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FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315

AUTH.NAME	AUTHOR AFFILIATION
KINGSEED,J.	Indiana Michigan Power Co.
SAMPSON,J.R.	Indiana Michigan Power Co.
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 98-014-00:on 980310,unanalyzed condition resulted from procedure not being consistent w/analysis of record.Caused by functional restoration procedure FRZ-1.Issued being reported under 10CFR50.73(a)(2)(ii)(A).W/980406 ltr.

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

INDIANA
MICHIGAN
POWER

April 6, 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:

98-014-00

Sincerely,

J. R. Sampson
Site Vice President

/mbd

Attachment

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

1/1

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	

TITLE (4)
Interim LER - Unanalyzed Condition Results from Procedure Not Being Consistent with Analysis of Record

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
03	10	98	98	-- 014 --	00	04	06	98	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	5
POWER LEVEL (10)	0

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)		73.71c
	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)
Mr. Jeb Kingseed, Nuclear Safety and Analysis Manager	616/697-5106

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
X	YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		08	31	98

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

On March 10, 1998, with Units 1 and 2 in Mode 5, it was determined that both units had operated in an unanlyzed condition due to Functional Restoration Procedure FRZ-1, "Response to High-High Containment Pressure", not being consistent with the containment integrity analysis of record. In accordance with 10CFR50.72(b)2)(i), "[a]ny event found while the reactor is shutdown, that, had it been found while the reactor was in operation, would have resulted in the nuclear plant, including it [principle] safety barriers, being in an unanalyzed condition that significantly compromises plant safety", an ENS notification was made.

RHR spray is used to supplement containment spray to control containment pressure. The procedure directs that actuation of RHR spray be manually initiated at a containment pressure of 8 psig. Procedure development assumed that operator implementation of this step could be performed without an appreciable increase in containment pressure. Review of the procedure against design basis accident assumptions indicated that his is not the case. As a result, current procedures may be non-conservative with respect to this analysis. Had the procedure been implemented in it current form, a potential may have existed for the post-LOCA containment pressure to exceed its design basis limit of 12 psig. Therefore, this issue is being reported under 10CFR50.73(a)(2)(ii)(A) as an unanalyzed condition that significantly compromised plant safety.

Both the procedure and its associated analysis are being evaluated to determined the appropriate resolution of this issue. An update to this LER will be provided by August 31, 1998.