

CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 9705080285 DOC. DATE: 97/05/05 NOTARIZED: NO DOCKET #
 FACIL: 50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
 AUTH. NAME AUTHOR AFFILIATION
 GILLESPIE, R. Indiana Michigan Power Co.
 BLIND, A.A. Indiana Michigan Power Co.
 RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 97-007-00: on 970405, TS surveillance requirements not met. Caused by personnel error. Instrument isolation valves were opened on 970405.W/970505 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 4
 TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

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FULL TEXT CONVERSION REQUIRED

TOTAL NUMBER OF COPIES REQUIRED: LTTR 24 ENCL 24

Indiana Michigan
Power Company
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106



May 5, 1997

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Operating Licenses DPR-58
Docket No. 50-315

Document Control Manager:

In accordance with the criteria established by 10 CFR 50.73 entitled Licensee Event Report System, the following report is being submitted:

97-007-00

Sincerely,

A handwritten signature in cursive script that reads 'A. A. Blind'.

A. A. Blind
Site Vice President

/mbd

Attachment

c: A. B. Beach, Region III
E. E. Fitzpatrick
P. A. Barrett
S. J. Brewer
J. R. Padgett
D. Hahn
Records Center, INPO
NRC Resident Inspector

IE22.11

9705080285 970505
PDR ADDCK 05000315
S PDR



LICENSEE EVENT REPORT (LER)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)
Donald C. Cook Nuclear Plant - Unit 1DOCKET NUMBER (2)
50-315

Page 1 of 3

TITLE (4)

Technical Specification Surveillance Requirement 4.7.2.1 Not Met Due to Personnel Error

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 NAME	DOCKET NUMBER
04	05	97	97	007	00	05	05	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	6	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10) 00		20.2201(b)	20.2203(a)(3)(i)	50.73(a)(2)(iii)	73.71(b)
		20.2203(a)(1)	20.2203(a)(3)(ii)	50.73(a)(2)(iv)	73.71o
		20.2203(a)(2)(i)	20.2203(a)(4)	50.73(a)(2)(v)	OTHER
		20.2203(a)(2)(ii)	50.36(c)(1)	50.73(a)(2)(vii)	(Specify in Abstract below and in Text, NRC Form 366A)
		20.2203(a)(2)(iii)	50.36(c)(2)	50.73(a)(2)(viii)(A)	
		20.2203(a)(2)(iv)	X 50.73(a)(2)(I)	50.73(a)(2)(viii)(B)	
		20.2203(a)(2)(v)	50.73(a)(2)(ii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME

Mr. Robert Gillespie, Operations Superintendent

TELEPHONE NUMBER (Include Area Code)

616/465-5901, x2535

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)

YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On April 5, 1997, it was determined that the instruments for the Unit 1 Steam Generator (SG) pressure indications in the Control Room had been isolated since March 29, 1997. Personnel had been using these indicators during this time period to verify compliance with Technical Specification 4.7.2.1. This Technical Specification requires that the pressure in both the primary and secondary side of the SGs be verified once per hour as less than 200 psig when either the primary or secondary coolant in the SG is less than 70 °F. As the surveillance requirements were not met, this event is reportable under 10CFR50.73(a)(2)(i)(B), as operation prohibited by Technical Specifications.

The root cause of this event is personnel error, compounded by a lack of guidance in the procedures which support the evolutions. To prevent recurrence of this type of personnel error, procedure changes have been initiated to provide assurance that the SG secondary side pressure instruments are unisolated and available prior to use in the surveillance.

This event was evaluated and it was determined that it had no safety significance, as at no time was the health or safety of the public in jeopardy.

LICENSEE EVENT CONTINUATION

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
Cook Nuclear Plant - Unit 1	50-315	YEAR	SEQUENTIAL	REVISION	2 OF 3
		97	- 007 -	00	

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

Condition Prior to Event

Unit 1 was in Mode 6, Refueling, and the core offloaded.

Description of Event

On March 29, 1997, personnel were in the process of establishing containment integrity in preparation for Unit 1 core reload. The procedure that controls this process contains two methods that can be used to assure closure of the penetrations associated with the Steam Generators (SG), an "inside containment" method and an "outside containment" method. Work was in progress in the containment on the SGs at the time, and the decision was made to use the "outside" containment isolation method. This involved closing the isolation valves associated with the Main Steam (MS) pressure instruments. The valves were closed at 0609 hours on March 29, 1997.

In support of ongoing SG evolutions, it was necessary to add water to the SGs. The water came from the Unit 1 Condensate Storage Tank, with a temperature of less than 70 °F. At 0521 hours on March 29, 1997, personnel began filling SG 13. Due to the addition of the cooler water the SG 13 temperature dropped below 70 °F at 0818 hrs on March 29, 1997. Technical Specification 4.7.2.1 requires a once per hour verification that the pressure in each side of the SGs remained below 200 psig when either the primary or secondary coolant in the SG is less than 70 °F. At this time actions were initiated to verify, via the Control Room indication, that the SG and Reactor Coolant System (RCS) pressures were below 200 psig. As filling of the other SGs proceeded, their temperatures dropped below 70 °F, and were monitored as well.

Subsequently the SGs were warmed in preparation for pressure testing by the addition of warm water from the Unit 2 blowdown system. On April 5, 1997, SG 14, the first SG to be tested, was pressurized to 400 psig. The evolution was stopped when it was noted that the pressure indicators varied widely in their readings.

Personnel investigated and found the pressure instrumentation for SG 14 to be isolated. The instrumentation for the other SGs was checked, and found to be isolated as well. The valves were opened and the instruments returned to service at 1815 hours on April 5, 1997.

Cause of Event

The root cause of this event was personnel error. Personnel did not recognize that the SG pressure instruments had been isolated when initiating the surveillance for the SG pressures. Directly contributing to this event was the lack of guidance in the containment integrity procedure, and lack of direction to verify the transmitters are available before using them in the surveillance procedure. These factors, in combination, led Operations personnel to make the assumption that the instruments were available when initiating the surveillance for verification of SG secondary side pressure.

The variation in pressure readings on SG 14 instrumentation, which first alerted personnel to the problem, was attributed to valve leaky.

LICENSEE EVENT CONTINUATION

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		YEAR	SEQUENTIAL	REVISION	
Cook Nuclear Plant - Unit 1	50-315	97	- 007 -	00	3 OF 3

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

Analysis of Event

This event is being reported in accordance with 10CFR50.73(a)(2)(i)(B), as operation prohibited by Technical Specifications, in that the required surveillances were not performed for Technical Specification 4.7.2.1.

The limitation on SG pressure and temperature ensures that the pressure induced stresses in the SGs do not exceed the maximum allowable fracture toughness stress limits. The limitation of 70 °F and 200 psig are based on average SG impact values taken at +10 °F and are sufficient to prevent brittle fracture.

A review of work performed on the SGs during the time period when the instrumentation was isolated revealed that SG sparging was the only evolution which took place which had the potential to increase secondary side pressure. Per procedure, the safety valve on the nitrogen bank used for sparging was set at 150 psig, and a second regulator, set at 75 psig was also used to maintain the secondary side pressure well below the 200 psig limit.

During core reload, the SG nozzle dams were installed on all SG hot and cold legs to facilitate SG repair work, thereby isolating the secondary from the potential heat source of the core. In addition, the RCS temperature is maintained between 68 °F and 110 °F, which would minimize any heat transfer as well.

It can therefore be concluded that the SG were not inadvertently pressurized to a condition of greater than 200 psig while less than 70 °F during the period when the surveillance was not correctly performed. The activities in progress on the SG were controlled by procedure and precluded pressurization of the SG. It was concluded that at no time was the health of safety of the public in jeopardy.

Corrective Actions

The instrument isolation valves were opened on April 5, 1997.

To prevent this type of personnel error in the future, the following procedure changes have been undertaken:

- OHP 4030.STP.030, "Daily and Shift Surveillance Checks", was revised to include a CAUTION prior to initiating the surveillance. This CAUTION states that all pressure transmitters must be verified to be valved in and in working condition prior to implementing the surveillance. The revision has been completed on the procedures for both units.
- OHP 4030 STP.041, "Refueling Integrity", for both units, is the process of being revised. The revision, which will delete the requirement to isolate the SG pressure instrumentation and instead verify that they are installed and intact, will be implemented prior to its next use. The expected next use is for fall 1997 refueling outage on Unit 2. The revision will be implemented on the procedures for both units.

Failed Component Identification

None

Previous Similar Events

None