# ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8904060351 DOC.DATE: 89/03/26 NOTARIZED: NO DOCKET #
FACIL:50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 89-006-00:on 890224,ESF actuation due to steam flow feedwater flow mismatch coincident w/low SG level signal.

DISTRIBUTION CODE: IE22D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 4
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

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| | H ST LOBBY WARD | 1 | 1 | LPDR | 1 | 1 |
| | NRC PDR | 1 | 1 | NSIC MAYS,G | 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 3 |

TITLE (4) ENGINEERED SAFETY FEATURES ACTUATION (REACTOR TRIP) DUE TO STEAM FLOW FEEDWATER FLOW MISMATCH COINCIDENT WITH LOW STEAM GENERATOR LEVEL SIGNAL

| 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 2 | 2 4 | 8 9 | 8 9 | — 0 0 6 — | 0 0 | 0 3 | 2 6 | 8 9 | | 0 5 0 0 0 |

OPERATING MODE (9): 5

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) | 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 | |
| T. P. BEILMAN<br>INSTRUMENTATION AND CONTROL DEPARTMENT 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 |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |
| | | | | | | | | | |

| 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 February 24, 1989 at 0907 hours, an engineered safety features (ESF) actuation (rector trip sequence) occurred from instrumentation indicating a low level coincident with feedwater flow/steam flow mismatch on Steam Generator Number 24. Prior to the event, the reactor was in Mode 5 and preparations were being made for startup. There was no actual feedwater or steam flow at this time and many feedwater flow/steam flow mismatch trip signals were present because fill of the flow transmitter reference legs had not been completed yet. Steam Generator Number 24 level was steady at 35 percent. At the time of the reactor trip, the recorded steam generator level indicated a sudden decrease, then increase of level. The rate and extent of the indicated level change is not considered to be physically possible. The instrument channel which supplies the level recorder does not supply the low level trip, therefore two independent channels of Class 1E redundant level instrumentation for Steam Generator Number 24 must have sensed a level change. Instrumentation loops were checked with no problems found. An exact cause for the change of output in the steam generator level instrumentation could not be determined. We postulate that a pressure change in the steam generator occurred, which the transmitters sensed. The instrumentation was monitored closely through the subsequent unit startup and regular surveillance tests performed with all channels responding normally.




IE22

## Conditions Prior To Occurrence

Unit Two in Mode 5 (cold shutdown).

## Description of Event

On February 24, 1989 at 0907 hours, an engineered safety features (ESF) actuation (rector trip sequence) occurred from instrumentation indicating a low level coincident with feedwater flow/steam flow mismatch on Steam Generator Number 24 (EIIS/SG).

Prior to the event, the reactor was in Mode 5 and preparations were being made for startup. The Reactor Coolant System (RCS) was at half loop level with an average temperature of 130 degrees Fahrenheit. There was no actual feedwater or steam flow at this time and many feedwater flow/steam flow mismatch trip signals were present because fill of the flow transmitter (EIIS/FT) reference legs had not been completed yet. This is a normal condition for the plant during start up preparations. Steam Generator Number 24 level was steady at 35 percent. The reactor trip breakers were closed at 0814 hours in order to reset turbine trips for turbine testing.

At the time of the reactor trip, the recorded steam generator level (EIIS/FR) indicated a sudden decrease, then increase of level. The rate and extent of the indicated level change is not considered to be physically possible. The channel which is connected to the level recorder is not the same channel that supplies the low level trip, therefore two independent channels of Class 1E redundant level instrumentation for Steam Generator Number 24 must have sensed a level change. Instrumentation loops were checked with no problems found.

The reactor trip sequence was limited to opening of the reactor trip breakers (EIIS/BKR). Reactor control rods (EIIS/ROD) were not withdrawn prior to the event. There was no automatic or manual actuation of the emergency core cooling system (EIIS/EQ). The NRC was notified of the event via the ENS at 1205 hours.

## Cause of the Event

An exact cause for the change of output in the steam generator level instrumentation could not be determined. We postulate that a pressure change in the steam generator occurred which the transmitters sensed. The pressure change might have occurred if pressure or vacuum (from unvented level changes, or nitrogen cover, etc.) was relieved through secondary system manipulations, possibly while preparing the turbine for testing.


*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 9 | — 0 0 6 — | 0 0 | 0 3 OF 0 3 |

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

Analysis of Event

This engineered safety features actuation which resulted in a reactor trip is reportable pursuant to 10CFR50.73(a)(2)(iv).

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

Corrective Action

Immediate corrective actions 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. Instrumentation loops were checked with no problems found. The instrumentation was monitored closely through the subsequent startup and regular surveillance tests performed with all channels responding normally.

Failed Component Identification

No component failures were found to be the cause of this event.

Previous Similar Events

None.

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

AEP

INDIANA
MICHIGAN
POWER

March 27, 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:

89-006-00

Sincerely,

W. G. Smith, Jr.
Plant Manager

WGS:clw

Attachment

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