INDIANA MICHIGAN POWER

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November 13, 2008

AEP-NRC-2008-42
10 CFR 50.73

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 315/2008-006-00
MANUAL REACTOR TRIP DUE TO MAIN TURBINE HIGH VIBRATION

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

LER 315/2008-006-00: "Manual Reactor Trip Due to Main Turbine High Vibrations."

There are no commitments contained in this submittal.

Should you have any questions, please contact Mr. John A. Zwolinski, Regulatory Affairs Manager, at (269) 466-2478.

Sincerely,

Lawrence J. Weber
Site Vice President

JEN/rdw

Attachment

c: T. A. Beltz – NRC Washington DC
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/RPS, w/o attachment
NRC Resident Inspector

IE22
NRR

NRC Form 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES 08/31/2010

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.

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
|---|---|---|
| Donald C. Cook Nuclear Plant, Unit 1 | 05000315 | 1 of 5 |

**4. TITLE**

Manual Reactor Trip Due To Main Turbine High Vibration

| 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 |
| 09 | 20 | 2008 | 2008 | -- 006 -- | 00 | 11 | 13 | 2008 | 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) |
|---|---|
| John A. Zwolinski, Regulatory Affairs Manager | (269) 466-2478 |

**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 |
|---|---|---|---|---|---|---|---|
| X | YES (If Yes, complete EXPECTED SUBMISSION DATE). | | NO | | 02 | 26 | 2009 |

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

On September 20, 2008, at 2005 hours, Donald C. Cook Nuclear Plant Unit 1 control room operators initiated a manual reactor trip from 100% power following high-high vibration alarms on all main turbine supervisory instrumentation vibration points. In addition, numerous alarms on components in the condensate and feedwater flowpaths along with severe vibration and rumbling were noted by control room operators.

Upon the reactor trip, all control rods fully inserted, the Auxiliary Feedwater System (AFW) and other major plant components functioned as designed. The main generator was reported to be on fire, and the Shift Manager declared an Unusual Event (UE) when the fire was not extinguished within 15 minutes. The fire was declared extinguished at 2028 hours. At 2044 hours, the operators tripped closed the Main Steam Isolation Valves (MSIV) in accordance with procedure. Closure was based on an observed cooldown of the Reactor Coolant System (RCS). RCS temperature recovered following closure of the MSIVs.

The reactor trip, UE, AFW actuation, and MSIV closure were reported in accordance with 10 CFR 50.72(b)(2)(iv)(B), 10 CFR 50.72(a)(1)(i), and 10 CFR 50.72(b)(3)(iv)(A). The reactor trip, AFW actuation and MSIV closure are reportable as a Licensee Event Report (LER) in accordance with 10 CFR 50.73(a)(2)(iv)(A).

The cause of the reactor trip was high-high vibration of the main turbine. A Root Cause Evaluation of the vibration is ongoing. Corrective actions have commenced and include repair of the main turbine and generator. Additional corrective actions may be identified and reported in a supplement to this LER.

| 1. FACILITY NAME | 2. DOCKET | 6. LER NUMBER | | | 3. PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| Donald C. Cook Nuclear Plant, Unit 1 | 05000315 | 2008 | -- 006 | -- 00 | 2 of 5 |

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

**Conditions Prior to Event**

Unit 1 was in Mode 1 at 100% power.

**Description of Event**

On September 20, 2008, at 2005 hours, Donald C. Cook Nuclear Plant (CNP) Unit 1 operators initiated a manual reactor trip from 100% power when all main turbine bearing vibration monitors [IV] indicated high-high vibration. Upon the manual reactor trip, the reactor protection system [JG] operated as designed, the Auxiliary Feedwater System (AFW) [BA] started and performed as designed, and other major plant components functioned as designed, with the exception of the main generator [TB] trip relays [RLY].

The main generator trip from a main turbine [TA] trip has a designed 30-second delay for most trip scenarios. For this event, the main generator trip should have been delayed 30 seconds, but instead actuated immediately following the turbine trip. This was due to the actuation of the generator overall differential auxiliary relays [RLY 87]. Main turbine vibration mechanically induced an actuation of several of the main turbine exhaust high-high temperature switches [TS]. The high-high temperature switch input causes a bypass of the 30-second delay, which allowed the main generator to trip immediately following the turbine trip.

At the onset of the event, numerous control room annunciators [ANN] were received on the balance of plant equipment panels; additionally, the control room operators could feel vibration and hear loud rumbling coming from the area of the main turbine. Control room operators noted that all vibration points on the main turbine supervisory panel indicated high-high vibration. At 2005 hours, the operators performed a manual trip of the reactor and entered the Emergency Operating Procedure for a reactor trip. The main turbine automatically tripped as designed in response to the manual reactor trip. The main generator tripped immediately following the turbine trip as described above.

Because of the high-high main turbine bearing vibration, control room operators opened the main condenser [SG] vacuum breakers [V] in order to stop main turbine rotation more quickly.

Soon after the reactor trip, the main generator was reported to be on fire. At 2018 hours, the Shift Manager (SM) declared an Unusual Event (UE) based on Initiating Conditions H-4, fire in the protected area not extinguished within 15 minutes, and H-5, toxic or flammable gas release affecting plant operation (generator hydrogen). In addition, the SM had the Technical Support Center (TSC) activated. The UE was reported in accordance with 10 CFR 50.72(a)(2)(iv)(A). At 2020 hours, lube oil fire water spray [KP] to the main turbine was initiated. Main turbine lube oil pumps [TD] were secured at 2027 hours. The main generator fire was reported to be extinguished at 2028 hours, and fire protection water spray was secured at 2035 hours. Generator hydrogen [LJ] was manually isolated at 2040 hours.

| 1. FACILITY NAME | 2. DOCKET | 6. LER NUMBER | | | 3. PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| Donald C. Cook Nuclear Plant, Unit 1 | 05000315 | 2008 | -- 006 -- | 00 | 3 of 5 |

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

At 2044 hours, the operators tripped closed the Main Steam Isolation Valves (MSIV) [ISV] in accordance with the reactor trip response procedure; this was based on an observed Reactor Coolant System (RCS) [AB] cooldown. Investigation following the event revealed two reasons for the RCS average temperature (Tavg) lowering to 528 degrees Fahrenheit:

1) AFW flow reduction was performed in two smaller steps rather than one larger step. This method contributed to the RCS cooldown; however, even if actions to reduce flow were pursued more aggressively, they would not have been sufficient to arrest the cooldown and prevent closure of the MSIVs.

2) Unit 1 was supplying auxiliary steam [SA] loads at the time of the trip, and there was a delay of approximately 20 minutes before auxiliary steam loads were transferred to Unit 2. This delayed response was due to the Unit 2 operating crew responding to a secondary plant transient caused by the simultaneous start of several standby condensate [SD] pumps [P] and performing the required notifications of the UE. Vibration actuation of the condensate pump auto start pressure switches [PS] is the suspected cause of the unexpected automatic starts of the standby pumps. Prior to transferring auxiliary steam loads to Unit 2, auxiliary steam system safety valves were lifting. This extra steam flow was a significant contributor to the RCS cooldown. While the investigation into the reason for the lifting of the safety valves is ongoing, evidence shows that it was likely due to a sticking auxiliary steam pressure reducing valve [PCV] allowing header pressure to rise following the rise in main steam pressure.

RCS cool down was arrested and RCS temperature recovered within approximately 20 minutes following closure of the MSIVs. With the main steam lines isolated, decay heat was being removed with the steam generator Power Operated Relief Valves (PORVs) [RV] dumping steam to the atmosphere.

At 2113 hours, the Technical Support Center was activated.

At 2125 hours, the fire water header pressure low annunciator was received. Fire Protection personnel reported that the North Fire Water Storage Tank [KP] [TK] was empty. At this time, all three fire water pumps were stopped. Subsequent reports noted a breach in the buried fire header, outside on the west side of the turbine building. Fire protection personnel isolated the leak and established the required temporary fire suppression hose at 2309 hours. The breach was subsequently determined to have been caused by a separated Victaulic coupling [CPLG]. Follow-up investigation identified that the East Diesel Driven Fire Pump [P] was was damaged due to running with no flow.

The reactor trip, UE, AFW actuation, and MSIV closure were reported in accordance with 10 CFR 50.72(b)(2)(iv)(B), 10 CFR 50.72(a)(1)(i), and 10 CFR 50.72(b)(3)(iv)(A). The reactor trip, AFW actuation and MSIV closure are reportable as a Licensee Event Report (LER) in accordance with 10 CFR 50.73(a)(2)(iv)(A).

| 1. FACILITY NAME | 2. DOCKET | 6. LER NUMBER | | | 3. PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| Donald C. Cook Nuclear Plant, Unit 1 | 05000315 | 2008 | -- 006 | -- 00 | 4 of 5 |

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

At 0409 hours, the control room staff was informed by TSC personnel that the Site Emergency Coordinator terminated the UE.

**Cause of Event**

The reactor trip was manually initiated due to high-high vibration on the main turbine bearings. The root cause evaluation of the high-high vibration on the main turbine bearings is ongoing. A supplement to this LER will be submitted when the root cause evaluation has been completed.

**Analysis of Event**

The safety significance analysis is based on the Reactor Trip Review and the Preliminary CNP Unit 1 Post September 20, 2008, Main Turbine Failure Risk Assessment.

Probabilistic Risk Assessment (PRA) personnel walked down areas containing safety-related systems, structures, and components in the turbine building [NM] the morning of September 22, 2008, and there was no evidence of any significant impact to the AFW pumps and rooms, the Unit 1 switchgear areas that are in, or connect to, the Unit 1 main turbine area, or the Unit 1 Emergency Diesel Generator [EK] access hallway. The Unit 1 4160 Volt switchgear room was clean and dry, the Unit 1 600 Volt switchgear room contained some amount of large black dust-like material. This material appears to be soot brought in via the area ventilation system which always has a supply fan [FAN] in service. The AFW pump rooms in both units were clean and dry, as was the Emergency Diesel Generator access hallway. Thus, from a qualitative standpoint, there did not appear to be any serious challenge to successfully shutting the unit down and removing decay heat or any reason to consider that equipment located in those rooms had any functions impaired.

The Unit 2 Plant Air Compressor (PAC) [CMP] was in service and the Unit 1 PAC was in stand-by at the time of the event. The Unit 2 PAC operated satisfactorily throughout the event. After the event, the Unit 1 PAC was subsequently considered unavailable when it was found that its auxiliary oil pump [P] had lost power.

Extensive plant walkdowns were performed by structural, electrical, and mechanical personnel to identify and report damage to plant equipment. The result of these walkdowns was that no significant damage was found to safety-related structures, systems, or components as a result of this trip.

In summary, based on review of the control room log, the plant walkdown, and the Reactor Trip Review abstract, all plant systems performed as designed to shut down the unit and remove decay heat, and the trip event did not represent a significant risk. The only PRA functions affected were those related to the unavailability of the Unit 1 PAC and the inability to manually use the main feedwater system [SJ]. Main feedwater was unavailable since main condenser vacuum had been broken, and main feedwater's risk associated function is to be manually used in Emergency Operating Procedures to mitigate a potential complete AFW failure. PRA-STUDY-053 estimated the change in Conditional Core Damage Probability (delta-CCDP) using the current Safety

| 1. FACILITY NAME | 2. DOCKET | 6. LER NUMBER | | | 3. PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| Donald C. Cook Nuclear Plant, Unit 1 | 05000315 | 2008 | -- 006 | -- 00 | 5 of 5 |

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

Monitor CNP PRA model. The delta-CCDP associated with this event was determined to be 7.811E-07. This delta-CCDP value represents a nominal increase in plant risk from expected plant performance consistent with Nuclear Regulatory Commission Inspection Manual Chapter 0308, Reactor Oversight Process (ROP) Basis Document, and Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant Specific Changes to the Licensing Basis.

**Corrective Actions**

Immediate actions:

Control room operators manually tripped the reactor upon receipt of high-high main turbine bearing vibration alarms and indications.

Main turbine trip was verified and main condenser vacuum was broken in order to stop main turbine rotation more quickly.

In response to the fire, control room operators initiated lube oil fire water spray to the main turbine.

Main turbine and generator repairs have been initiated.

The fire header coupling has been repaired, and the East Diesel Driven Fire Pump has been replaced.

Additional corrective actions will be determined based on the results of ongoing plant and component evaluations.

**Previous Similar Events**

LER 05000315-2008-001-00 documents a Unit 1 manual trip due to high main turbine bearing vibration in February of 2008.