CATEGORY 1

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ACCESSION NBR:9707090394 DOC.DATE: 97/07/01 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 97-002-01:on 970128.stresses for piping was found to exceed allowable valves during postulated DBA.Caused by inadequate analysis during original design.Review completed as part of 120 day response to GL 96-06.W/970701 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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INDIANA
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POWER

July 1, 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:

97-002-01

Sincerely,

A. Alan 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

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 2

TITLE (4)
LER Retraction - Stresses for Piping Found to Exceed Allowable Values During Postulated Design Basis Accident Due to Inadequate Analysis During Original Design

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: Cook Unit 2	DOCKET NUMBER: 50-316
01	28	97	97	-- 002 --	01	07	01	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
POWER LEVEL (10)	96.6		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)		50.73(a)(2)(i)		50.73(a)(2)(viii)(B)		
			20.2203(a)(2)(v)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Paul Schoepf, Engineering Manager for Safety Related Mechanical Systems	616/465-5901, x2408

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 January 28, 1997, with Unit 1 at 96.6 percent Rated Thermal Power and Unit 2 at 100 percent Rated Thermal Power, initial analyses performed in response to Generic Letter (GL) 96-06 on thermal over pressurization were completed. At that time, it was determined that the reactor coolant pump seal water return line, the reactor coolant sample lines, and the accumulator sample line could potentially over pressurize during a design basis accident; thus, in accordance with 10 CFR 50.72(b)(1)II)(B), this condition was reported.

As outlined in our 120 day response to GL 96-06, supplemental analyses were completed to determine the internal pressure and the resultant wall stresses in each of the aforementioned lines following a design basis accident. Those analyses concluded that the fluid lines in question would not over pressurize and that the pipe wall stresses would remain within the design limits under all conditions. For these reasons, the Licensee Event Report (LER) previously submitted on February 26, 1997, as LER 315/97-002-00 is being retracted.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL	REVISION	
Cook Nuclear Plant - Unit 1	50-315	97	-- 002 --	01	2 OF 2

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

Conditions Prior to Occurrence

Unit 1 was in Mode 1 at 96.6 percent Rated Thermal Power.
Unit 2 was in Mode 1 at 100 percent Rated Thermal Power.

Description of Event

In response to GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions", evaluations and analyses were performed to determine which fluid lines penetrating containment, if any, would be susceptible to thermal over pressurization during a design basis event.

On January 28, 1997, it was determined that the reactor coolant pump seal water return line, the accumulator sample line, and the reactor coolant samples lines, which are isolated during an accident, had the potential of becoming pressurized with the resulting stresses exceeding the allowable stress limits given in the FSAR. Subsequent analyses have been performed, the results of which are summarized in the Analyses section of this document, and in AEP:NRC:1256B, the supplemental response submitted on May 20, 1997.

Analysis of the Event

As part of the 120 day response to Generic Letter 96-06, a review was completed to determine if the fluid lines that penetrate containment are susceptible to thermal over pressurization following a postulated accident. The review identified five lines in each unit where pressure relief could not be predicted to prevent the stresses in the pipe wall from exceeding FSAR allowable stresses for emergency conditions. Those lines are the reactor coolant pump seal water return line, the accumulator sample line, and the three reactor coolant sample lines. Additional analyses were completed to determine if pressure relief is necessary for those five lines.

For the reactor coolant pump seal water return line and the accumulator sample line, a thermodynamic enthalpy balance was completed to determine the maximum internal pressure that would develop in the fluid lines following a postulated accident. That calculated pressure was then compared to the pressure required to induce wall stresses in excess of FSAR allowable stresses for emergency conditions. The analyses concluded that the internal pressure that would develop in the lines following a design basis accident would not be great enough to induce wall stress in excess of the FSAR allowable stresses.

For the reactor coolant sample lines, the fluid temperature during normal plant operation is greater than the post-accident containment temperature. Consequently, following a postulated accident, the samples lines will actually cool down instead of heat up. As such, the reactor coolant system sample lines are not susceptible to thermal over pressurization. As it has been shown that these lines will NOT over pressurize during a design basis accident; therefore, LER 315/97-002-00, submitted on February 26, 1997, is being retracted.