INDIANA MICHIGAN POWER®
A unit of American Electric Power

Indiana Michigan Power
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
AEP.com

July 18, 2007

AEP:NRC:2573-38
10 CFR 50.73
10 CFR 50.4

Docket No. 50-316

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Unit 2
LICENSEE EVENT REPORT 316/2006-006-00
FAILURE TO COMPLY WITH TECHNICAL SPECIFICATION 3.5.2,
ECCS – ECC OPERATION

In accordance with the criteria established by 10 CFR 50.73, Licensee Event Report System, the following report is being submitted:

LER 316/2006-006-00: "Failure to Comply with Technical Specification 3.5.2, ECCS – ECC Operation"

There are no commitments contained in this submittal.

Should you have any questions, please contact Ms. Susan D. Simpson, Regulatory Affairs Manager, at (269) 466-2428.

Sincerely,

Joseph N. Jensen
Site Vice President

RAM/rdw

Attachment

IE22

NRR

c: J. L. Caldwell, NRC Region III
K. D. Curry – AEP Ft. Wayne, w/o attachment
INPO Records Center
J. T. King, MPSC – w/o attachment
MDEQ – WHMD/RPMWS – w/o attachment
NRC Resident Inspector
P. S. Tam, NRC Washington DC

NRC Form 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES 6/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Donald C. Cook Nuclear Plant Unit 2	05000-316	1 of 4

4. TITLE

Failure to Comply with Technical Specification 3.5.2, ECCS – Operating

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
04	19	2006	2006	-- 006 --	00	07	18	2007	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☒ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100%	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A
	☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Susan D. Simpson, Regulatory Affairs Manager	(269) 466-2428

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
Other	RHR	2SV104E 2SV104W	Crosby	Yes					

14. SUPPLEMENTAL REPORT EXPECTED			15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (If Yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On April 10, 2006, Donald C. Cook Nuclear Plant (CNP) tested the Unit 2 "B" Residual Heat Removal (RHR) system heat exchanger outlet safety valve and on April 19, 2006, CNP tested the Unit 2 "A" RHR system heat exchanger outlet safety valve. The safety valves for both trains of RHR had an unsatisfactory as-found lift pressure test (high).

Technical Specification (TS) 3.5, Emergency Core Cooling Systems (ECCS); 3.5.2, ECCS – Operating, requires two trains of ECCS to be OPERABLE when in MODES 1, 2, and 3. When one or more trains are inoperable, Condition A requires that the inoperable train(s) be restored to OPERABLE status within 72 hours. Since a similar cause was determined for the unsatisfactory as-found lift pressures, this condition may have arisen over a period of time, and there is a likelihood that the affected safety valves on both trains of RHR may not have been OPERABLE during plant operation for a time longer that allowed by TS. Therefore, this occurrence is considered reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by CNP's TS and 10 CFR 50.73(a)(2)(vii) as a common cause of inoperability.

The apparent cause of the occurrence is nozzle disc bonding. Corrective action included replacing both affected safety valves with new valves and the system was declared OPERABLE. Additionally, Indiana Michigan Power Company (I&M) performed an extent of condition review for the corresponding valves in CNP Unit 1 and determined the event was isolated to RHR safety valves in Unit 2. This licensee event report is being submitted greater than 60 days after the event due to CNP's failure to recognize the multiple test failures constituted a reportable condition.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Donald C. Cook Nuclear Plant Unit 2	05000316	2006 --	006 --	00	2 of 4

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

Conditions Prior to Event

Unit 2 - 100% power

Description of Event

On April 10, 2006, Donald C. Cook Nuclear Plant (CNP) tested the Unit 2 "B" Residual Heat Removal (RHR) system heat exchanger outlet safety valve and on April 19, 2006, CNP tested the Unit 2 "A" RHR system heat exchanger outlet safety valve. The safety valves for both trains of RHR had an unsatisfactory as-found lift pressure test (high).

Technical Specification (TS) 3.5, Emergency Core Cooling Systems (ECCS); 3.5.2, ECCS - Operating, requires two trains of ECCS to be OPERABLE when in MODES 1, 2, and 3. When one or more trains are inoperable, Condition A requires that the inoperable train(s) be restored to OPERABLE status within 72 hours. Since a similar cause was determined for the unsatisfactory as-found lift pressures, this condition may have arisen over a period of time, and there is a likelihood that the affected safety valves on both trains of RHR may not have been OPERABLE during plant operation for a time longer than allowed by TS. Therefore, this occurrence is considered reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by CNP's TS and 10 CFR 50.73(a)(2)(vii) as a common cause of inoperability.

10 CFR 50.73(a) requires licensees to submit licensee event reports (LER) within 60 days of discovery of the event. This LER is being submitted greater than 60 days after the event due to CNP's failure to recognize that the multiple test failures constituted a reportable condition.

Cause of Event

The apparent cause of the occurrence is nozzle disc bonding.

Analysis of Event

As described above, both Unit 2 RHR system heat exchanger outlet safety valves failed to initially lift at 1.25 times their design setpoint (design setpoint is 600 psig).

The failure of these safety valves to lift at the setpoint pressure has no direct influence on the behavior of other components, equipment, or conditions. Thus, these failures do not increase the probability of any initiating event in the CNP Probabilistic Risk Assessment (PRA) model and have no impact on plant risk from that standpoint. Failure of these safety valves could impact mitigation capabilities of the RHR and safety injection (SI) [BJ] systems under specific circumstances when these systems would be required to operate under high pressure

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

conditions. Events which could lead to these conditions include a small break loss-of-coolant accident (SBLOCA) and an extended loss of heat sink accident. Following one of these accidents, the potential exists in the SI-to-RHR injection piping configuration for the SI system to over pressurize the RHR system if the Reactor Coolant System (RCS) remains above 600 psig. With either an SBLOCA or extended loss of heat sink, an additional failure would have to occur for the failure of one of these safety valves to affect its associated RHR train. Specifically, the RHR check valve upstream of the SI-to-RHR tie-in would also have to fail to prevent backflow/leakage. Thus, for the SI system to overpressurize the RHR system causing rupture or significant leakage of the RHR piping, the following would be required:

- an SBLOCA or loss of heat sink,
- SI system operation against an RCS pressure above 600 psig, and
- failure of the associated RHR train-related check valve.

Control Room alarms would alert operators to abnormal pressure conditions in the RHR pump discharge lines. The annunicator response procedures provide direction for operator action. In the worst case, a single SI and RHR train could be disabled due to a large rupture. If the piping ruptured and spilled water to either the SI or RHR pump rooms, control room alarms would indicate significant leakage in the associated room. Room Sump Level Alarms indicate leakage in the room. The annunciator response procedure directs operators to determine the source of the leakage and isolate it. Given the training that Operations personnel receive on RHR and SI system operation, they would be expected to recognize that RHR/SI system operation was causing this alarm and take action to stop or minimize the impact.

Both RHR safety valves did lift on a second attempt within 5% of the design setpoint and appeared to reseat satisfactorily based on subsequent lift tests. Given this behavior, there is a reasonable probability that the valves would have functioned to protect the RHR system. Valve opening would avoid significant RHR train damage and associated leakage.

Neither of the RHR safety valves is explicitly included in the PRA model. In order for an RHR safety valve to actuate, failure of an RHR check valve upstream of the SI-to-RHR tie-in must be assumed to cause failure of the associated SI train. Implicitly, the PRA model does credit the outflow from these valves during Interfacing System LOCA events. However, in such events, full RCS pressure would be applied to the valve and would likely open it fully. On this basis, there is no quantitative PRA impact of the RHR heat exchanger outlet safety valves lifting above their design setpoint. Nonetheless, barriers exist to limit the impact of this condition. Specifically:

- particular events are required (extended loss of auxiliary feedwater or SBLOCA),
- certain RCS conditions (RCS > 600 psig) are required,
- an additional failure (failure of the East RHR To Reactor Coolant Loops #1 and #4 Check Valve; or West RHR To Reactor Coolant Loops #2 & #3 Check Valve) is required, and

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

- redundant alarms exist to alert personnel to isolate/mitigate the leak.

The likelihood of these conditions occurring simultaneously is extremely small based upon probabilistic insights.

Corrective Actions

Both Unit 2 RHR system heat exchanger outlet safety valves were replaced with new, pre-tested valves and were declared operable.

The Unit 1 and Unit 2 RHR safety valves are necessary for RHR heat exchanger overpressure protection. Expansion of the test population was performed in accordance with I&M's ISI testing program to identify the extent of condition and included one of the two Unit 1 valves. This valve passed and did not exhibit indications of nozzle disc bonding.

I&M will continue to work with its vendors and industry peers to ensure it fully understands and addresses this condition, with expanded testing and adjustments to be performed as appropriate.

Previous Similar Events

05000316/2006-002-00, MSSV Trevi Testing Failures

The causal evaluation and corrective actions for this previous similar event have been reviewed. Based on the differing system operating parameters for the main steam safety valves and the RHR heat exchanger outlet safety valves, I&M has determined that the extent of condition review and corrective actions taken for LER 05000316/2006-002-00 could not have reasonably been expected to prevent the event being reported in this LER.