/AB-RV) was detected via temperature and acoustic monitor indications. Since the leak could not be traced to a single valve, two of the three valves that had the potential to contribute to the leakage (2-SV-45B and 2-SV-45C) were removed and sent to Wyle Laboratories for analysis.

On February 10, during Laboratory Bench Testing, it was determined that the initial lift pressure for 2-SV-45B was 2526 psig, 16 psi greater than the maximum setpoint pressure allowed by Technical Specification 3.4.3 (2460 to 2510 psig). 2-SV-45B was subsequently bench tested three more times, and each time it was within the setpoint allowable tolerances (2497 PSIG, 2475 psig, 2495 psig). The as-found lift pressure for 2-SV-45C was within allowable values. 2-SV-45B was last tested in September 1986 and no problems were encountered at that time. There were no additional inoperable components, structures, or systems that contributed to this event. The cause for the setpoint drift could not be determined.

A Safety Evaluation revealed that the RCS safety limit would not have been exceeded. This event did not have any safety consequences and did not represent a significant hazard to the public health or safety.

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PDR ADOCK 05000316
S PDC

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
D. C. Cook Nuclear Plant Unit - 2	0 5 0 0 0 3 1 6	8 9	— 0 0 4	— 0 0	0 2 OF 0 4

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

Conditions Prior to Occurrence

Unit 2 in Mode 5 (Cold Shutdown) following Steam Generator replacement, with no fuel in core.

Description of Event

On November 24, 1989, during a hydrostatic test of the RCS (Reactor Coolant System), leakage through the pressurizer safety valves (EIIS/AB-RV) was detected via temperature and acoustic monitor indications. Since the leak could not be traced to a single valve, two of the three valves that had the potential to contribute to the leakage (2-SV-45B and 2-SV-45C) were removed and sent to Wyle Laboratories for analysis.

On February 10, during Laboratory Bench Testing, it was determined that the initial lift pressure for 2-SV-45B was 2526 psig, 16 psi greater than the maximum setpoint pressure allowed by Technical Specification 3.4.3 (2460 to 2510 psig). 2-SV-45B was subsequently bench tested three more times, and each time it was within the setpoint allowable tolerances (2497 PSIG, 2475 psig, 2495 psig). The as-found lift pressure for 2-SV-45C was within allowable values. 2-SV-45B was last tested in September 1986 and no problems were encountered at that time.

There were no additional inoperable components, structures or systems that contributed to this event.

Cause of Event

This event is attributed to setpoint drift. The cause for the setpoint drift could not be determined. A factory representative inspected the valve and could not identify any problems which would have caused the valve to lift out of tolerance.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Analysis of Event

In Technical Specification 2.1.2, A Reactor Coolant System Pressure Safety Limit of 2735 psig is stipulated. The corresponding Basis explains that the reactor vessel and the pressurizer were designed to ASME B&PV Section III, which permits a maximum transient pressure of 2735 psig (110 percent of the design pressure of 2485 psig.) In addition, the RCS piping, valves and fittings, are designed to ANSI B 31.1, 1967 edition, which permits a maximum transient pressure of 2985 psig (120 percent of 2485 psig.) The Basis also states that the entire RCS is hydrotested at 3107 psig (125 percent of 2485 psig.)

During operation, all pressurizer code safety valves must be operable to prevent the RCS from being pressurized above its safety limit of 2735 psig. The Basis for Technical Specification 3.4.3 explains that the combined relief capacity of the three code safety valves is greater than the maximum surge rate resulting from a complete loss of load, assuming no reactor trip until the first Reactor Protection System trip setpoint is reached. No credit is taken for a direct reactor trip on the loss of load, and assuming no operation of either power operated relief valves or the steam dump valves. In the instance of high set point deviation (2526 psig), the RCS pressure could have reached a value of 2602 psig (2526 psig plus 3 percent code allowable accumulation for a valve to attain its rated lift.)

An assessment of the safety consequences and implications revealed that this event did not have any safety significance. The safety limit of 2735 psig would not have been exceeded under any circumstances since the two remaining code safety valves were operable, and within allowable tolerances. Their lift setpoints were 2510 psig or less, and the third safety valve would have lifted at 2602 psig in the worst case.

In conclusion, this event did not have any safety consequences and it did not represent a significant hazard to the public health or safety.

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

Corrective Actions

Safety Valve Refurbishment:

1. A Crosby Valve Co. serviceman inspected 2-SV-45B and could not determine the cause for 2-SV-45B to lift out of tolerance. The Serviceman lapped and polished the disc and seat and verified that the disc and seat measurements were within allowable tolerances.

2. 2-SV-45C had the same work performed on it as was performed on 2-SV-45B. The disc and seat were within allowable tolerances and the valve is in acceptable condition.

Failed Component Identification

Manufacturer: Crosby Valve Co
Model No: HB-86-BP
EIIS Code: AB-RV

Previous Similar Events

None

Indiana Michigan
Power Company
Cook Nuclear Plant
P.O Box 458
Bridgman, MI 49106
616 465 5901

INDIANA
MICHIGAN
POWER

March 10, 1989

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Operating License DPR-74
Docket No. 50-316

Document Control Manager:

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

89-004-00

Sincerely,

W. G. Smith, Jr.
W. G. Smith, Jr.
Plant Manager

WGS:clw

Attachment

cc: D. H. Williams, Jr.
A. B. Davis, Region III
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