CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9711250109 DOC.DATE: 97/11/17 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315

AUTH.NAME	AUTHOR AFFILIATION
GILLESPIE,R.	Indiana Michigan Power Co.
BLIND,A.A.	Indiana Michigan Power Co.
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 97-019-00:on 970911,operation contrary to design bases w/RHR suction valves ACI defeated in modes 4 & 5.Caused by inadequate safety review.Requested TS amend to remove surveillance re ACI on RHR valves.W/971117 ltr.

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

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CATEGORY 1 DOCUMENT

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AEP
INDIANA
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November 17, 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:

97-019-01

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

1/1

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 4

TITLE (4)
Operation Contrary to the Design Bases with Residual Heat Removal Suction Valves Automatic Closure Interlock Defeated In Modes 4 and 5

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
09	11	97	97	-- 019 --	01	11	17	97	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)		73.71(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)
Mr. Robert Gillespie, Operations Superintendent	616/465-5901, x2535

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)

In September 1997, during the NRC Architect and Engineering Design Inspection, a discrepancy was identified between the Updated Final Safety Analysis Report (UFSAR) and an Operations department procedure. The procedure also conflicted with Technical Specification (T/S) requirements. This condition was reported as a notification under 10 CFR 50.72(b)(2)(i), for an unanalyzed condition on September 11, 1997, and as a condition outside the design basis of the plant. This update is submitted to provide the cause and safety significance of the event.

The event has been attributed to an inadequate safety review performed at the time the procedure change was made in 1980. Additionally, a procedure change was used to effectively accomplish a design change to the plant, further reducing the chances, under the processes in existence at that time, of identifying the impact on the design basis. A T/S amendment has been formally requested to remove the surveillance related to automatic valve closure. The change was requested on September 19, 1997, via our letter designated AEP:NRC:1278. A design change will also be processed to document the change appropriately under our design change process.

The procedure steps involved were intended to prevent inadvertent auto-closure of the valves which would result in a loss of Residual Heat Removal (RHR) suction during shutdown cooling operation. The procedure accomplishes this by removing power from the valves when the valves are opened to the Reactor Coolant System (RCS). The RCS and the RHR systems were protected from over-pressurization by the Low Temperature Over-Pressure Protection (LTOP) equipment, at all times that the original RHR suction valve interlock was designed to protect the RHR system. Therefore, this condition did not significantly impact the health or safety of the public or pose a threat to plant safety.

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

Condition Prior to Event
Unit 1 Mode 5, Cold Shutdown
Unit 2 Mode 5, Cold Shutdown

Description of Event
In September 1997, personnel were reviewing information being provided to the NRC Architect Engineer Design Inspection team. During this review, a discrepancy was identified between the Updated Final Safety Analysis Report (UFSAR) and an Operations department procedure. Further evaluation identified that the procedure also conflicts with Technical Specification (T/S) requirements.

UFSAR Chapter 9 describes the interlocks associated with the Residual Heat Removal (RHR) suction valves IMO-128 and ICM-129. The valves are located in series on the single RHR cooldown suction line. They are interlocked through separate channels of the RCS pressure instrumentation to provide automatic closure of both valves whenever RCS pressure exceeds the RHR design pressure. T/S 3.5.2 and 3.5.3 require the interlock to be operable in Modes 1 through 4.

The requirement for this auto-closure capability dates back to our original T/S and FSAR documents. The over-pressure protection was designed to prevent an intersystem loss of coolant accident, caused by an over-pressure condition in the RCS, which could result in a break in the RHR system.

However, since June 1980, this interlock has been defeated on both units whenever the RHR system is operating in the normal cooling configuration. This practice began in order to prevent inadvertent auto-closure of the valves which would result in loss of RHR suction during shutdown cooling operation. The interlock is defeated by removing power from the valves. This action is taken as soon as the valves are opened to place RHR in service for shutdown cooling, in Mode 4.

In May of 1980, IE Information Notice 80-20, "Loss of Decay Heat Removal Capability at Davis-Besse Unit 1 While in a Refueling Mode" and IE Bulletin 80-12, "Decay Heat Removal System Operability" were issued to the industry to highlight NRC concern that licensees maintain diverse and redundant means of decay heat removal. In our response to IE Bulletin 80-12, we committed to lock out power to both RHR system suction valves whenever the RHR system is in service for RCS cooling, to prevent inadvertent valve closure and loss of suction to the RHR pumps. Review of a previous bulletin, IE Bulletin No. 79-20, in November 1979, "Loss of Non-class 1E Instrumentation and Control Power System Bus During Operation" identified that our system was vulnerable to loss of either 120 volt AC vital instrumentation busses Control Room Instrumentation Distribution (CRID) I or CRID IV, which would generate a close signal to its associated RHR suction valve, IMO-128 or ICM-129.

Cause of Event
The cause of the event has been attributed to an inadequate safety review performed at the time the procedure change was made. The impact of the change on the UFSAR and the T/Ss was not identified. Additionally, we used a procedure change to effectively accomplish a design change to the plant. It is believed that the processes in place for formal design changes would have been more likely to identify that the change could not be accomplished without NRC approval due to the impact on Technical Specifications.

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

Analysis of Event

The condition was originally reported pursuant to 10 CFR 50.72(b)(2)(i) as an unanalyzed condition on September 11, 1997 and as a condition outside the design basis of the plant. An interim LER was submitted pursuant to 10 CFR 50.73(a)(2)(ii) as a condition outside the design basis on October 10, 1997. This update is submitted to provide the cause and safety significance of the event.

When the procedure change was made in June 1980, to remove power from the valves, thereby defeating the interlock, it was noted that the "low pressure Reactor Coolant System (RCS) over-pressurization control system" was in service. This refers to the two pressurizer Power Operated Relief Valves (PORVs) with their setpoints reduced to low pressure conditions. The setpoint for these relief valves was 435 psig in 1980, as it is today.

These two pressure relief valves, together with the RHR suction safety valve set at 450 psig, were in the same configuration then as they are now. This configuration would have provided protection for the RCS and the RHR system from over-pressure conditions in mode 4 with the suction valves open and the auto-closure interlock blocked.

These three pressure relief devices form the basis for the Low Temperature Over-Pressure (LTOP) system. The plant heatup and cooldown procedures delineate the configuration requirements of LTOP based on plant configuration.

The LTOP system is supported by the analysis in WCAP-13235, "D. C. Cook Units 1 and 2, Analysis of Low Temperature Over-pressurization Mass Injection Events with Pressurizer Steam Bubble and RHR Relief Valve," dated March 1992. Protection of the RCS and the RHR is assured based on this analysis and the administrative controls delineated in the procedures, 1/2-OHP 4021.001.001, "Plant Heatup from Cold Shutdown to Hot Standby" and 1/2-OHP 4021.001.004, "Plant Cooldown from Hot Standby to Cold Shutdown."

The heatup and cooldown procedures were reviewed, along with procedures 1/2-OHP 4021.017.002, "Placing in service the RHR system" and 1/2-OHP 4021.017.003, "Removing from service the RHR system." The LTOP equipment (two pressurizer PORVs at low setpoint in addition to the RHR suction safety relief valve) is always placed in service prior to opening of the suction valves to place RHR in service for normal RCS cooling. The RHR system is removed from service, and the suction valves closed prior to the LTOP system being removed from service during heatup evolutions. Revisions dating back to the 1980 time period were researched and it was determined that LTOP equipment has always been placed in service by procedure, while the RHR is operating in a shutdown cooling configuration.

The RCS and the RHR systems were always and continue to be protected from over-pressurization by the LTOP equipment, at all times that the original RHR suction valve interlock was designed to protect the RHR system. This includes the periods in Mode 4 and 5 which were judged by this event to be outside the design basis specified in the UFSAR, and outside the T/Ss in Mode 4. Thus, this event did not significantly impact the health and safety of the public and did not pose a significant threat to plant safety.

NRC FORM 366A	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT CONTINUATION		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.

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Cook Nuclear Plant - Unit 1	50-315	97	-- 019 --	01	4 OF 4

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

Corrective Actions

For operation in Mode 4 with the normal RHR cooling configuration in place, the auto-closure feature associated with the RHR suction valves makes the plant unacceptably vulnerable to a loss of RHR cooling. Characterization of this vulnerability is based on both industry and Cook Nuclear Plant operating experience. As a result, a T/S amendment has been formally requested to remove the surveillance related to automatic valve closure on the RHR system suction valves from the RCS. The change was requested on September 19, 1997, via our letter designated AEP:NRC:1278.

A 10 CFR 50.59 safety evaluation was performed to evaluate operation in Mode 5 with the RHR suction valves open and the automatic closure interlock defeated by removing power from the valves. This evaluation concluded that this configuration does not constitute an unreviewed safety question. Appropriate reviews have been completed relative to the defeat of the automatic closure interlock; however, a design change package has been initiated to document it as a design change.

Given that a weak screening review was the root cause of this event, the following action related to 10 CFR 50.59 screenings and reviews has been initiated. Complete and accurate safety screenings are being stressed by the Operations Department Managers. Although the requirements for performance of safety screenings are proceduralized, the general steps to be taken to ensure the quality of those safety screenings were re-emphasized to all Operations procedures writers. These included:

- Computer based word searches are to be used to determine affected section of the FSAR or UFSAR.
- The affected sections should be reviewed using the hard copy of those documents, including all tables, graphs, figures, and flow diagrams.
- Review NRC correspondence and previous safety reviews to ensure that the proposed changes are consistent with the guidance in these documents.
- Review of Technical Specifications (T/S) shall include not only the T/S itself but also the surveillance requirements, the Bases section and the Administrative Section.

The procedure writers were also reminded of how to proceed if it becomes obvious during the screening process that the existing procedure did not satisfy the assumptions in the FSAR or UFSAR, and how to proceed with a proposed change that appears to affect the UFSAR and requires a complete 10CFR50.59 Safety Evaluation.

As discussed in the NRC's confirmatory action letter to the Cook Nuclear Plant, dated September 19, 1997, we are assessing the problems identified during the recent AE Design Inspection to determine whether these types of problems exist in other safety related systems and whether they affect system operation. In the longer term, we will evaluate our programs for improvements to assure these kinds of engineering problems are promptly identified, thoroughly evaluated, and resolved. The results of our reviews and assessments, as well as necessary preventive actions, will be communicated separately to the NRC.

Failed Component Identification

Not applicable

Previous Similar Events

315/97-012-01
315/97-016-01