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FACIL: 50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315

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SUBJECT: LER 86-020-01: on 851102, main steam safety valves discovered out of spec during surveillance testing. Caused by lack of procedural instructions. Safety valves setpoints reset. W/871029 ltr.

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EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	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 1	0 5 0 0 0 3 1 5	1 OF 0 5

TITLE (4)
Main Steam Safety Valves Out of Specification 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)
1 1	0 2	8 5	8 6	0 2 0	0 1	1 0	2 9	8 7	D. C. Cook - Unit 2	0 5 0 0 0 3 1 6
										0 5 0 0 0

OPERATING MODE (9): 3
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 (AREA CODE)
J. B. Droste - Maintenance 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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S B	R V	D 2 4 3	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 is a supplemental report to a previously submitted LER, 315/86-20-00.

On November 2, 1985, with the reactor in Hot Stand-By, eight of twenty Unit 1 Main Steam Safety Valves (MSSV) lift setpoints were found out of specification during surveil-lance testing. Four of twenty Unit 2 MSSV lift setpoints were discovered out of specification during surveillance testing on June 23, 1986. Also, similar surveillance test failures on October 16, 1983 and July 2, 1984 were not properly reported. These events were determined to be reportable on August 25, 1986 after a review of documentation on MSSV setpoint verification. A lack of procedural instructions contributed to the failure to report the events within 30 days as required by 10CFR50.73. In each case the MSSVs' lift setpoints were corrected and left operable prior to completion of the Surveillance Test Procedure (STP). The apparent MSSV setpoint drift could have been attributable to two factors, 1) testing method, and; 2) setpoint drift due to valve design/application. The investigation concluded that the old testing method had a high probability of contributing to the apparent MSSV setpoint drift. To prevent recurrence, MSSV setpoints have been tested with an improved testing method. Also, the applicable STP has been modified to ensure that future MSSV failures are promptly reported.

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1/1

Conditions Prior to Occurrence

Unit One in Mode 3 (Hot Stand-By), 0 percent reactor thermal power.
Unit Two in Mode 3 (Hot Stand-By), 0 percent reactor thermal power.

Description of Event

This is a supplemental report to a previously submitted LER, 315/86-20-00.

On August 25, 1986, at 0900 hours during a review of Main Steam Safety Valve (MSSV) (EIIS/SB-RV) setpoint verification documentation, it was discovered that reportability requirements of 10CFR50.73 had not been met. Eight of the twenty Unit 1 MSSV's were found outside of the setpoint range acceptance criteria of Surveillance Test Procedure (STP) **12MHP4030.STP.002 (Main Steam Setpoint Verification - Secondary System Safety Valve Settings) when tested on November 2, 1985. Four of twenty Unit 2 MSSV's also failed to meet this acceptance criteria when tested on June 23 and 24, 1986. As a result of a procedure deficiency, these events were not promptly identified as being reportable. It was also determined that similar surveillance test failures of MSSV's which occurred in October 1983 and July 1984 were not properly identified and reported. Upon discovery of the low setpoint values the MSSV setpoints were immediately adjusted to bring the values within the Technical Specification required range. The action statement for Technical Specification 3.7.1.1 was complied with during the performance of the testing. The main steam headers of each of the four steam generators (EIIS/SB-SG) in both Unit 1 and Unit 2 are equipped with 5 safety valves for a total of 20 valves per unit.

The required relief pressure setpoint range and their as-found conditions for Unit 1 and Unit 2 safety valves are listed below for the valves found out of specification:

Unit 1 (1983)

Valve Serial No.	Valve Identification No.	Steam Generator	Technical Specification Required Range (psi)	As-Found Setpoint (psi)
BN-6305	SV-1B-1	1	1054-1076	1079
BN-6306	SV-1B-2	2	1054-1076	1032
BN-6311	SV-1A-2	2	1054-1076	1046

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)

Unit 2 (1984)

Valve Serial No.	Valve Identification No.	Steam Generator	Technical Specification Required Range (psi)	As-Found Setpoint (psi)
BN-6324	SV-1A-3	3	1054-1076	1078
BN-6327	SV-1B-4	4	1054-1076	1036
BN-6329	SV-1B-3	3	1054-1076	1053
BN-6338	SV-2A-2	2	1064-1086	1038
BN-6341	SV-3-2	2	1074-1096	1072

Unit 1 (1985)

Valve Serial No.	Valve Identification No.	Steam Generator	Technical Specification Required Range (psi)	As-Found Setpoint (psi)
BN-6304	SV-1B-3	3	1054-1076	1081
BN-6305	SV-1B-1	1	1054-1076	1099
BN-6306	SV-1B-2	2	1054-1076	1106
BN-6308	SV-1A-1	1	1054-1076	1087
BN-6312	SV-2B-2	2	1064-1086	1089
BN-6314	SV-2B-1	1	1064-1086	1058
BN-6315	SV-2B-3	3	1064-1086	1054
BN-6321	SV-3-2	2	1074-1096	1132

Unit 2 (1986)

Valve Serial No.	Valve Identification No.	Steam Generator	Technical Specification Required Range (psi)	As-Found Setpoint (psi)
BN-6327	SV-1B-4	4	1054-1076	1052
BN-6334	SV-2A-1	1	1064-1086	1063
BN-6338	SV-2A-2	2	1064-1086	1057
BN-6343	SV-3-1	1	1064-1086	1063

No other structures, components or systems that were related to the events were inoperable at the times of occurrence.

Cause of Event

The apparent MSSV setpoint drift could have been attributed to two factors, 1) testing method, and; 2) setpoint drift due to valve design/application. The investigation concluded that the old testing method was inherently less accurate and had a high probability of contributing to the apparent MSSV setpoint drift.

The failure to report the incorrect readings at the time of the event is considered to be due to a procedure deficiency. The STP lacked reportability instructions in instances where the safety valves failed to be within specifications.

Analysis of Event

The safety valve setpoints discovered in this event were found to be out of compliance with the Technical Specification (T/S) 3.7.1.1 requirements and therefore reportable per 10CFR50-73 (a) (2) (i) (B). The Unit 2 as-found setpoints were below the required values in most cases. The Unit 1 as-found setpoints were mixed. Some were below the Technical Specification requirements, and some were above. The lowest average setting that was found was for SV-1B-2 (BN-6306) at 1032 psi, and the highest was for SV-3-2 (BN-6321) at 1132 psi.

The following FSAR Chapter 14 accident analyses consider secondary-side pressure relief:

1. Loss of External Electrical Load (Appendix 14C.3.6)
2. Loss of Normal Feedwater (Appendix 14C.3.7)
3. Loss of All A.C. Power to the Station Auxiliaries
4. Steam Generator Tube Rupture (14.2.4)
5. Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates the ECCS (Appendix 14E.1)

The maximum relief requirement is associated with the loss-of-load accident. The analysis assumes that the steam dump was not available and that there was no power-operated relief valve actuation.

The deviation from the Technical Specification setpoint is judged to have minimal effect on the ability of the plant to accomodate this accident. The installed relief capacity is approximately 117 percent of the steam generation rate at full power. Thus, with a reactor trip following an accident, it is judged that there would be adequate capacity to relieve the steam generated during the transient. Thus, the pressure reached would be a function of the setpoint plus the accumulation which occurs as the valve opens.

Using the highest as-found setpoint to obtain a conservative peak pressure estimate, it is judged that the peak pressure would have been below the ASME code allowable of 110 percent of design pressure.

The low "as-found" setpoints would have resulted in the early opening of the safety valves during the accident. However, because opening of the safety valves has been considered in the accident analysis, and the magnitude of the deviation was small, it is judged that this would not have adversely affected plant safety.

Corrective Action

The immediate corrective action, as required by **12MHP4030.STP.002, was to reset the safety valves setpoints to within their specified ranges. To prevent recurrence, future MSSV setpoints will be tested with an improved testing method. This will provide a more accurate determination of the MSSV setpoints.

The procedure has been modified to include instructions on reportability. The new instructions include:

- A requirement to notify the Shift Supervisor immediately if a safety valve fails to lift in its specified range. (NOTE: This step is necessary to allow the initiation of any Technical Specification 3/4 7.1.1 action statement requirements as testing is conducted while in Hot Stand-By. The Shift Supervisor will again be notified when the valve is operable.)
- A requirement to initiate a condition report as documentation of any safety valves failing to be in compliance.

Failed Component Identification

Main Steam Safety Valve

Manufacturer: Dresser Consolidated Valves
Model: 3707RA-RT21
EIIS Code: RV

Previous Similar Events

LER 315/87-007-00, Main Steam Safety Valves Out of Specification Due to Apparent Setpoint Drift.

Indiana Michigan
Power Company
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P.O. Box 458
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INDIANA
MICHIGAN
POWER

October 29, 1987

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

Operating License DPR-58
Docket No. 50-315

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:

86-020-1

Sincerely,

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

WGS:afh

Attachment

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