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February 24, 2006

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
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Subject: Licensee Event Report 50-313/2005-003-00
Arkansas Nuclear One – Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

In accordance with 10CFR50.73(a)(2)(iv)(A), enclosed is the subject report concerning an automatic actuation of the Reactor Protection System and an invalid actuation of the Emergency Feedwater System.

This correspondence contains no commitments.

Sincerely,

Dale E. James
Manager, Licensing

DEJ/fpv

Enclosure

JE22

cc: Dr. Bruce S. Mallett
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| NRC FORM 366 (6-2004) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB: NO. 3150-0104 | EXPIRES: 06/30/2007 |
|---|---|---|---|

# LICENSEE EVENT REPORT (LER)

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



| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
|---|---|---|
| Arkansas Nuclear One – Unit 1 | 05000313 | 1 OF 4 |

**4. TITLE** Reactor Trip due to Automatic Actuation of the Reactor Protection System on Main Turbine Trip and Invalid Actuation of the Emergency Feedwater System

| 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 05000 |
| 12 | 26 | 2005 | 2005 | - 003 - | 00 | 02 | 24 | 2006 | FACILITY NAME | DOCKET NUMBER 05000 |

**9. OPERATING MODE:** 1

**10. POWER LEVEL:** 95

**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 |
| ☐ 20.2203(a)(2)(vi) | ☐ 50.73(a)(2)(i)(B) | ☐ 50.73(a)(2)(v)(D) | Specify in Abstract below or in NRC Form 366A |

**12. LICENSEE CONTACT FOR THIS LER**

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Fred Van Buskirk, Nuclear Safety and Licensing Specialist | 479-858-3155 |

**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 |
|---|---|---|---|---|---|---|---|---|---|
| B | TD | V | W120 | Y | | | | | |

| 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)*

At 1047 CST, on December 26, 2005, Arkansas Nuclear One, Unit 1 (ANO-1), experienced an automatic actuation of the Reactor Protection System due to a Main Turbine trip caused by low turbine bearing lube oil pressure. The Reactor Protection System performed as designed resulting in a reactor trip from 95 percent power. Analysis of the event has determined that the reduction in turbine bearing lube oil pressure was caused by a failure of the lube oil ejector discharge check valve, LO-79. Subsequent analysis established that a weld failure within the valve internals had caused the disc to separate from the hinge resulting in blockage of lube oil flow. Examination of the failed welds indicated that they were undersized for the applied load. To correct this condition, a replacement valve assembly, meeting original equipment manufacturer design requirements, was installed. Following the reactor trip, a spurious actuation of Emergency Feedwater (EFW) occurred which was caused by an invalid low steam generator level signal generated by the Emergency Feedwater Initiation and Control system. To reduce the likelihood of recurrence of an invalid EFW actuation, a Technical Specification amendment was prepared and implemented upon NRC approval, allowing a reduction of the EFW low steam generator level initiation setpoint and an increase in the low level actuation time delay.

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|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

A. Plant Status

At the time of this event, Arkansas Nuclear One, Unit 1 (ANO-1) was operating in Mode 1 at approximately 95 percent power.

B. Event Description

At 1047 CST, on December 26, 2005, ANO-1 experienced an automatic actuation of the Reactor Protection System (RPS)[JC] due to a Main Turbine [TA] trip in response to low turbine bearing lube oil [TD] pressure. The low lube oil pressure condition was the consequence of a failure of oil ejector discharge check valve, LO-79, when failure of a weld within the valve internals caused the valve disc to separate from the hinge. The disc lodged in the valve body and momentarily blocked adequate flow of bearing lube oil within the system, creating the low pressure condition which caused the Main Turbine to trip. Backup motor driven pumps started immediately to supply bearing lube oil, but not before a turbine trip.

The RPS performed as designed in response to the turbine trip resulting in an automatic shutdown of the reactor from approximately 95 percent power. Following the reactor trip, an invalid actuation of Emergency Feedwater (EFW)[BA] occurred in response to a spurious low "A" Once Through Steam Generator (OTSG) [AB] level signal which was generated by the Emergency Feedwater Initiation and Control System (EFIC)[JB]. Using alternate steam generator level instrumentation, operators verified that an actual low level condition was not present in "A" steam generator. Accordingly, operator action was taken in accordance with procedures to restore normal steam generator level control. The plant was promptly stabilized in Hot Standby (Mode 3) conditions.

C. Root Cause

The reactor trip was initiated by a trip of the Main Turbine which was caused by low turbine bearing lube oil pressure. Analysis, which included radiography testing and ultimately the disassembly of the turbine lube oil ejector discharge check valve, LO-79, determined that the valve disc had separated from the hinge due to a failure of the attachment weld. The disc then lodged in the valve body obstructing lube oil flow to the extent that oil pressure was reduced below the Main Turbine trip setpoint. Examination of the failed welds indicated failure from overload stresses, meaning that the welds were undersized for the applied load. The most recent maintenance on LO-79 prior to this event was performed to repair the hinge sleeve and disc wear pad, which were discovered to be cracked. This weld repair was performed in October, 2005, during refueling outage 1R19. Subsequent evaluation of the weld and design engineering processes employed during this repair activity determined that these processes did not adequately facilitate the implementation of the intended like-for-like repair of the valve.

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

C. Root Cause (continued)

Consequently, adequate welding design information for reconstruction of the hinge-to-disc configuration was not developed or applied during maintenance. These factors resulted in a failure to accomplish a like-for-like repair of LO-79, and the weld repair made during the 1R19 refueling outage was structurally inadequate for the application.

The OTSGs were replaced during the fall 2005 refueling outage. Both the original and replacement OTSGs contain an adjustable flow orifice in the feedwater inlet downcomer region to provide flow / level stability during power operation. Flow orifice settings for the new OTSGs were adjusted to achieve these conditions. However, during normal operations, the setting position of the adjustable flow orifice also impacts the EFIC low range level indicated value. Due to effects associated with the location of the EFIC low level instrument taps (one tap on each side of the orifice), indicated level from these instruments trends downward, deviating from actual OTSG level as Main Feedwater [SJ] flow rates increase during power ascension. It is important to note that these offset effects are only present with Main Feedwater in operation at moderate to high power levels and disappear immediately upon a loss of normal feedwater. Therefore, the EFIC low level instrumentation remains capable of performing its specified safety functions. Nevertheless, the flow orifice setting associated with the installation of the replacement steam generators resulted in lower than anticipated indicated levels from these instruments at normal operating conditions. Thermal hydraulic conditions in the OTSG feedwater inlet downcomer are difficult to model and are sensitive to small changes in OTSG conditions (density, fouling, etc.) as well as the orifice setting and dimensions themselves. The analysis methodology used by the manufacturer to assess the impact of these conditions in the new steam generators did not adequately account for the sensitivity of this indication to small changes. Although the low level initiate function included a time delay to prevent spurious actuation due to the back pressure wave phenomenon through the OTSG after the turbine stop valves close on a turbine trip, the risk of spurious invalid actuation of EFW was increased due to the closer proximity of indicated level from the EFIC low level instruments to the actuation setpoint.

D. Corrective Actions

Repairs to Check Valve LO-79 have been completed using original equipment manufacturer supplied parts. Actions are also in progress to strengthen the engineering and welding program processes to ensure that the appropriate level of rigor is applied when design documentation is not available.

To reduce the risk of spurious actuation of Emergency Feedwater, a reduction in the steam generator low-level actuation setpoint has been made and the time delay was increased. These changes were implemented following receipt of an NRC approved Safety Evaluation Report which permitted a reduction of the Technical Specification allowable value for low OTSG level actuation along with an extension of the actuation time delay. These changes provide for additional protection from invalid actuations.

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

D. Corrective Actions (continued)

In the interim, while the Technical Specification change was undergoing NRC review and approval, reactor power was limited to approximately 98 percent in order to provide margin between the EFIC low steam generator indicated level and the Emergency Feedwater actuation setpoint.

E. Safety Significance

The Reactor Protection System operated as designed to initiate the automatic reactor trip in response to the trip of the Main Turbine. Although there was an invalid actuation of Emergency Feedwater following the reactor trip, post-trip response was not significantly complicated by this occurrence, and the EFW system actuated and functioned as designed in response to the spurious low steam generator level signal. The post-trip plant response was normal with stable Hot Standby conditions (Mode 3) promptly established. Therefore, this event had minimal safety significance.

F. Basis for Reportability

The automatic actuation of the Reactor Protection System is reported in accordance with 10CFR50.73(a)(2)(iv)(A). A report of this event was made to the NRC Operations Center at 1435 CST on December 26, 2005, in accordance with 10CFR50.72(b)(2)(iv)(B). The invalid actuation of Emergency Feedwater is reported in accordance with 10CFR50.73(a)(2)(iv)(A).

G. Additional Information

There have been no previous similar events reported by ANO as Licensee Event Reports.

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