REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8707110025 DOC. DATE: 87/07/02 NOTARIZED: NO DOCKET #
FACIL: 50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316

AUTH. NAME	AUTHOR AFFILIATION
BLIND, A. A.	Indiana & Michigan Electric Co.
SMITH, W. G.	Indiana & Michigan Electric Co.

RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 87-005-00: on 870602, engineered safety features actuation occurred due to low low steam generator level in number 23 Steam Generator. Caused by personnel, error. Personnel implements plant procedure for proper response. W/870702 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 5
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
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INTERNAL:	ACRS MICHELSON	1	1	ACRS MOELLER	2	2
	AEOD/DOA	1	1	AEOD/DSP/ROAB	2	2
	AEOD/DSP/TPAB	1	1	DEDRO	1	1
	NRR/DEST/ADE	1	0	NRR/DEST/ADS	1	0
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	NRR/DEST/MTB	1	1	NRR/DEST/PSB	1	1
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	NRR/DREP/RPB	2	2	NRR/PMAS/ILRB	1	1
	NRR/PMAS/PTSB	1	1	REG FILE 02	1	1
	RES DEPY GI	1	1	RES TELFORD, J	1	1
	RES/DE/EIB	1	1	RGN3 FILE 01	1	1
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
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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) ESF Actuation (Reactor Trip) Due to Low-Low Steam Generator Level Resulting from Personnel Error

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 6	0 2	8 7	8 7	— 0 0 5 —	0 0	0 7	0 2	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 2
POWER LEVEL (10): 0 0 5

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)	[X] 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)	[ ] 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)
A. A. Blind - Assistant Plant Manager	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

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)

On June 2, 1987, at 1949 hours, an Engineered Safety Features actuation (Reactor Trip) occurred due to low-low Steam Generator level in number 23 Steam Generator. Prior to the trip, the reactor had been stable in Mode 2 (Reactor Startup) at 4.5 percent Reactor Thermal Power. Steam Generator levels were being maintained in manual utilizing low flow feedwater preheating.

Due to the operator assigned the responsibility to maintain Steam Generator levels becoming involved with other related Control Room activities, Steam Generator number 23 dropped substantially in level. In an attempt to avoid a trip actuation, the operator increased steam flow via the steam dump system while simultaneously increasing feed-water flow. Steam Generator level increased momentarily, however, shortly thereafter level dropped and at 1949 hours the low-low level setpoint was reached actuating the reactor trip sequence.

The cause of this event was determined to be personnel error. Appropriate administrative action was taken concerning the individual involved. In addition, a memo was issued to all operators describing lessons learned from this event.

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Conditions Prior to Occurrence

Unit 2 in Mode 2 (reactor startup) at 4.5 percent Reactor Thermal Power.

Description of Event

On June 2, 1987, at 1949 hours, an Engineered Safety Features (ESF) Actuation (Reactor Trip) occurred due to low-low Steam Generator (EIIS/SG) level in number 23 Steam Generator. Prior to the trip, the reactor had been stable in Mode 2 at 4.5 percent Reactor Thermal Power. Steam Generator levels were being maintained in manual utilizing low flow feedwater preheating. This is the standard means used to control Steam Generator levels during low power levels. As part of Steam Generator level control (and to provide a means of improving secondary system chemistry) the blowdown system (EIIS/WI) is placed in service via the startup flash tank (EIIS/TK). While the necessary valve lineups were being completed, it was determined that two of the blowdown system containment isolation valves (EIIS/ISV) were not indicating full open as expected. The Unit Supervisor (Licensed Reactor Operator) who was aligning these valves, notified the auxiliary equipment operator (non-licensed) and requested that the valve limit switches (EIIS/33) be checked to verify actual valve position. The auxiliary equipment operator called the Control Room feedwater operator (Licensed Reactor Operator), who was concerned with placing the blowdown system in service, to discuss the containment isolation valve situation.

During this communication, Steam Generator numbers 23 and 24 dropped substantially in level. By the Time the feedwater operator noticed the levels had dropped, they were low enough that a significant increase in feedwater flow would have tripped the Unit on low-low Steam Generator level (due to shrink). In an attempt to avoid the trip actuation, the feedwater operator increased steam flow via the steam dump system while simultaneously increasing feedwater flow. Both Steam Generators 23 and 24 increased in level momentarily, however, shortly thereafter level continued to drop and at 1949 hours the low-low level setpoint of 19 percent was reached in Steam Generator number 23, actuating the reactor trip sequence [opening of the reactor trip breakers (EIIS/BKR), insertion of reactor control rods (EIIS/ROD), feedwater isolation, automatic starting of the motor driven auxiliary feedpumps (EIIS/P)]. There was no automatic or manual actuation of the intermediate head safety injection system (EIIS/BQ).

Operations personnel immediately implemented special emergency operating procedure 2-OHP-4023.E-0 to verfiy proper response of the automatic protection system (EIIS/JC) and to assess plant conditions for initiating appropriate recovery action.

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

The Unit was stabilized in Mode 3 (Hot Standby) at approximately 2020 hours, June 2, 1987. The NRC was notified of the event via the ENS at 2040 hours June 2, 1987.

With the exception of the two containment isolation valvelimit switches, there were no inoperative structures, components, or systems that contributed to this event.

Cause of Event

The cause of this event was determined to be personnel error. The feedwater operator (Licensed Reactor Operator) was not maintaining sufficent control of the Steam Generator water levels. Even though aligning the blowdown system is related to maintaining Steam Generator levels, the operator involved allowed himself to become occupied with the containment isolation valve concerns and did not ensure Steam Generator levels were being maintained within proper limits.

Analysis of Event

This Engineered Safety Features actuation which resulted in a reactor trip is reportable pursuant to 10 CFR 50.73 (a)(2)(iv).

The automatic protection system responses; reactor trip, and resultant actuations, were all verified to have functioned properly as a result of the Engineered Safety Features actuation. Based on the above, it is concluded that the health and safety of the public were not affected.

It was noted however, that the Operation Sequence Monitor (OSM) failed to print the correct or accurate equipment actuation times, consequently a complete time study was not possible. Through secondary means, it was determined that the reactor trip breakers actuated in less than 100 ms which is consistent with previous data. All other applicable equipment actuations were also verified. The OSM was repaired, verified operable, and returned to service on June 3, 1987.

Corrective Actions

Immediate corrective action involved Operations personnel implementing plant procedures to verify proper response of the automatic protection system and to assess plant conditions for initiating appropriate recovery actions. Appropriate administrative action was taken concerning the individual involved. In addition, a memo was issued to all operators describing lessons learned from this event.

★U.S.GPO:1986-0-624-538/455

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

Failed Component Identified

No component failures were directly related to the cause of this event.

Previous Similar Events

50-315/86-018-00
50-315/86-015-00
50-316/85-002-00
50-316/85-001-00

American Electric Power System

INDIANA & MICHIGAN ELECTRIC COMPANY
DONALD C. COOK NUCLEAR PLANT
P.O. Box 458, Bridgeman, MI 49106
Telephone (616) 465-5901

July 2, 1987

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:

87-005-00

Sincerely,

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

WGS/afh

Attachment

cc: A. B. Davis, Region III
D. L. Wigginton, NRC
J. E. Dolan
M. P. Alexich
R. F. Kroeger
H. B. Brugger
R. W. Jurgensen
G. Charnoff, Esq.
R. C. Callen, MPSC
D. Hahn, MDPH
INPO
NRC Resident Inspector
Dottie Sherman, ANI Library
A. A. Blind
J. G. Feinstein/B. P. Lauzau
PNSRC
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