REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8705120408 DOC. DATE: 87/04/08 NOTARIZED: NO DOCKET #
FACIL: 50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315
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SMITH, W. G. Indiana & Michigan Electric Co.
RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 87-005-00: on 870408, during cooldown & depressurization, pressure/temp limits of Tech Spec 3.4.9.1 exceeded. Caused by personnel error. Administrative controls & procedural enhancements incorporated. W/870408 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 5
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
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	REG FILE 02	1	1	RES SPEIS, T	1	1
	RGN3 FILE 01	1	1			
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
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	NSIC HARRIS, J	1	1	NSIC MAYS, G	1	1

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
D. C. Cook Nuclear Plant, Unit One	0 5 0 0 0 3 1 5	1 OF 0 4

TITLE (4) Technical Specification Reactor Coolant System Pressure/Temperature Limits Exceeded as a Result of Personnel Error

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 4	0 8	8 7	8 7	— 0 0 5 —	0 0	0 5	0 8	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER *(Specify in Abstract below and in Text, NRC Form 366A)*
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
K. R. Baker, Operations Superintendent	6 1 6	4 6 5 - 5 9 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)* (16)

On April 8, 1987, during a Unit One plant cooldown and depressurization the pressure/temperature limits identified in Technical Specification 3.4.9.1 were exceeded. This event occurred from 1915 hours to 2003 hours, a period of 48 minutes while the unit was in Mode 4 (Hot Shutdown).

This event is the result of personnel error on the part of the operating crew involved in this event (both licensed and non-licensed operators).

Upon determining that the pressure/temperature limits had been exceeded the plant was immediately depressurized and restored to within the Technical Specification limits. In addition, Administrative Controls and procedural enhancements have been incorporated to prevent recurrence of this event.

A subsequent evaluation of the pressures and temperatures experienced during this event indicated that these parameters were well within the acceptable operating range for the reactor vessel and therefore the structural integrity of the reactor vessel was not jeopardized.

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

Conditions Prior to Occurrence

Unit One was in Mode 4 (Hot Shutdown) with the Reactor Coolant System pressure and temperature at 1075 psig and 338.5°F respectively.

Description of Event

On April 8, 1987, at 1005 hours a plant cooldown and depressurization was commenced. This cooldown was to place the unit in Mode 5 (Cold Shutdown) to facilitate repairs to several Reactor Coolant System leakage sources identified within containment.

At 1830 hours on April 8, 1987, a decision was made to stabilize the plant at a temperature and pressure of approximately 300°F and 1100 psig while remaining within Technical Specification limits so that the repairs could be accomplished without causing undue flexing of the Reactor Coolant Pump (EIIS-AB-P) seal o-rings (EIIS-SEAL).

This information was relayed to the Unit One operating crew, who inturn stated that they could forsee no difficulty in reaching these parameters, based on a review of the Plant Safety System Display System (PSSD) (EIIS-CPU). Therefore at approximately 1845 hours, pressure was allowed to slowly increase to achieve the requested 1100 psig pressure; while continuing the cooldown.

During all cooldowns and heatups evolutions, the Reactor Coolant System pressure and temperature parameters are logged and plotted against the Technical Specification temperature/pressure limit curve every fifteen minutes. This function was assigned to a member of the Control Room crew. At approximately 1910 hours on April 8, 1987, this individual began taking the required shiftly surveillance readings. Due to the additional work load and the fact that the cooldown trend was good, he stopped plotting the parameters against the pressure/temperature curve, however he continued to log these parameters at 15 minute intervals. At 2000 hours on April 8, 1987, he resumed plotting these parameters starting with the first one not plotted at 1915 hours. At this time he realized that the pressure and temperature readings had exceeded the Technical Specification limits and immediately notified the Unit Supervisor. The Unit Supervisor immediately ordered the depressurization of the Reactor Coolant System. The Reactor Coolant System was subsequently restored to within the Technical Specification limits at 2003 hours on April 8, 1987. The largest disparity between actual conditions and the Technical Specification limits was at 306°F and 1160 psig compared to the Technical Specification limit of 306°F and 980 psig.

The Reactor Coolant System pressure and temperature had exeeded the Technical Specification limits for a period on approximately 48 minutes from 1915 hours to 2003 hours on April 8, 1987.

*U.S.GPO:1986-0-624-538/455

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

There were no inoperative systems, structures or components that contributed to this event.

Cause of Event

This event was the result of personnel error on the part of the operating crew. The Shift Supervisor, Unit Supervisor and Reactor Operator (Utility-licensed operator) failed to verify that the requested pressure and temperature parameters were acheivable by utilizing the Technical Specification pressure/temperature limit curve. Instead they relied upon a Plant Safety System Display terminal display (PSSD)(EIIS-CPU) to insure that the Technical Specification limits would be maintained. This display was subsequently determined to be non-conservative with respect to the Technical Specification parameters.

Secondly the Auxiliary Equipment Operator (utility non-licensed operator) stopped plotting the pressure temperature parameters for a period of approximately 45 minutes. This resulted from the increased work load created by the performance of shiftly surveillance. It should be noted that both the Unit Supervisor and Reactor Operator were confident that the Auxiliary Equipment Operator was logging and plotting the readings for they had noted him taking these parameters during this time frame.

Analysis of Event

This event was considered reportable under the criteria set forth in 10 CFR 50.73(a)(2)(i).

The RCS pressure and temperature conditions, noted below, were outside the acceptable region of the Technical Specification Cooldown Curve (Figure 3.4-3) which is valid for up to 12 EFPYs of operation.

TIME	PRESSURE (PSI)	TEMPERATURE °F
1915	1110	320.6
1930	1120	316.3
1945	1150	311.0
2000	1160	306.0

An evaluation to determine Reactor Coolant System integrity for continued operation has been performed. A new cooldown curve which reflects the actual conditions existing at the time of occurrence was developed by Southwest Research Institute at our request. The basis of the curve was established using Regulatory Guide 1.99 Rev. 1, a cooldown rate of 20°F/hr and eight EFPYs of reactor operation.

★U.S.GPO:1986-0-624-538/455

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

Review of the curve shows that the pressures and temperatures referenced above were will within the acceptable operating range for the reactor vessel, and therefore the structural integrity of the reactor vessel was not jeopardized.

Based upon the evaluation, it is concluded that this event did not pose a threat to the health and safety of the public.

Corrective Action

1) The pressure and temperature limits were immediately returned to their Technical Specification required parameters.

2) An operations memo has been written which requires that evolutions such as the cooldown/depressurization evolution are to be controlled by one operator who is specifically assigned to that task alone.

3) The PSSD computer display has been corrected to accurately reflect the proper temperature and pressure limits of Technical Specification 3.4.9.1. This has been accomplished on both units.

4. The graphs of both the heatup and cooldown curves have been redrawn to make them easier to use and more distinct with respect to the required Technical Specification limits.

Failed Components Identification

None

Previous Similar Events

RO-50-315/76-18

American Electric Power System

INDIANA & MICHIGAN ELECTRIC COMPANY

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May 8, 1987

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 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 Reporting System, the following report is being submitted:

87-005-0

Sincerely,

W. G. Smith, Jr.
Plant Manager

/afh

Attachment

cc: John E. Dolan
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