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ACCESSION NBR: 8708310112 DOC. DATE: 87/08/21 NOTARIZED: NO DOCKET #
FACIL: 50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316

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SUBJECT: LER 87-008-00: on 870722, ESF actuation occurred due to extreme high level in steam generator 24. Caused by erratic response of main feedwater pump delta P controller. Plant procedures implemented. W/870821 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 6
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

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	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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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) ESF Actuation (Reactor Trip) Due to Extreme High Steam Generator Level As a Result of Component Failure

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
0 7	2 2	8 7	8 7	0 0 8	0 0	0 8	2 1	8 7		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 1 9

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)
J.R. Sampson - Safety and Assessment 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 J	D M T R	F 1 8 0	N					

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 July 22, 1987, at 2232 hours, an engineered safety features (ESF) actuation (reactor trip sequence) occurred due to an extreme high level in the No. 24 Steam Generator. Prior to the trip, a unit power ascension was in progress, Steam Generator levels were being maintained in manual utilizing low flow feedwater preheating. The main feedwater delta pressure (Delta P) controller was in automatic and responding as designed with no apparent difficulties.

At approximately 2210 hours, an occurrence involving the erratic response (intermittent failure) of the feedwater pump delta P controller meter resulted in a severe transient in the Steam Generators. The resultant feedwater transient increased the level within the No. 24 Steam Generator to the extreme high trip setpoint actuating the reactor trip sequence.

The main feedwater delta P controller was inspected/tested immediately following the event. No faulty components were identified and the response/failure experienced could not be reproduced - however, as a preventive measure, the delta P controller meter was replaced. In addition, a memo was issued to all Operators describing the lessons learned from this event.

8708310112 870821
PDR ADOCK 05000316
S PDR

NRC Form 366A (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

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 7	— 0 0 8	— 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 1 (Power Operation) at 19 percent reactor thermal power (RTP) with a unit power ascension in progress.

Description of Event

On July 22, 1987, at 2232 hours, an engineered safety features (ESF) actuation (reactor trip sequence) occurred due to an extreme high level in the No. 24 Steam Generator (EIIS/AB-SG).

Prior to the trip, 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 operation. The main feedwater delta pressure (Delta P) controller (EIIS/SJ-PC) was in automatic and responding as designed with no apparent difficulties. At 2154 hours, with the reactor (EIIS/AB-RCT) at approximately 8 percent RTP and the Main Turbine (EIIS/TA-TRB) at 1800 RPM, preparations were made to parallel the Unit 2 Main Generator (EIIS/TB-GEN). Once the Main Generator was parallel to the grid, reactor power was increased steadily with no secondary control problems identified. As power ascension continued, it was noted (at approximately 16 percent RTP) that the main feed pump (EIIS/SJ-P) speed was not increasing and Steam Generator levels were lower than expected. Investigation revealed that the main feedwater pump delta P controller meter (EIIS/SJ-DMTR) had stuck at approximately two increments on the low side of the nulled position. The Assistant Shift Supervisor (Utility - Licensed Operator) was informed of this situation. He promptly placed the delta P controller in manual and began increasing the pump speed supplying the generators with the "requested" feedwater flow. Once levels were stable, the delta P controller was returned to automatic and continually monitored. Following the malfunction of the controller, the Unit Supervisor (US) (Utility - Licensed Operator) informed the Reactor Operator (RO) (Utility - Licensed Operator) that the reactor core axial flux difference was shifting to positive. At approximately 2210 hours, the US instructed the RO to stop the power increase, dilute the Reactor Coolant System (EIIS/AB) and insert the reactor control rods (EIIS/AA-ROD) into the core to reduce axial flux difference. This attempt to restore the axial flux difference, combined with another simultaneous occurrence involving the erratic response of the feedwater pump delta P controller meter, resulted in a severe level transient in the Steam Generators. The delta P controller was again placed in manual, however, the resultant feedwater transient increased the level within the No. 24 Steam Generator to the extreme high trip setpoint actuating the reactor trip sequence [opening of the reactor trip breakers (EIIS/JE-BKR), insertion of the reactor control rods, feedwater isolation (EIIS/JB), automatic starting of the motor driven and turbine driven auxiliary feedwater pumps (EIIS/BK-P)].

★U.S.GPO:1986-0-624-538/455

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 7	— 0 0 8	— 0 0	0 3 OF 0 4

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

Description of Event Con't

Operations personnel immediately implemented the emergency operating procedure, 1-OHP 4023.E-0, to verify proper response of the automatic protection system (EIIS/JC) and to assess Plant conditions for initiating appropriate recovery actions. There was no automatic or manual actuation of the intermediate head safety injection system (EIIS/BQ).

The unit was stabilized in Mode 3 (Hot Standby) at approximately 2258 hours, July 22, 1987. The NRC was notified of the event via the ENS at 2233 hours, July 22, 1987.

With the exception of the erratic response of the main feedwater pump delta P controller, 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 the erratic response (intermittent failure) of the main feedwater pump delta P controller. The reduction in the power ascension rate, due to reactor core axial flux difference, also contributed to the initial Steam Generator level transient - however, if the delta P controller had responded as designed, no adverse affects would have resulted.

Analysis of Event

This engineered safety features actuation, which resulted in a reactor trip sequence, is reportable pursuant to 10 CFR 50.73(a)(2)(iv).

The Operations sequence monitor functioned as designed. A time study of parameters monitored concluded that all automatic protection system responses; reactor trip and resulting actuations, functioned properly as a result of the engineered safety feature actuation.

Based on the above, it is concluded that the event did not constitute an unreviewed safety questions as defined in 10 CFR 50.59(a)(2), nor did it adversely impact health and safety.

*U.S.GPO:1986-0-624-538/455

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 7	— 0 0 8	— 0 0	0 4 OF 0 4

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

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. The main feedwater delta P controller was inspected/tested immediately following the event. No faulty components were identified and the response/failure experienced could not be reproduced - however, as a preventive measure, the delta P controller meter was replaced. In addition, a memo was issued to all operators describing the lessons learned from this event.

Failed Component Identification

It could not be conclusively determined if the following component failed during/contributed to the subject event. It was replaced, however, as a preventive measure.

Plant Description: Main Feedwater Pump Delta Pressure Controller Meter

Manufacturer: Foxboro

Manufacturer I.D. Number: N0196SC

EIIS Code: SJ-DMTR

Previous Similar Events

None - no previous similar events involving the malfunction of the Main Feedwater Pump Delta Pressure Controller were identified.

American Electric Power System INDIANA & MICHIGAN ELECTRIC COMPANY
Donald C. Cook Nuclear Plant
P.O. Box 458, Bridgman, Michigan 49106

August 21, 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:

87-008-00

Sincerely,

W. G. Smith, Jr.
Plant Manager

/afh

Attachment

cc: John E. Dolan
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1/1