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Indiana Michigan Power
Cook Nuclear Plant
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October 22, 2007

AEP:NRC:2573-40
10 CFR 50.73
10 CFR 50.4

Docket No. 50-315

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

Donald C. Cook Nuclear Plant Unit 1
LICENSEE EVENT REPORT 315/2007-001-00
UNIT 1 AUTOMATIC REACTOR TRIP

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

LER 315/2007-001-00: "Unit 1 Automatic Reactor Trip"

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/jen

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.

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4. TITLE

Unit 1 Automatic Reactor Trip

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
08	28	2007	2007	- 001 -	00	10	22	2007	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

☐ 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)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 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

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 August 28, 2007, at 1354 hours, Donald C. Cook Nuclear Plant (CNP) Unit 1 received a reactor trip and main turbine trip signal as a result of low Steam Generator #11 water level coincident with a steam flow - feedwater flow mismatch. All control rods fully inserted and the Auxiliary Feedwater System (AFW) started and performed as designed.

The reactor trip was uncomplicated and all major plant components functioned as designed. The reactor trip was reported in accordance with 10 CFR 50.72(b)(2)(iv)(B) and the AFW actuation was reported in accordance with 10 CFR 50.72(b)(3)(iv)(A). The reactor trip and AFW actuation are reportable as a Licensee Event Report in accordance with 10 CFR 50.73(a)(2)(iv)(A).

The trip was the result of an over voltage condition on one of the Digital Control System (DCS) power supplies. The over voltage condition caused the controller for the East Main Feed Pump control valve to automatically trip, causing an inadequate feedwater flow condition to the steam generators. Corrective actions included replacement of the DCS power supply.

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

Conditions Prior to Event

100 percent reactor power

Description of Event

On August 28, 2007, at 1354 hours, Donald C. Cook Nuclear Plant (CNP) Unit 1 received a reactor trip and main turbine trip signal as a result of low Steam Generator #11 water level coincident with a steam flow - feedwater flow mismatch. All control rods [AA] fully inserted and the Auxiliary Feedwater System (AFW) [BA] started and performed as designed.

The reactor trip was uncomplicated and all major plant components functioned as designed. The reactor trip was reported in accordance with 10 CFR 50.72(b)(2)(iv)(B) and the AFW actuation was reported in accordance with 10 CFR 50.72(b)(3)(iv)(A). The reactor trip and AFW actuation are reportable as a Licensee Event Report in accordance with 10 CFR 50.73(a)(2)(iv)(A).

The initial design of the Feed Pump Turbine Digital Control System (DCS) [JJ] did not include an air conditioning system. However, shortly after initial installation and operation of the DCS system, CNP recognized the need for and installed an air conditioning system to supply cool air to DCS cabinets. The installed air conditioning system failed, causing temperatures to elevate in the DCS cabinet resulting in a DCS trouble alarm in the Control Room. CNP personnel were troubleshooting the alarm and identified the DCS cabinet air conditioning had failed. Actions were implemented to restore the cabinet's air conditioning at about the time the trip occurred and cabinet temperatures returned to normal shortly thereafter.

The DCS has two power supplies, which are designed to work in parallel and share the DCS power loads. Each of the power supplies is capable of carrying the full load upon loss of the other power supply. One of the two DCS power supplies had degraded and become overly temperature sensitive. This degraded power supply, when exposed to elevated temperatures, produced an elevated output voltage. The over voltage condition caused the alternate power supply to default to a zero output condition, per design, and the over voltage supply then carried the full load. The degraded power supply output voltage increased to a point that the protective circuit on the East Main Feed Pump DCS controller tripped on high supply voltage level. The trip of the controller caused the loss of the East Main Feed Pump. This resulted in inadequate feedwater flow to the steam generators. Subsequently, the reactor and main turbine tripped due to low Steam Generator #11 water level coincident with a steam flow - feedwater flow mismatch.

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

Cause of Event

The design of the DCS failed to ensure that DCS Power Supply Trip Setpoints were adequate to ensure that a degraded/failing power supply would not cause loss of downstream components, such as the feed pump control cards, prior to the power supply tripping and transferring the downstream loads to the alternate power supply.

In February 2007, CNP recognized that the DCS power supplies were temperature sensitive and that one power supply was degraded. Cabinet air conditioning was installed to ensure the power supplies were maintained within an acceptable temperature band. The failure of the DCS cabinet air conditioner in August 2007 resulted in elevated temperatures within the DCS cabinets, which in turn caused the degraded power supply to produce an over voltage condition. This failure created an elevated output voltage from the power supply, which ultimately resulted in the tripping of the controller for the East Main Feed Pump. This adverse impact on the Main Feed Pump controllers was not previously considered in the design or previous evaluation of the degraded power supply. Therefore, actions were not implemented to replace the power supply prior to the next scheduled refueling outage.

Analysis of Event

This event was consistent with the primary success path in the existing CNP risk analysis associated with unplanned reactor trips with the main condenser available. There were no significant post-trip complications or failures. This trip resulted from the loss of the Unit 1 East Main Feedwater Pump as a result of a short-term overheating condition in the DCS cabinets. The three AFW pumps in the CNP Unit 1 AFW System, the cross-ties from the two Motor Driven AFW Pumps in the CNP Unit 2 AFW System, and the CNP Unit 1 West Main Feed Pump, along with the remainder of the Unit 1 Condensate System [SD], were available to supply feedwater to the Unit 1 Steam Generators (SG) for heat removal.

Loss of the East Main Feedwater Pump causes no appreciable increase in CNP's estimated zero test and maintenance core damage frequency or large early release frequency. This low risk significance is due to the equipment available to supply feedwater to a unit's SG in a post-trip condition. Failure of the DCS cabinet air conditioning and the subsequent reactor trip did not degrade systems that maintain core decay heat removal, assure containment integrity, or maintain defense-in-depth and safety margins. Operators took procedurally-directed actions and responded to the transient in an appropriate and timely manner, resulting in a safe and stable plant configuration. Automatic post-trip features functioned dependably.

For these reasons, this event is not considered to represent a significant risk to the plant or surrounding population.

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

Corrective Actions

The DCS power supplies were replaced.

The air conditioning system was returned to operational status.

The evaluation of why CNP failed to recognize design limitations of the DCS system and failed to recognize system vulnerability associated with the degraded power supply are captured within the CNP corrective action program.

Previous Similar Events

The following LERs identify automatic reactor trips in the past three years. The causes of these reactor trips were not similar in nature to the cause of this trip.

05000316/2004-001-00,	Automatic Reactor Trip Due to RPS Actuation, While Manipulating Reactor Trip Bypass Breaker
05000316/2004-002-00,	Unplanned Automatic Reactor Protection System Actuation Due to Feedwater Transient During a Power Reduction
05000316/2005-001-00,	Reactor Trip from RCP Bus Undervoltage Signal Complicated by Diesel Generator Output Breaker Failure