CATEGORY 1

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ACCESSION NBR:9703040253 DOC.DATE: 97/02/26 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 97-002-00:on 970128,stresses for piping found to exceed allowable values during postulated design basis accident due to inadequate analysis during original design.Alternat stress analysis was performed.W/970226 ltr.

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AEP
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February 26, 1997

United States Nuclear Regulatory Commission
Document Control Desk
Rockville, Maryland 20852

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:

97-002-00

Sincerely,

A. Alan Blind

A. A. Blind
Site Vice President

/mbd

Attachment

c: A. B. Beach, Region III
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NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

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

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)
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 –	00	02	26	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	96.6

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[illegible]
	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, analyses performed in response to Generic Letter 96-06 on thermal overpressurization were completed. As a result of these analyses, it was determined that during a design basis accident portions of the reactor coolant pump seal water return line, the accumulator sample lines and the reactor coolant sample lines could experience stresses which exceed the design basis allowables stated in the FSAR. This event was determined to be reportable under the provisions of 10 CFR 50.72(b)(1)(ii)(B), as a condition outside the design basis. A one hour notification was made at 1523 hours on January 28, 1997.

The root cause of the event is inadequate analysis for thermal overpressurization during original design of the plant. Supplemental analyses are being performed to review assumptions, service conditions and methodologies used to calculate internal system pressures and related pipe stresses. These analyses will be completed by July 31, 1997.

An alternate stress analysis was performed using ASME Section III, Appendix F, "Level D Service Limits", and the results of these analyses provide assurance that the piping is capable of performing its containment isolation function. Based on this analysis, it was determined that this event was of minimal safety consequence, and did not endanger the health or safety of the public.

NRC FORM 366A 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 CONTINUATION

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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 Generic Letter (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 overpressurization during a design basis event.

On January 28, 1997, it was determined that the reactor coolant pump seal water return line, the accumulator sample lines and the reactor coolant sample lines, which are isolated during the accident, had the potential to become pressurized with the resulting stresses exceeding the allowable stress limits given in the FSAR. Following a postulated accident, the containment is filled with steam and the containment temperature increases. As the isolated piping sections inside containment are exposed to its environment, the water inside the piping could be heated, and the expansion of the heated fluid could cause the pressure in the piping to increase. As a result of the pressure increase, the stresses in the piping could exceed the allowable stress limits given in the FSAR.

Cause of Event

The root cause of the event is inadequate analysis for thermal overpressurization of fluid filled piping systems penetrating the containments during original design of the plant.

Analysis of the Event

This event is being reported under 10 CFR 50.73(b)(1)(ii)(B) as a condition outside of the design basis of the plant. It was determined that during a design basis accident portions of the reactor coolant pump seal water return line, the accumulator sample lines and the reactor coolant sample lines could experience stresses which exceed the design basis allowables stated in the FSAR.

Fluid flow through these lines is not required following a design basis accident; thus, the function of the piping and the associated valves is to provide containment isolation. In this regard, the structural integrity of the piping must be maintained. The ability of the affected piping to perform its containment isolation function was evaluated using the criteria of ASME Section III, Appendix F, "Level D Service Limits". These evaluations showed that although the stresses in the piping exceeded the FSAR allowables, the stresses were within the Appendix F limits, and it was concluded that the piping sections are capable of performing their containment isolation functions. These analyses are described in detail in AEP:NRC:1256A, the D.C. Cook response to GL 96-06.

Based on the results of the analyses, it was determined that the event had minimal safety significance, and did not endanger the health or safety of the public.

LICENSEE EVENT CONTINUATION

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

Corrective Action

As required by GL 96-06, a review was conducted of all fluid filled lines penetrating the containments to determine vulnerability to thermal overpressurization. This review identified several systems for which the original design may be inadequate for thermal overpressure protection. Alternate stress analysis was performed using ASME Section III, Appendix F, "Level D Service Limits", and the results of these analyses provide assurance that the piping is capable of performing its containment isolation function.

Analysis of the piping systems is continuing to determine if modifications are necessary to limit stresses to within FSAR allowables. This includes review of assumptions, service conditions, and methodologies used to calculate internal system pressures and related pipe stresses. These supplemental analyses will be completed by July 31, 1997. Plant modifications, if required, will be completed during refueling outages after January, 1998.

Upon conclusion of all analyses and any required plant modifications, design basis documents will be updated to reflect the results of the GL 96-06 reviews as appropriate.

Failed Component Identification

Not Applicable

Previous Similar Events

None