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ACCESSION NBR:9505150175 DOC.DATE: 95/05/08 NOTARIZED: NO DOCKET #
FACIL:50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana M 05000316
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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 95-003-00:on 950203,two pressurizer safety valves failed to meet TS required surveillance test criteria.Partially disassembled valves 2-SV-45A & 2-SV-45C.W/950508 ltr.

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

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May 8, 1995

United States Nuclear Regulatory Commission
Document Control Desk
Rockville, Maryland 20852

Operating Licenses DPR-74
Docket No. 50-316

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:

95-003-00

Sincerely,

A. Alan Blind

A. A. Blind
Plant Manager

/mr

Attachment

c: J. B. Martin, Region III
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NRC FORM 366
(5-92)

NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
D. C. COOK NUCLEAR PLANT - UNIT 2	05000 316	1 OF 4

TITLE (4) FAILURE OF TWO PRESSURIZER SAFETY VALVES TO MEET TECHNICAL SPECIFICATION REQUIRED SURVEILLANCE TEST CRITERIA

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
02	03	95	95	-- 003 --	00	05	08	95	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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 (Include Area Code)
G. A. WEBER - PLANT ENGINEERING SUPERINTENDENT	616-465-5901

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	AB	RV	C710	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 15 single-spaced typewritten lines) (16)

On October 5, 1994 with Unit 2 in Mode 6 (no fuel), the Pressurizer Safety Valves in service prior to the Unit 2 Refueling Outage were removed for shipment to an offsite lab for set point testing. On February 3, 1995, with Unit 2 in Mode 1 at 100 percent reactor thermal power, Wyle Laboratories determined that two of three Unit 2 Pressurizer Safety Valves were found with lift settings outside the Technical Specification acceptance criteria. Acceptable settings are between 2461 psig and 2509 psig. Valve 2-SV-45A was found to have a lift setpoint of 2524 psig, and valve 2-SV-45C had a lift setpoint of 2538 psig. There was no safety significance because the safety valves would still have limited the peak transient pressure to 2615 psig in the event of an overpressure transient . This is below the Technical Specification safety limit of 2735 psig. No specific cause for the drift was determined. Both 2-SV-45A and 2-SV-45C were partially disassembled (retaining spring compression) and inspected. No problems were noted. The nozzle and disc seating surfaces were lapped and polished. The valves were reassembled and tested satisfactorily.

Conditions Prior to Occurrence

Unit 2 - Mode 1, 100 percent Reactor Thermal Power (reflects condition at the time that Wyle Labs determined the event)

Description of Event

On October 5, 1994 with Unit 2 in Mode 6 (no fuel), the Pressurizer Safety Valves in service prior to the Unit 2 Refueling Outage were removed for shipment to an offsite lab for set point testing. On February 3, 1995, with Unit 2 in Mode 1 at 100 percent reactor thermal power, it was determined that two of the three Pressurizer Safety Valves (EIIS/AB-RV), Crosby Valve Model Numbers HB-BP-86 and HB-86-BP, had lift settings outside Technical Specification 3.4.3 acceptance criteria. Technical Specification Surveillance 4.4.3 requires that each Pressurizer Code Safety Valve be demonstrated operable per Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a. The safety valves are tested at a test laboratory using steam at nominal temperature and pressure. The valves are required to lift at 2485 psig +/- 1 percent, (i.e. between 2461 and 2509 psig). Valve 2-SV-45A lifted at 2524 psig and 2-SV-45C lifted at 2538 psig. The third valve, 2-SV-45B, had an acceptable lift setpoint of 2500 psig.

The test report was completed by Wyle Labs on March 8, 1995, and forwarded to AEPSC in Columbus, Ohio. The test report was sent to the D.C. Cook Plant in early April 1995. A condition report was written by the plant on April 7, 1995, and this event was determined reportable on April 13, 1995.

Cause of Event

The phenomena of pressurizer safety valve setpoint drift outside of design criteria experienced at the D.C. Cook Plant is similar to that experienced throughout the nuclear industry for pressurizer as well as main steam safety valves. Crosby Valve and Wyle Labs were contacted to discuss the test results, but no specific cause for the drift was determined.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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	9 5	— 0 0 3 —	0 0	0 3 OF 0 4

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

Analysis of Event

This event has been determined reportable under the provisions of 10CFR50.73 (a) (2) (i) (B) as an operation prohibited by Technical Specification 3.4.3, which requires all of the Pressurizer Safety Valves to be operable with a lift setting of 2485 psig +/- 1 percent.

The as-found lift setpoints of safety valves 2-SV-45A and 2-SV-45C did not have any actual impact on the Reactor Coolant System (RCS) since the pressure would not have exceeded the maximum transient limit of 2735 psig per ASME B&PV Section III, which is 110 percent of the design pressure (2485 psig). There was no impact on the health or safety of the public.

Safety Valve 2-SV-45C (worst case) had a lift setpoint of 2538 psig. The valve would have reached it's rated capacity at an RCS pressure of 2615 psig (2538 psig plus 3 percent accumulation). Valve 2-SV-45A would have attained its full rated lift at 2600 psig (2524 psig plus 3 percent).

Also, the reactor vessel and pressurizer were designed to ASME B&PV Section III which permits a maximum transient pressure of 2735 psig, 110 percent of design pressure (2485 psig). The RCS piping, valves and fittings are designed to ANSI B31.1, 1967 Edition, which permits a maximum transient pressure of 2985 psig, 120 percent of design pressure (2485 psig). In addition, the entire RCS was hydro tested to 3107 psig, 125 percent of design pressure (2485 psig), to demonstrate system integrity prior to initial operation.

In conclusion, this event did not have any safety significance and did not represent a hazard to the public health and safety. The safety limit of 2735 psig would not have been exceeded since the maximum RCS pressure would not have exceeded 2615 psig (2-SV-45C setpoint of 2538 plus 3 percent).

Corrective Action

Valves 2-SV-45A and 2-SV-45C were partially disassembled (retaining spring compression) and the nozzle and disc seating surfaces were lapped and polished. The valves were then reassembled, retested and reset to comply with the acceptance criteria set point, and to verify that the seat leakage requirements were met.

Corrective Action continued

Since a specific root cause could not be determined, no preventive action is planned at this time. AEPSC is continuing to participate in an ongoing dialogue with other utilities in an effort to identify ways to improve valve performance. Finally, as part of Unit 1 Cycle 16 and the Unit 2 Cycle 12, AEPSC is pursuing a change in pressurizer safety valve set point tolerances from +/- 1 percent to +/- 3 percent. For Unit 1 the Westinghouse analysis has been received and a request will be submitted to the NRC in the near future. It is anticipated that the analysis for Unit 2 will be completed by the end of 1995.

Failed Component Identification

Pressurizer Safety Valve Plant Designation: 2-SV-45A
Manufacturer: Crosby Valve Company
Model: HB-BP-86
EIIS Code: AB-RV

Pressurizer Safety Valve Plant Designation: 2-SV-45C
Manufacturer: Crosby Valve Company
Model: HB-86-BP
EIIS: AB-RV

Previous Similar Events

LER: 50-315/90-16, 92-09, 94-04
LER: 50-316/89-04, 92-06