CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9802270048 DOC.DATE: 98/02/13 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 97-007-01:on 970405,determined that SG pressure indications in CR had been isolated since 970329.Caused by procedure inadequacy.Revised SG isolation methodology contained in refueling integrity surveillance.W/980213 ltr.

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

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February 13, 1998

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:

97-007-01

Sincerely,

J. R. Sampson

J. R. Sampson
Site Vice President

/mbd

Attachment

c: A. B. Beach, Region III
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IE22 1/1

NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

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 (MNBB 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.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	
Donald C. Cook Nuclear Plant - Unit 1	50-315	Page 1 of 3

TITLE (4)
Technical Specification Surveillance Requirement 4.7.2.1 Not Met Due to Procedural Inadequacy

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 –	01	02	13	98	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	6
POWER LEVEL (10)	00

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

	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)		73.71(c)
	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	TELEPHONE NUMBER (Include Area Code)
Mr. Robert Gillespie, Operations Superintendent	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)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES	X	NO				

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

On April 5, 1997, it was determined that the Unit 1 Steam Generator (SG) pressure indications in the control room had been isolated since March 29, 1997. Personnel had been using these indications to verify compliance with Technical Specification Surveillance 4.7.2.1. This Technical Specification requires that the pressure in both the primary and secondary side of the SGs be verified as less than 200 psig once per hour when either the primary or secondary coolant in the SG is less than 70 degrees Fahrenheit (°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 was deficient procedures used in the processes of isolating the SGs to establish refueling containment integrity and initiate the Technical Specification surveillance. To prevent recurrence, the SG isolation methodology contained in the refueling integrity surveillance has been revised, as has the daily surveillance for verification of SG pressures. The daily surveillance now verifies that the SG secondary side pressure instruments are unisolated and available prior to utilization in the surveillance.

During the time period the instrument was isolated, it has been determined that at no time was the SG pressure greater than 200 psig concurrent with either the primary or secondary temperature being less than 70 °F; therefore, this event had no safety significance.

LICENSEE EVENT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL	REVISION	
Cook Nuclear Plant - Unit 1	50-315	97	– 007 –	01	2 OF 3

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 off loaded.

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 degrees Fahrenheit (°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 deficient procedures used to isolate the SGs to establish refueling containment integrity and initiate the TS 4.7.2.1 surveillance. The pressure instruments were non-functional due to isolation valves that were closed earlier and not reopened. The procedure to establish containment integrity did not allow acceptable alternatives for the outside SG penetration isolation, such as verifying the instrumentation was installed. The procedure associated with the pressure surveillance (TS 4.7.2.1) did not have provisions to verify that the instruments were functional prior to use. The procedures were deficient in that the combination of conditions, changes in the schedule, and the procedures did not assure the process was completed correctly.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

LICENSEE EVENT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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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.

During the time period the instrument was isolated, it has been determined that at no time was the SG pressure greater than 200 psig concurrent with either the primary or secondary temperature being less than 70 °F; therefore, this event had no safety significance.

Corrective Actions

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

To prevent this error from recurring, 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", was revised to delete the requirement to isolate the SG pressure instrumentation and now only simply requires verification that they are installed and intact. This revision has been completed on the procedures for both units.

Failed Component Identification

None

Previous Similar Events

None