

# REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8707130305 DOC. DATE: 87/07/02 NOTARIZED: NO DOCKET #  
 FACIL: 50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315  
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 BLIND, A. A. Indiana & Michigan Electric Co.  
 SMITH, W. G. Indiana & Michigan Electric Co.  
 RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 87-008-00: on 870604, ESF actuation occurred due to  
 extreme high level in steam generator 12. Caused by  
 mispositioning of both level controller & feedwater flow  
 selector switch. Level controller refurbished. 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:

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	AEOD/DOA	1 1	AEOD/DSP/ROAB	2 2
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	NRR/DEST/ADE	1 0	NRR/DEST/ADS	1 0
	NRR/DEST/CEB	1 1	NRR/DEST/ELB	1 1
	NRR/DEST/ICSB	1 1	NRR/DEST/MEB	1 1
	NRR/DEST/MTB	1 1	NRR/DEST/PSB	1 1
	NRR/DEST/RSB	1 1	NRR/DEST/SGB	1 1
	NRR/DLPQ/HFB	1 1	NRR/DLPQ/QAB	1 1
	NRR/DOEA/EAB	1 1	NRR/DREP/RAB	1 1
	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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## LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) D.C. Cook Nuclear Plant, Unit 1										DOCKET NUMBER (2) 0 5 0 0 0 3 1 5 1					PAGE (3) 1 OF 4									
TITLE (4) ESF Actuation (Reactor Trip) Due to extreme High Steam Generator Level Resulting from Mispositioned Control Switches																								
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	4	8	7	8	7	0	0	8	0	0	0	7	0	2	8	7	0	5	0	0	0	0
OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)																						
1		20.402(b)				20.405(e)				<input checked="" type="checkbox"/> 50.73(a)(2)(iv)				73.71(b)										
POWER LEVEL (10)		0 9 0				50.38(e)(1)				50.73(a)(2)(v)				73.71(c)										
		20.405(a)(1)(i)				50.38(e)(2)				50.73(a)(2)(vi)				OTHER (Specify in Abstract below and in Text, NRC Form 368A)										
		20.405(a)(1)(ii)				50.73(a)(2)(i)				50.73(a)(2)(vii)(A)														
		20.405(a)(1)(iii)				50.73(a)(2)(ii)				50.73(a)(2)(vii)(B)														
		20.405(a)(1)(iv)				50.73(a)(2)(iii)				50.73(a)(2)(viii)														
		20.405(a)(1)(v)				50.73(a)(2)(iv)				50.73(a)(2)(ix)														
LICENSEE CONTACT FOR THIS LER (12)																								
NAME A. A. Blind - Assistant Plant Manager										TELEPHONE NUMBER 6 1 64 6 5 1 5 9 0 1														
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)																								
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPDOS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPDOS														
SUPPLEMENTAL REPORT EXPECTED (14)												EXPECTED SUBMISSION DATE (15)		MONTH	DAY	YEAR								
<input type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE)												<input checked="" type="checkbox"/> NO												

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

On June 4, 1987, at 1034 hours, an ESF actuation (turbine/reactor trip) occurred due to an extreme high level in Steam Generator number 12. The level transient occurred during the performance of a test procedure. During the execution of the test, an operator was directed to place the automatic/manual level controller in the manual position, then transfer the steam flow and feedwater flow selector switches from the Channel I position to the Channel II position. Once transferred, the level controller can be placed in the automatic position and testing can commence on Channel I.

The cause of the event was the mispositioning of both the level controller (due to malfunction) and the feedwater flow selector switch (due to misleading switch position labeling). This resulted in an increase in feedwater pump speed, the automatic opening of the Steam Generator number 12 feedwater flow control valve and subsequent extreme high Steam Generator level.

The level controller was removed, inspected, refurbished and determined to be operable. The escutcheon plate on the feedwater flow selector switch has been relabeled to identify the correct position. In addition, a memo was issued to all operators describing lessons learned from this event.

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## LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104

EXPIRES: 8/31/88

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
D.C. Cook Nuclear Plant, Unit 1	0 5 0 0 0 3 1 5	8 7	— 0 0 8	— 0 0	0 2	OF	0 4

TEXT (If more space is required, use additional NRC Form 368A's) (17)

Conditions Prior to Occurrence

Unit 1 in Mode 1 (Power Operation) at 90 percent Reactor Thermal Power.

Description of Event

On June 4, 1987, at 1034 hours, an Engineered Safety Features actuation (turbine trip/reactor trip) occurred due to a extreme high level in Steam Generator (EIIS/SG) number 12. The Steam Generator level transient occurred during the performance of a surveillance test procedure, 1-THP-4030-STP.019 (Steam Generator 1 and 2 mismatch set number 1). During the execution of the test, the control room operator is directed to place the automatic/manual level controller (EIIS/LC) in the manual position, then transfer both the steam flow and feedwater flow selector switches (EIIS/33) from the "Channel I" position to the "Channel II" position. Once transferred, the automatic/manual level controller can be placed in the automatic position and testing can commence on channel I.

When the control room operator (Licensed Reactor Operator) acted to place the automatic/manual level controller in manual, the switch did not fully travel to the manual position detent (the switch is spring loaded to the manual position and should have traveled to manual once it was moved, even slightly, from the automatic detent position). Consequently the controller remained in automatic, unknown to the operator. The operator then rotated steam flow selector switch from channel I to channel II. No adverse effects resulted from this transfer. The operator then acted to transfer the feedwater flow selector switch from channel I to channel II. It was later determined that the selector switch was not fully rotated to the channel II position due to misleading switch position labeling and therefore neither the channel I or channel II output was selected. With the level controller still in automatic, it compared a full power steam flow signal with a zero feedwater flow signal. This resulted in an increase in feedwater pump (EIIS/P) speed and the automatic opening of the steam generator number 12 feedwater flow control valve (EIIS/FCV).

The ensuing transient occurred so rapidly the actions taken by the operator to regain control of steam generator level were ineffective [this was complicated by the fact that there was an erroneous low steam generator level indication available to the operator due to a mechanical interference within the steam generator level recorder (EIIS/LR)] and a turbine trip/reactor trip sequence was initiated [opening of the reactor trip breakers (EIIS/BKR), insertion of reactor control rods (EIIS/ROD), feedwater isolation, automatic starting of the motor-driven and turbine-driven auxiliary feedwater pumps (EIIS/P)]. There was no automatic or manual actuation of the intermediate head safety injection system.

## LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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APPROVED OMB NO. 3150-0104

EXPIRES: 8/31/88

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TEXT (If more space is required, use additional NRC Form 368A's) (17)

Operations personnel immediately implemented the special 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.

The Unit was stabilized in Mode 3 (Hot Standby) at approximately 1100 hours, June 4, 1987. The NRC was notified of the event via the ENS at 1120 hours, June 4, 1987.

With the exception of the malfunction of the level controller switch mechanism and the steam generator level recorder, there were no inoperative structures, components, or systems that contributed to this event.

#### Cause of Event

Several unrelated items (both human factors related) combined to result in this event: 1) the automatic/manual level controller transfer switch malfunctioned which prevented typical (spring assisted) travel to the manual position, and 2) the feedwater flow selector switch was not fully rotated to the channel II position [The reason this did not occur is twofold. First, the escutcheon plate on the subject selector switch incorrectly identified the channel II position at 45 degrees from channel I (in reality the channel II position is 90 degrees from channel I). Secondly, the subject selector switch is the only switch with the channel II position at 90 degrees, the channel II position on all corresponding selector switches is at the 45 degree position.

#### 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; turbine trip/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 control room temperature indication, the chart drive for the wide range reactor coolant system temperature recorders, and the Unit Megawatt Recorder malfunctioned. Further investigation revealed that these abnormal responses were due to a broken wire on a critical control room power circuit breaker. This broken wire was the result of a cable pull

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

operation which took place approximately 15 minutes prior to the reactor trip actuation. In addition, 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 actuations were also verified.

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 automatic/manual level controller was removed, inspected, refurbished, and determined to be operable. The steam generator number 12 level recorder was repaired and returned to service. The escutcheon plate for the number 12 Steam Generator feedwater flow selector switch has been relabeled to identify the required 90 degree throw. In addition, a memo was issued to all operators describing lessons learned from this event. Corrective action has been taken on all attendant equipment malfunctions noted in the event analysis section.

Failed Component Identification

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

Previous Similar Events

None.



**INDIANA & MICHIGAN ELECTRIC COMPANY**

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July 2, 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:

87-008-00

Sincerely,

  
W. G. Smith, Jr.  
Plant Manager

WGS:afh

Attachment

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