NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

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 4

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 0	1 0	1 0	8 6		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(c)	[ ] 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)	[X] 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	
K. R. Baker, 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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

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

EXPECTED SUBMISSION DATE (15): MONTH 0 1, DAY 0 1, YEAR 8 7

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

On September 12, 1986, during the review of Unit Two intermediate head Safety Injection equipment outage activities a tentative determination was made that the valve alignment utilized 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.

Lack of specificity regarding intermediate head Safety Injection cross-tie capability in the Technical Specification requirements was the cause of this event.

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

The evaluation for this event has been completed; however, due to the afore-mentioned past surveillance practices an additional evaluation is required. A supplemental report containing this analysis will be submitted by January 1, 1987.

~~8610200393~~

NRC Form 366
(9-83)

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

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 0	0 2 OF 0 4

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

Conditions Prior to Occurrence

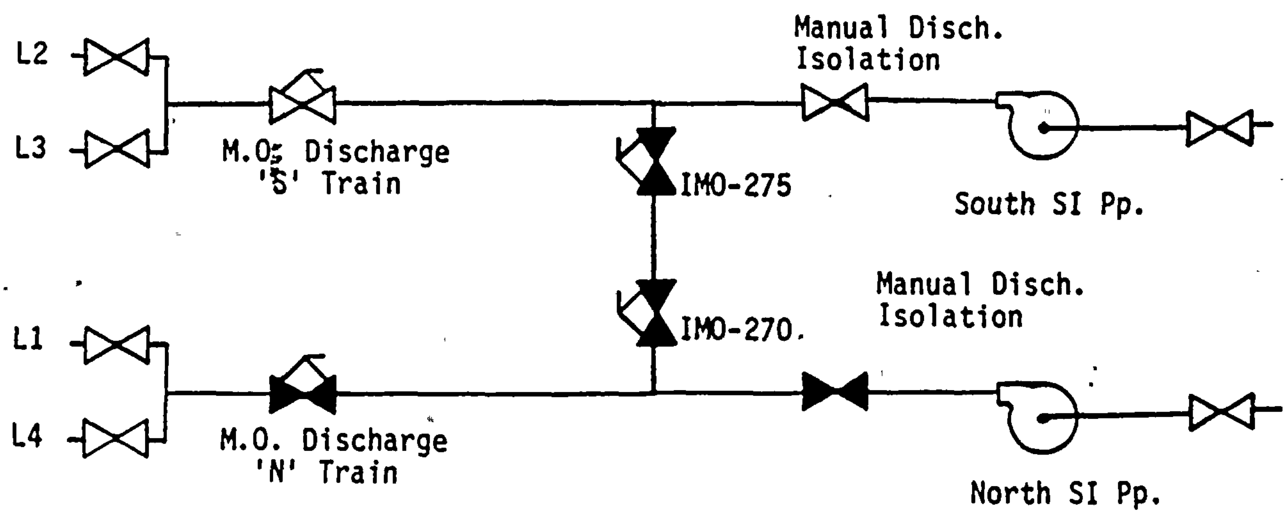
Unit 2 - Mode 1 (power operation) - 80 percent reactor thermal power.

Description of Event

On September 12, 1986, during the review of Unit Two intermediate head 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 intermediate head Safety Injection system outage began on September 4, 1986, at 0645 hours and lasted for a period of 18 hours and 53 minutes. At that 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 were 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.



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. Evaluation of these past ECCS configurations is currently underway.

An information call concerning this additional finding was made to the NRC via ENS at 1114 hours on October 2, 1986.

Cause of the Event

The cause of this event is the result of a lack of specificity in the Technical Specification requirements regarding an operable intermediate head Safety Injection flow path in that the Limiting Condition for Operation does not recognize cross-tie capability. Therefore, the interpretation in the past was that an operable intermediate head Safety Injection train did not require that the cross-tie valves be open. As a result of this silence in the Technical Specifications, closing the cross-tie valves for either preventive maintenance or surveillance testing was not perceived to be a problem.

Past surveillance testing configurations again point to the Technical Specifications lack of specificity. Testing in these configurations was done as allowed by Technical Specifications and not perceived to be contrary to the Final Safety Analysis Report.

Analysis of Event

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

As a result of the isolation boundary established for required maintenance being performed on one of the intermediate head 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 the small break loss of coolant accident there would have been an average flow reduction of 5.84 percent for the limiting case, a four-inch break. This would have resulted in a peak clad temperature of 1791°F, a value less than the 2200°F limit.

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 analysis for past surveillance practices is not complete, but will be submitted in the supplemental report.

Corrective Action

Administrative controls have been placed on the cross-tie valves and other identified Emergency Core Cooling System valves on both Units to prevent isolation. In addition, operational and Technical Specification direction has been sought to ensure compliance with the surveillance requirements for this system.

Failed Component Identification

None

Previous Similar Events

None

Attachment 4 to AEP:NRC:1024

One-Time Exemption for Testing Valves