REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8704140250 DOC. DATE: 87/04/07 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 86-026-00: on 860912, determined that valve alignment used placed unit in unanalyzed condition per LOCA analysis in FSAR. Caused by misunderstanding of Tech Spec 3.5.2 re ECCS operability. Administrative controls created. W/870407 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 6
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
	PD3-3 LA	1	1	PD3-3 PD	1	1
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	ACRS WYLIE	1	1	AEOD/DOA	1	1
	AEOD/DSP/ROAB	2	2	AEOD/DSP/TAPB	1	1
	NRR/ADT	1	1	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/EPB	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 SPEIS, T	1	1	RGN3 FILE 01	1	1
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
	LPDR	1	1	NRC PDR	1	1
	NSIC HARRIS, J	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): D. C. Cook Nuclear Plant - Unit Two

DOCKET NUMBER (2): 0 5 0 0 0 3 1 6

PAGE (3): 1 OF 0 5

TITLE (4): Lack of Specificity in Technical Specification Requirements Resulted In Operation With Unanalyzed Emergency Core Cooling System Configuration

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 9	0 4	8 6	8 6	0 2 6	0 1	0 4	0 7	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 8 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(a)(1)(i)
- [ ] 20.405(a)(1)(ii)
- [ ] 20.405(a)(1)(iii)
- [ ] 20.405(a)(1)(iv)
- [ ] 20.405(a)(1)(v)
- [ ] 20.405(c)
- [ ] 50.36(c)(1)
- [ ] 50.36(c)(2)
- [ ] 50.73(a)(2)(i)
- [X] 50.73(a)(2)(ii)
- [ ] 50.73(a)(2)(iii)
- [ ] 50.73(a)(2)(iv)
- [ ] 50.73(a)(2)(v)
- [ ] 50.73(a)(2)(vii)
- [ ] 50.73(a)(2)(viii)(A)
- [ ] 50.73(a)(2)(viii)(B)
- [ ] 50.73(a)(2)(x)
- [ ] 73.71(b)
- [ ] 73.71(c)
- [ ] OTHER *(Specify in Abstract below and in Text, NRC Form 366A)*

LICENSEE CONTACT FOR THIS LER (12)

NAME: K. R. Baker, Operations Superintendent

TELEPHONE NUMBER: AREA CODE 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)

- [ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)*
- [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On September 12, 1986, during the review of Unit Two Safety Injection equipment outage activities a tentative determination was made that the valve alignment used placed the plant in an unanalyzed condition in respect to the Loss of Coolant Accident Analysis found in the Plant's Final Safety Analysis Report (FSAR). In addition, the investigation conducted after this event determined that past surveillance practices also placed the Emergency Core Cooling System (ECCS) in a configuration contrary to the FSAR.

We believe the cause of the event was a legitimate misunderstanding of Technical Specification (T/S) 3.5.2 regarding ECCS operability. We had assumed that this T/S was written to bound the safety analyses. The evaluation for this event has determined that the limits of 10 CFR 50.46 would not have been exceeded. The evaluation for impact of past surveillance practices on both Units is provided in revision one to LER 315-86-021 which was written as a result of the investigation for this LER.

Administrative Controls have been placed on the cross-tie valves and other identified ECCS valves on both Units to prevent undesired isolation.

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

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

Conditions Prior to Occurrence

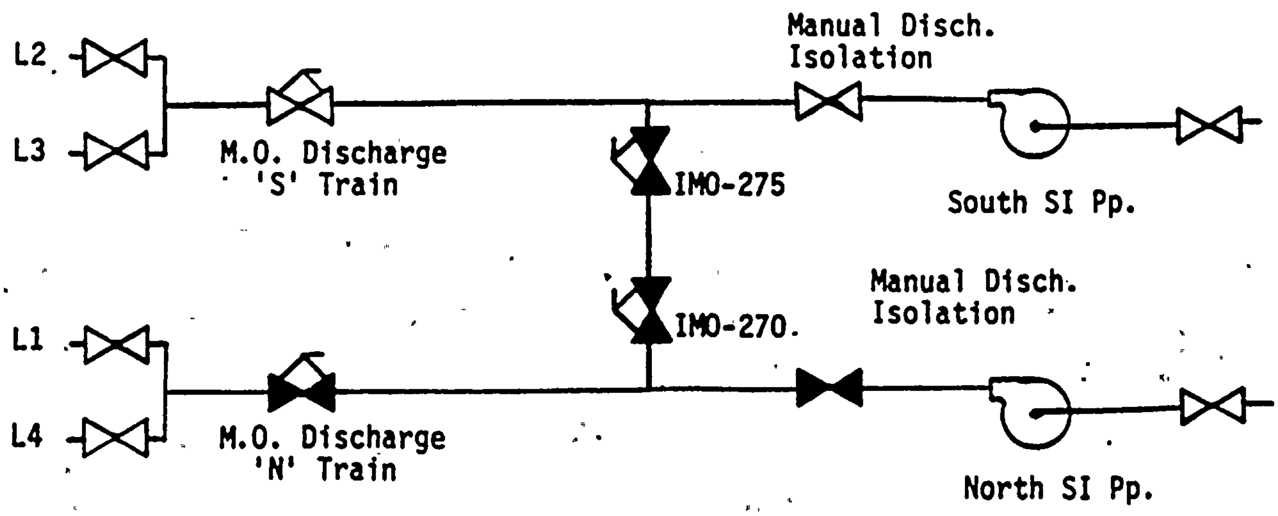
Unit 2 - Mode 1 (Power Operation) - 80 percent reactor thermal power.

Description of Event

On September 12, 1986, during the review of Unit Two Safety Injection (EIIS-BQ) equipment outage activities, a tentative determination was made that the alignment utilized to perform the work placed the plant in an unanalyzed condition in respect to the Loss of Coolant Accident Analysis found in the Plant's Final Safety Analysis Report.

The Safety Injection system outage began on September 4, 1986, at 0645 hours and lasted for a period of 18 hours and 53 minutes. At this time Unit Two was in Mode 1 (Power Operation) operating at 80 percent reactor thermal power. The purpose of this outage was to repair a body to bonnet leak on IMO-270, one of the two Safety Injection discharge cross-tie valves (EIIS-MOV). This required the isolation of both the motor operated (EIIS-MOV) and manual discharge isolation valves (EIIS-ISV) for the North Safety Injection Pump (EIIS-BQP) and the remaining discharge cross-tie (IMO-275). As a result the injection points to two of the four loops was lost.

An informational call concerning this event was made to the NRC via ENS at 1400 hours on September 12, 1986. There were no inoperative structures, components or systems that contributed to this event.



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

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

The investigation of this event determined that prior to May, 1985, surveillance testing on the Emergency Core Cooling System (ECCS) also resulted in system configurations contrary to the Final Safety Analysis Report. Testing on the ECCS was performed during this time in accordance with the Technical Specification surveillance requirements.

An information call concerning this additional finding was made to the NRC via ENS at 1114 hours on October 2, 1986. LER 315-86-21 was written to address the past surveillance practices.

Cause of the Event

Technical Specification 3.5.2 states that "two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,
b. One OPERABLE safety injection pump,
c. One OPERABLE residual head removal heat exchanger,
d. One OPERABLE residual heat removal pump,
e. One OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation."

The Technical Specification statement allows one ECCS subsystem to be inoperable for up to 72 hours. This implies that there are two independent ECCS subsystems or trains with independent flow paths. We assumed that the Technical Specifications were written to bound the safety analyses; however since the safety analyses assumed four injection points were available, our assumption was not valid.

Plant maintenance and testing procedures were written with the understanding that the ECCS consists of two independent subsystems. We believe this to be a legitimate misunderstanding of the requirements of Technical Specifications. Further, from discussions with personnel from other nuclear plants and review of IE Notice 87-01 "RHR Valve Misalignment Causes Degradation of ECCS in PWRs", we believe this type of misunderstanding to be common within the PWR portion of the nuclear industry.

Analysis of Event

This event was considered reportable under the criteria set forth in 10 CFR 50.73 (a)(2)(ii).

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

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

As a result of the isolation boundary established for required maintenance being performed on one of the Safety Injection system cross-tie valves, the North Safety Injection Pump was not available for service and the South Safety Injection Pump was only capable of delivering flow to two, rather than four injection points. This condition lasted for approximately 19 hours while the valve was being repaired.

The alignment described above could have resulted in decreased flow to the core had a Reactor Coolant System break occurred. However, flow from the remaining Emergency Core Cooling System components (two charging pumps, four accumulators, and two Residual Heat Removal pumps) would have provided an adequate amount of cooling water to the core.

An evaluation of the event has shown that for a small break loss of coolant accident with one SI pump injecting to two Reactor Coolant System loops, the peak clad temperature will be maintained well below the 10 CFR 50.46 limit of 2200°F.

For the large break loss of coolant accident, Safety Injection flow is a very small fraction of the total Emergency Core Cooling System flow and it is judged that the reduced flow would have had negligible effect.

Based on this analysis, it is concluded that this event did not pose a threat to the health and safety of the public.

The evaluation for the past surveillance practices discussed in the Desciption of Event section, is given in revision one to LER 315-86-21.

Corrective Action

The following corrective actions have been taken:

1) Administrative controls in the form of caution tags were placed on the safety injection and residual heat removal systems in both units to prevent isolation of injection points.

2) An Operations Department memo, dated October 3, 1986, was issued to identify the affected valves.

3) Guidance on entry into T/S 3.0.3 was issued to the plant by the Nuclear Safety and Licensing (NS&L) section of the American Electric Power Service Corporation (AEPSC) in a letter dated September 26, 1986. The letter stated that it was considered permissible by the NRC (both region III and NRR) to voluntarily enter T/S 3.0.3 to perform planned activities. Train-related valve-stroking procedures were revised by December 24, 1986, to preclude isolation of injection points within the 72-hour action statement, but they do allow entry into T/S 3.0.3 for up to one hour.

4) We are presently in the process of submitting a series of safety evaluations which reanalyze the small and large-break LOCA scenarios for Units 1 and 2. The new small-break LOCA analyses assume that one SI pump injects water into only two RCS loops, one of which is the break leg. The new large-break LOCA analyses assume that one RHR pump injects water into only two RCS loops (one of which spills), while one SI pump injects into all four RCS loops. If these analyses are accepted and the requested interpretation of T/Ss 3.5.2 and 3.5.3 is allowed, we will be able to isolate two injection points on the RHR or SI flow paths (not both simultaneously). The requested interpretation would allow for the needed flexibility to perform maintenance and surveillance procedures while in Modes 1-4. The initial letter (AEP:NRC:1024) to request this interpretation was submitted on March 23, 1987.

Failed Components Identification

None

Previous Similar Events

None

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April 7, 1987

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Operating License No. DPR-74
Docket No. 50-316

Document Control Manager:

In accordance with the criteria established by 10CFR50.73 entitled Licensee Event Reporting System, the following report is being submitted:

86-026-01

Sincerely,

A. Alan Blind

W. G. Smith, Jr.
Plant Manager

/afh

Attachment

cc: John E. Dolan
A. B. Davis, Region III
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IE22
1/1