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1CAN061002

June 22, 2010

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Licensee Event Report 50-313/2010-004-00
Arkansas Nuclear One – Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

In accordance with 50.73(a)(2)(iv)(A), enclosed is the subject report concerning an automatic Reactor Protection System (RPS) actuation that resulted in a reactor trip.

There are no commitments contained in this submittal.

Sincerely,

DBB/slp

Enclosure – LER 50-313/2010-004-00

cc: Mr. Elmo Collins
Regional Administrator
U. S. Nuclear Regulatory Commission
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NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

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



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**4. TITLE** Automatic Reactor Protection System Actuation that Resulted in a Reactor Trip Due to Inadequate Procedure Use and Adherence and Workers Acting Independently

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| 04 | 25 | 2010 | 2010 | 004 | 00 | 06 | 22 | 2010 | FACILITY NAME | DOCKET NUMBER |

**9. OPERATING MODE:** 1

**10. POWER LEVEL:** 19.5

**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**

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| David B. Bice, Acting Manager, Licensing | 479-858-4710 |

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

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| A | | | | | | | | | |

| 14. SUPPLEMENTAL REPORT EXPECTED | | 15. EXPECTED SUBMISSION DATE | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| ☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE) | ☒ NO | | | | |

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

On April 25, 2010, at approximately 2126 Central Daylight Time (CDT), with the plant operating at ~19.5% heat balance power following a refueling outage (1R22), a Reactor Protection System (RPS) actuation during Nuclear Instrument (NI) calibration resulted in a reactor trip at Arkansas Nuclear One, Unit 1. During performance of the NI calibration, a large neutron error signal developed resulting in a mismatch between selected power and indicated power which resulted in Group 7 control rods withdrawing for 38 seconds. The rod withdrawal initiated an RPS trip on 2 of the 4 channels, resulting in an automatic reactor trip. The failure to appropriately use and adhere to procedures was determined to be the major factor in this event. Procedure steps were not performed as written and not read verbatim to the operator as required by procedure guidance. As a result, immediate stand-downs with involved departments were conducted and additional control room oversight was established.

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## NARRATIVE

A. Plant Status

At the time this condition was identified, Arkansas Nuclear One, Unit 1 (ANO-1) was operating at ~19.5% heat balance power.

B. Event Description

On April 25, 2010, at 1918 Central Daylight Time (CDT), ANO-1 ended refueling outage (1R22) when the main generator output breakers [TB] [BKR] were closed tying the main generator to the grid. Following generator startup, power was raised and the reactor [RCT] was stabilized at ~19.5% heat balance power. Due to low power reactor physics testing, excore Nuclear Instrument (NI)[IG] gains had been conservatively set high resulting in a deviation between excore NI and heat balance power of approximately 11%. As required by station operating procedure OP-1102.004, "Power Operations", Operations requested the Instrument & Controls (I&C) department to perform an NI calibration for a deviation of more than 4%.

According to post-event interviews conducted, the lead I&C technician established initial conditions for performance of the NI calibration, then using a non-specific communication stated to the Operations Control Board Operator Turbine (CBOT), "we are ready to place the ICS (Integrated Control System) [JA] to manual." Without proper verification and peer check the CBOT responded, "ICS is in manual." In response the lead I&C technician signed off the following procedural step in the NI calibration procedure, OP-1304.032, "Unit 1 Power Range Linear Amp Calibration at Power":

> *Have operations place the Reactor Demand H/A and Diamond Rod Control Stations into Manual using OP-1104.004, ICS Operating Procedure.*

The Reactor Demand station was in manual. Contrary to the procedural requirements the Diamond Rod Control Station was not in manual but instead was in automatic as would be expected in this mode of operation.

I&C then continued with the NI calibration in accordance with OP-1304.032, a frequently performed procedure which is normally performed twice weekly. All four NI channels were to be adjusted in the downward direction to the heat balance value of 19%. After completion of "A" Channel Reactor Protection System (RPS)[JC], the "B" Channel RPS was placed in manual bypass at ~2125 CDT. When the I&C technician placed the switch on the Power Range Test Module in the Test/Operate position, a large negative neutron error resulted in an automatic rod withdrawal.

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**NARRATIVE**

B. Event Description - continued

Indicated excore NI power had rose from 30% to 49.55% resulting in a high neutron flux trip on RPS Channel “C”. The amount of energy added to the Reactor Coolant System (RCS) [AB] by the control rod withdrawal caused RCS temperature and pressure to rise resulting in a high RCS pressure trip on "A" Channel RPS. The trip of these two channels caused an automatic reactor trip at ~2126 CDT.

Post reactor trip system response was normal as expected. All control rods fully inserted into the core and no safety systems, other than RPS actuated. Emergency Feedwater (EFW) [BA] did not actuate and was not needed. No primary safety valves lifted. Seven secondary safety valves lifted and subsequently reseated. The plant stabilized in Hot Standby (Mode 3) conditions.

C. Root Cause

The failure to appropriately use and adhere to procedures was determined to be the major factor in this event. The procedure step was not performed as written and not read verbatim to the operator as required by procedure guidance. The I&C procedure specifically required Operations to place the Reactor Demand H/A and Diamond Rod Control Station into manual using OP-1105.004, "Integrated Control System Operating Procedure." The lead technician should have communicated the procedure step exactly as written. This procedure step clearly stated that the Diamond Rod Control was to be placed in manual.

Additionally, the CBOT made the decision to act independently without guidance from supervision by allowing the I&C technician to proceed by communicating an interpretable, non-standard message. The CBOT also failed to properly verify and have his action peer checked by an independent control room operator.

D. Corrective Action

As a result of this event, an immediate Operations and Maintenance supervisor meeting was conducted. This meeting was followed by a site stand-down. Additional control room oversight was established which included focus on Operations and Maintenance interface.

Interim guidance was established for work activities conducted in the control room to ensure plant conditions are appropriately established prior to commencing maintenance.

The lessons learned from this event, supervisory skills, and human performance training are being incorporated into Operations and Maintenance requalification training.

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## NARRATIVE

E. Safety Significance

A risk evaluation for the ANO-1 automatic reactor trip during NI calibration was performed to evaluate the condition with respect to safety risk for the general public, nuclear safety, radiological safety, and industrial safety. The change in core damage risk for ANO-1 from the event was calculated and compared to the criteria for significance, as defined in MD 8.3, RG 1.174, and EPRI TR-105396. The risk for the transient event is deemed insignificant. Post reactor trip system response was normal as expected. All control rods fully inserted into the core and no safety systems, other than RPS actuated. EFW did not actuate and was not needed and no primary safety valves lifted. Seven secondary safety valves lifted and subsequently reseated. The plant stabilized in Hot Standby (Mode 3) conditions.

The increased risk of having a Pellet Cladding Interaction (PCI) fuel failure at ANO-1 due to the power increase is judged to be insignificant. This is based on review of the available data from ANO-1, mechanisms that can result in PCI fuel failures, engineering judgment, and discussions with the vendor. RPS actuated as designed to shutdown the reactor. Post reactor trip system response was normal for the plant and no industrial safety issues occurred. There were no radiological safety issues created by this event and at no time during the course of the event was the general safety of the public compromised.

F. Basis for Reportability

Actuation of the RPS is reportable under 10 CFR 50.73(a)(2)(iv)(A). Additionally, actuation of the RPS when the reactor is critical is reportable under 10 CFR 50.72(b)(2)(iv)(B). An immediate notification was made to the NRC Operations Center on April 25, 2010.

G. Additional Information

During the last five years, there was one other previous similar event reported as LER-2005-003-00 for ANO-1 concerning actuation of the RPS; however, the root cause was not similar. This condition was due to a Main Turbine trip caused by low turbine bearing lube oil pressure.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].