CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9812010195 DOC.DATE: 98/11/23 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
AUTH.NAME AUTHOR AFFILIATION
BRASSART,G. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
SAMPSON,J.R. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 98-014-02:on 980310,determined that plant had operated in unanalyzed condition.Caused by inadequate interface with W re assumptions used in safety analysis.Will revise functional restoration procedure FRZ-1.With 981123 ltr.

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

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD3-3 PD	1	1	STANG,J	1	1
INTERNAL:	AEOD/SPD/RAB	2	2	AEOD/SPD/RRAB	1	1
	FILE CENTER	1	1	NRR/DE/ECGB	1	1
	NRR/DE/EELB	1	1	NRR/DE/EMEB	1	1
	NRR/DRCH/HICB	1	1	NRR/DRCH/HOHB	1	1
	NRR/DRCH/HQMB	1	1	NRR/DRPM/PECB	1	1
	NRR/DSSA/SPLB	1	1	RES/DET/EIB	1	1
	RGN3 FILE 01	1	1			
EXTERNAL:	L ST LOBBY WARD	1	1	LITCO BRYCE,J H	1	1
	NOAC POORE,W.	1	1	NOAC QUEENER,DS	1	1
	NRC PDR	1	1	NUDOCS FULL TXT	1	1

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LISTS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTROL DESK (DCD) ON EXTENSION 415-2083

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 22 ENCL 22

AO

Indiana Michigan
Power Company
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
616 465 5901

AEP
INDIANA
MICHIGAN
POWER

November 23, 1998

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Operating License 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:

LER 98-014-02, "'Response to High-High Containment Pressure' Procedure Not Consistent with Analysis of Record"

Sincerely,

John R. Sampson

J. R. Sampson
Site Vice President

/mbd

Attachment

c: J. L. Caldwell (Acting), Region III
R. P. Powers
P. A. Barrett
J. B. Kingseed
R. Whale
D. Hahn
Records Center, INPO
NRC Resident Inspector

9812010195 981123
PDR ADOCK 05000315
S PDR

Ie22

NRC Form 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), 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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Cook Nuclear Plant Unit 1	05000-315	1 of 3

TITLE (4)

"Response to High-High Containment Pressure" Procedure Not 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	DOCKET NUMBER
03	10	98	98	-- 014 --	02	11	23	98	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
		0.2201 (b)	0.2203(a)(2)(v)	0.73(a)(2)(i)	0.73(a)(2)(viii)
POWER LEVEL (10)	00	0.2203(a)(1)	0.2203(a)(3)(i)	X 0.73(a)(2)(ii)	0.73(a)(2)(x)
		0.2203(a)(2)(i)	0.2203(a)(3)(ii)	0.73(a)(2)(iii)	3.71
		0.2203(a)(2)(ii)	0.2203(a)(4)	0.73(a)(2)(iv)	THER
		0.2203(a)(2)(iii)	0.36(c)(1)	0.73(a)(2)(v)	Specify in Abstract below or n NRC Form 366A
		0.2203(a)(2)(iv)	0.36(c)(2)	0.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Mr. Gary Brassart, 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 EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If Yes, complete EXPECTED SUBMISSION DATE).	NO		12	23	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 unanalyzed condition due to Functional Restoration Procedure FRZ-1, "Response to High-High Containment Pressure", not being consistent with the containment integrity analysis of record. Had the procedure been implemented in the current form, the potential existed for post-accident containment pressure to exceed its design basis limit of 12 psig. In accordance with 10CFR50.72(b)(2)(i), any 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 its principle safety barriers, being in an unanalyzed condition that significantly compromises plant safety, an ENS notification was made. This LER is therefore submitted in accordance with 10CFR50.73(a)(2)(ii)(A), for an unanalyzed condition, and 10CFR50.73(a)(2)(ii)(B), for a condition outside the design basis.

The root cause of this condition was inadequate interface with Westinghouse regarding the assumptions used in the safety analysis. The procedure will be revised to direct initiation of RHR spray at the appropriate point in time after the event. A program will be established to identify, document and control key aspects of the EOPs that have an operator interface, and EOP setpoints that should be controlled by Engineering. Additional actions will be taken to strengthen the communications between Operations and vendors performing safety analyses that might impact actions taken by the operators.

The evaluation of the safety significance of this condition is still ongoing. The results of that evaluation should be available for submittal by December 23, 1998.

FACILITY NAME (1)	DOCKET NUMBER(2)	LER NUMBER (6)					PAGE (3)
		YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
Cook Nuclear Plant Unit 1	05000-315	98	--	014	--	02	2 of 3

TEXT *(If more space is required, use additional copies of NRC Form (366A)* (17)

Conditions Prior to Event
Unit 1 was in Mode 5, Cold Shutdown
Unit 2 was in Mode 5, Cold Shutdown

Description of Event
On March 10, 1998, while performing a Containment Spray self assessment, it was determined that the actions directed by Functional Restoration Procedure 1,2-4023.OHP.FRZ-1, "Response to High-High Containment Pressure", were not consistent with the assumptions in the containment integrity analysis of record.

Residual Heat Removal (RHR) spray is designed to supplement the pressure mitigation function of containment spray during either a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). In accordance with containment integrity analysis input assumptions, FRZ-1 directs that RHR spray be manually initiated when containment pressure reaches 8 psig. The procedure and the safety analysis did not make allowance for the time delay between containment pressure reaching 8 psig and the delivery of RHR spray to containment. This time delay results from the summation of the time required for the operator to recognize that containment pressure has reached 8 psig; for the RHR spray valves to open and RHR to Reactor Coolant system isolation valves to close/throttle; and for RHR flow to fill the spray line and spray headers. Had the use of this procedure in its current form been required, conditions after LOCA or MSLB may not have been mitigated sufficiently to maintain containment pressure below the 12 psig design basis.

Cause of Event
This condition was the result of an inadequate interface with Westinghouse regarding the assumptions used in the safety analysis and how they were implemented at the plant. Equipment response times and operator action times were not included by Westinghouse when assumptions regarding RHR spray were incorporated into the analysis.

Analysis of Event
This condition was determined to be reportable in accordance with 10CFR50.73(a)(2)(ii)(A), for an unanalyzed condition that significantly compromises plant safety, and 10CFR50.73(a)(2)(ii)(B) for a condition outside the design basis.

The evaluation of the safety significance of this condition is still ongoing. The results of that evaluation should be available for submittal by December 23, 1998.

Corrective Actions
Functional Restoration Procedure FRZ-1, "Response to High-High Containment Pressure", will be revised to initiate RHR spray at a specific time after the start of an accident.

The containment integrity analysis will be revised to incorporate an assumption regarding the time at which RHR spray flow to the upper containment volume will be established. The analysis will assume that spray is initiated later than the time at which the operators will be required to initiate spray. This will provide additional margin for the time required for the initiation of RHR spray, repositioning of the valves, and filling of RHR spray lines.

To alleviate the interface problem with Westinghouse, a program will be developed and implemented to identify, document and control the key accident analyses assumptions used in the safety analyses that can be impacted by operator action in the EOPs; the key events involving operator action duration that can impact the safety analyses and are part of the EOPs; and the setpoints that should be subject to engineering control that are part of the EOPs.

Vendors who perform safety analyses will be required to identify any impact of the above program items on their safety analyses. Plant procedures will require that any change in operator actions that is driven by the safety analyses will require concurrence from the Operations Superintendent or Plant Manager.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION
(6-1998)

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER(2)	LER NUMBER (6)					PAGE (3)
		YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
Cook Nuclear Plant Unit 1	05000-315	98	--	014	--	02	3 of 3

TEXT *(If more space is required, use additional copies of NRC Form (366A)* **(17)**

Failed Component Identification
Not Applicable

Previous Similar Events
None