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)	DOCKET NUMBER (2)	PAGE (3)
D. C. Cook Nuclear Plant, Unit 1	0 5 0 0 0 3 1 5	1 OF 0 4

TITLE (4) Failure to Verify Time Response of Turbine Trip - Reactor Trip Function from Steam Generator High-High Due to Procedural Deficiency

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 3	2 6	8 6	8 6	0 0 7	0 1	0 8	0 9	8 6		0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 0

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)	X 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
T. K. Postlewait Technical Engineering Superintendent	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)

During a review to ensure procedural compliance with Technical Specifications, a discrepancy was noted between the scope of the Reactor Protection and Engineered Safeguards System Time Response test procedure and the requirements of Technical Specifications.

The discrepancy involved time response testing of turbine trip - reactor trip from Steam Generator level high-high. The surveillance procedure included time response through only part of the circuitry involved. The apparent cause is indeterminate. The surveillance procedure has been in effect since time response testing was implemented. It went undetected until the line-by-line Technical Specification review which was conducted to find discrepancies of this type. A decision was made on July 10, 1986 to perform the time response test for the Turbine Trip - Reactor Trip signal from Steam Generator water level high-high. The test included the portion of the circuit previously not tested. The circuit functioned as designed and below the Technical Specification limit of 2.5 seconds.

To prevent recurrence the Engineered Safeguards System time response test procedure will be revised to include the circuitry from the signal origin to the steam stop valve position switch prior to the next scheduled surveillance.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Conditions at Time of Event

Unit 1 in Mode 1, Power Operation at 35 Percent Reactor Thermal Power.

Description of Event

During a review to ensure procedural compliance with Technical Specifications, a discrepancy was noted between the scope of the Reactor Protection and Engineered Safeguards System Time Response test procedure and the requirements of Technical Specifications.

The discrepancy involved Technical Specification (T/S) Table 3.3-5, Item 8a which requires the turbine trip - reactor trip response time to be less than or equal to 2.5 seconds after a Steam Generator (S/G) (EIIS/AB) water level high-high signal. The relevant surveillance procedure includes the portion of the circuit from the origin of the signal, through the Solid State Protection System (SSPS)(EIIS/XC), and up to the output of the turbine trip control master relay (EIIS/94). It did not include the remainder of the circuit which is from the output of the turbine trip control master relay through the electro-hydraulic controls to the output of the oil pressure control relay (EIIS/94), or the parallel part of the circuit from the output of the turbine trip control master relay through the steam stop valve circuitry to the output of the steam stop valve position switch (EIIS/33).

On July 10, 1986, during review of the original LER, an NRC inspector questioned the engineering conclusions that the accident analysis did not include a S/G water level high-high trip. The initial engineering review had been in error in that the S/G water level high-high trip was in fact included in the accident analysis (it was previously concluded that no credit was taken for the high-high level trip function due to an error on this topic in the T/S bases). Westinghouse Corporation was contacted and verified that the S/G level high-high trip function was included in the accident analysis of the Feedwater System malfunction analysis. At 1110 hours it was decided that the plant was in non-compliance with Technical Specifications. A unit shutdown commenced at 1120 hours . A time response test was conducted at 1613 hours. This test included the circuit from the output of the Turbine Trip control master relay through the steam stop valve circuitry to the output of the steam stop valve position switch. The time response from S/G water level high-high to steam stop valve closure was 0.768 seconds for Train A and 1.134 seconds for Train B. These times are well below the T/S limit of less than or equal to 2.5 seconds. The S/G water level high-high trip was declared operable at 1808 hours on July 10, 1986. The time response from steam stop valve closure to reactor trip was determined from an actual trip to be .453 seconds. Thus the total time from steam generator high-high level to reactor trip of 1.587 seconds is well below T/S required less than or equal to 2.5 seconds.

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Description of Event (cont'd)

No other structures, components or systems related to this event were inoperable at this time.

Cause of Event

The apparent cause of this event is indeterminate. The surveillance procedure has been in effect since time response testing was implemented. The discrepancy between the procedure and T/S went undetected until the line-by-line Technical Specification review which was conducted to find discrepancies of this type.

Analysis of Event

This revision to LER 315-86-00, dated April 24, 1986, is being submitted to report an error in the initial engineering review and results of subsequent testing.

This LER is being submitted per the requirement of 10 CFR paragraph 50.73 (a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications.

The S/G water level hi-hi trip function is used to protect against a feedwater system (EIIS/SJ) malfunction. Although the complete circuit was not previously tested for time response, the subsequent testing proved the components to function properly and within the T/S limits. Additionally, the overpower-overtemperature protection features (high neutron flux, overpower delta T, and overtemperature delta T trips) are available for protection against this accident. Based on this, it is concluded that this event did not constitute an unreviewed safety question as defined in 10 CFR 50.59 nor did it pose a threat to the health and safety of the public.

Corrective Actions

The immediate corrective action taken was to commence a reactor shutdown and perform the correct time response test. To prevent recurrence, the Engineered Safety Feature time response test procedure will be revised to include the circuitry from the signal origin to the steam stop valve position switch prior to the next required surveillance. This condition was discovered during a continuing review of T/S surveillance requirements. This review should assure full compliance with Technical Specifications.

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Failed Component Identification

There were no failed components as a result of this event.

Previous Similar Events

Similar events of T/S and surveillance procedures discrepancies were reported in the following LERs:

315-86-014	315-85-049
315-86-013	315-85-047
315-86-008	315-85-043
315-86-007	315-85-032
315-86-006	316-85-025
315-86-002	316-85-023
315-85-064	316-85-021
315-85-050	

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