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FACIL:50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
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SUBJECT: LER 89-007-00:on 890310,steam generator lo-lo level reactor trip signal due to personnel error.

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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 3

TITLE (4) Steam Generator Low-Low Level Reactor Trip Signal Due to Personnel Error During Cooldown With Reactor Trip Breakers Open

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 3	1 0	8 9	8 9	— 0 0 7	— 0 0	0 4	0 3	8 9		0 5 0 0 0

OPERATING MODE (9)	3	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)					
POWER LEVEL (10)	0 0 0	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
H. F. Runser, Operations 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	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	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 March 10, 1989, at 1738, a reactor trip signal was received due to a low-low steam generator (SG) level on number 21 SG during a cooldown using steam dumps to the condensers. Prior to the trip signal, a high SG blowdown rate was established for chemistry cleanup. The reactor operator controlling the SG levels did not adequately anticipate the amount of auxiliary feedwater which would be required to accommodate the cooldown and the blowdown. The auxiliary feedwater flow was increased significantly when level was close to the trip setpoint, but the increase in flow was not soon enough and the resultant shrink due to the cold water dropped the level to the trip setpoint.

The reactor trip breakers were open prior to the cooldown evolution. After the trip the opening of the motor operated auxiliary feedwater valves, the only anticipated automatic feature, performed as designed.

This event was caused by the failure of the Reactor Operator to adequately anticipate the required auxiliary feedwater flowrate and failure to take timely action to correct the situation or to notify other Control Room personnel of the situation. The involved Reactor Operator has been counseled on this event.

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PDR ADOCK 05000316
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D. C. Cook Nuclear Plant - Unit 2	0 5 0 0 0 3 1 6	8 9	— 0 0 7	— 0 0	0 2 OF 0 3

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

Conditions Prior to Occurrence

Unit Two in Mode 3 (Hot Standby).

Description of Event

On March 10, 1989, at 1738, a reactor trip signal was received due to a low-low steam generator (SG) (EIIS/AB-SG) level on number 21 SG during a cooldown using condenser steam dumps (EIIS/SB-FCV).

Prior to the cooldown, an extensive job briefing was given by the Unit Supervisor (senior licensed operator) to the three Reactor Operators (licensed operators) who were to perform the cooldown. Among other things, the briefing included instructions to use good communications, make all changes slowly and to ask for help if any problems were encountered.

A cooldown rate of 30°F/hour was established using the condenser steam dumps with SG levels at about 44%. Chemistry personnel then requested that the SG blowdown (EIIS/WI) rate be increased for cleanup. The SG blowdown lineup was then changed from the normal to startup alignment. The Reactor Operator (RO) controlling SG levels then initiated a high blowdown rate on all four steam generators. The auxiliary feedwater (AFW) (EIIS/BA) flow rate was then increased some, but the levels started to drop. The RO thought that the level decrease would stop, so he observed the levels for a while without increasing AFW flow rate or reducing the SG blowdown rate.

The levels continued to decrease and the RO controlling the cooldown prompted the RO controlling the SG levels to increase the AFW flow rate. The blowdown from number 21 SG was stopped and the AFW flow rate was increased. The increased AFW flow rate was not initiated soon enough. The resultant shrink from the cold AFW dropped the number 21 SG level to the low-low level trip setpoint of 21%.

Prior to the cooldown evolution the reactor trip breakers (EIIS/JE-BKR) were open. Upon receipt of the reactor trip signal the AFW motor operated valves (EIIS/BA-ISV) traveled full open as designed and then throttled to the flow retention position. This was the only automatic actuation which should have occurred under the existing plant conditions. Operations personnel verified proper system response and plant conditions. There were no equipment or instrument malfunctions during this event.

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

Cause of Event

This event was caused by personnel error. The RO controlling the SG levels did not adequately anticipate the required AFW flow rate and failed to take timely action to correct the situation. The RO also failed to notify other Control Room personnel of the difficulty which was being encountered.

Analysis of Event

This event is being reported in accordance with 10 CFR 50.73 (a)(2)(iv) as an event that resulted in an unplanned automatic actuation of an Engineered Safety Feature including the Reactor Protection System.

The automatic protection system responses were verified to have functioned properly as a result of the engineered safety features actuation. Based on the above, it is concluded that the event did not constitute an unreviewed safety question as defined by 10 CFR 50.59(a)(2) nor did it adversely impact the health and safety of the public.

Corrective Action

Immediate corrective action involved verifying proper system response and plant conditions. The SG levels were then restored to the proper levels.

The RO involved in this event has been counseled on the need to take timely actions and to communicate when difficulties are encountered.

Failed Component Identification

None

Previous Similar Events

No previous similar events involving failure to properly control SG levels during a cooldown evolution.

Indiana Michigan
Power Company
Cook Nuclear Plant
P.O. Box 458
Bridgman, MI 49106
616 465 5901

INDIANA MICHIGAN POWER

April 3, 1989

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:

89-007-00

Sincerely,

W. G. Smith, Jr.
Plant Manager

WGS:clw

Attachment

cc: D. H. Williams, Jr.
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