CATEGORY 1

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ACCESSION NBR:9705080291 DOC.DATE: 97/05/05 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 97-008-00:on 970404,cycle 15 operation of SG outside TS tube degradation acceptance criteria.Caused by inadequate analysis of eddy current data.1997 SG eddy current tube insps conducted.W/970505 ltr.

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Bridgman, MI 49106

INDIANA
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POWER

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:

97-008-00

Sincerely,

A. Alan Blind

A. A. Blind
Site Vice President

/mbd

Attachment

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NRC FORM 366 (5-92) NUCLEAR REGULATORY COMMISSION

OVED 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)	
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TITLE (4)
Cycle 15 Operation of Steam Generators Outside Technical Specification Tube Degradation Acceptance Criteria Due to Inadequate Analysis of Eddy Current Data

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	04	97	97	– 008 –	00	05	05	97	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©
	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. David Petro, Engineering Projects and Programs Supervisor	616/697-5107

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 (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On April 4, 1997, with Unit 1 in Mode 6, it was determined that a number of the hot leg tubesheet region Steam Generator (SG) tubes with indications had been placed back into service at the end of the 1995 Cycle 15 refueling outage after performing SG tube inspections and tube repairs. During the 1997 inspection a large number of tubes in the hot leg tubesheet region of each SG were identified as degraded. Re-analysis of the Cecco-5 probe data from the previous SG inspection of the largest tubesheet crevice indications determined that these indications were also present in 1995, implying that tubes with defects had been placed back in service, and not repaired in accordance with Technical Specification 4.4.5.4.a.6 and 4.4.5.4.b. In accordance with 10CFR50.73 (a)(2)(i)(B), this event is reportable as operation prohibited by the plant's Technical Specifications.

The cause of this event is attributed to inadequate analysis of the Cecco-5 eddy current data in 1995. For the 1997 outage, EPRI qualified bobbin coil and rotating coil inspection techniques and an EPRI qualified data analyst were used. For future inspections, use of a qualified process with better data analysis capability, use of highly trained and qualified analysts, and independent data review is expected to significantly minimize or preclude occurrence of a similar event.

Eddy current tube inspections and a bounding tube pressure test program at the end of Cycle 15 determined that the tubes with the largest defects met the draft RegGuide 1.121 for leakage at Main Seam Line break conditions and for burst at 3 times the normal operating pressure differential. Therefore, there was no hazard to the health and safety of the public during Cycle 15.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

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LICENSEE EVENT CONTINUATION

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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)

Condition Prior to Event
Unit 1 was in Mode 6, Refueling

Description of Event
During eddy current tube inspection and tube repairs, conducted March 15 through April 9,1997, a large number of tubes in the hot leg tubesheet region of each Steam Generator (SG) were identified as being degraded. This region of the SG was eddy current inspected the full length of the tubesheet with a rotating coil probe. The degradation consisted of axial crack indications and was identified as either Outside Diameter Stress Corrosion Cracking (ODSCC) at or near the top of the tubesheet, or Primary Water Stress Corrosion Cracking (PWSCC) at the partial depth tube roll transition. Due to the large numbers of defects found in the tubes, analysis of the Cecco-5 probe data from the previous refueling outage for 28 of the largest ODSCC indications determined that the indications were present in 1995. Some were initially identified in the bobbin coil data from 1995, but had been dispositioned as acceptable based on the Cecco-5 data. When re-evaluated with the aid of the 1997 data, a number of the indications could be seen on rotating coil data from 1994. Although present then, they were smaller than observed in 1997. After completion of the re-analysis, it was concluded that tubes with defects had been placed into service for fuel Cycle 15.

Cause of Event
The cause of this event is attributed to inadequate analysis of the Cecco-5 eddy current data back in 1995. Though the technique was qualified for the inspection of tubing in this region of the SGs, data analysis techniques used at the time were inadequate. Analysis of Cecco-5 data is complicated by the many channels of data and its presentation format to the analyst. Improvements have been made in analyzing the Cecco-5 data through the use of improved guidelines, analysis software and also, analyst training and qualification and industry experience. These improvements enabled the re-analysis process to identify the tube degradation in 1997. A review of SG secondary side chemistry and Cecco-5 probe qualification requirements did not reveal evidence of a different cause.

Analysis of Event
This event is reportable under the provisions of 10CFR50.73(a)(2)(i)(B) as an operation prohibited by plant Technical Specification 4.4.5.4 a.6 and 4.4.5.4 b. Technical Specification 4.4.5.4a.6 and 4.4.5.4b require that all tubes exceeding the repair limit and all tubes containing through-wall cracks be plugged or repaired. Based on the re-analysis of the 1995 Cecco-5 data it was evident that the data analysis conducted in 1995 was discrepant in that tubes with defects were returned to service.

Evaluation of the 1997 rotating coil eddy current data and a bounding tube in-situ pressure test program confirmed that the tube structural integrity met draft RegGuide 1.121 requirements. Based on the insitu pressure testing results for all degradation mechanisms, including tube sheet region ODSCC and PWSCC, main steam line break leakage was determined to be 0.64 gpm for the most limiting SG loop, which is well below the Technical Specification allowable of 8.4 gpm. Maintenance of leakage below this limit in the event of a Main Steam Line (MSL) break ensures that the calculated off-site doses would be within 10 percent of the 10 CFR 100 limits. Also, for those tested at three times normal operating pressure differential as part of the program, no tube burst occurred in any of the tubes.

It is therefore concluded that this event did not represent a hazard to the health or safety of the public.

NRC FORM 366A — U.S. NUCLEAR REGULATORY COMMISSION

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LICENSEE EVENT CONTINUATION

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

<u>Corrective Action</u>

The 1997 SG eddy current tube inspections were conducted using EPRI qualified bobbin coil and rotating coil inspection techniques and an EPRI qualified data analyst. Data analysis software used for the rotating coil inspections is much easier to analyze than that used for Cecco-5. It has fewer channels to review and uses a terrain map format which provides a better pictorial representation of the data for recognizing crack like indications. In addition, analysts were required to pass a site specific examination. Data analysis was also checked during the 1997 inspections by employing an independent data analysis process.

Use of the qualified process with better data analysis capability, highly trained and qualified analysts, and independent data review is expected to significantly minimize or preclude occurrence of a similar event in the future. These requirements have been incorporated into the technical scope document which governs this work.

In addition, use of Cecco-5 probes for future tube inspections has been discontinued until the data analysis techniques have been improved, qualified, and gained wide spread industry acceptance.

<u>Failed Component Identification</u>

N/A

<u>Previous Similar Events</u>

None