UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BOULEVARD
ARLINGTON, TEXAS 76011-4511

November 13, 2018

Mr. Richard L. Anderson, Site Vice President
Arkansas Nuclear One
Entergy Operations, Inc.
N-TSB-58
1448 S.R. 333
Russellville, AR 72802-0967

SUBJECT: ARKANSAS NUCLEAR ONE – NRC INTEGRATED INSPECTION REPORT 05000313/2018003 AND 05000368/2018003

Dear Mr. Anderson:

On September 30, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Arkansas Nuclear One, Units 1 and 2. On October 2, 2018, the NRC inspectors discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented five findings of very low safety significance (Green) in this report. All of these findings involved violations of NRC requirements. Additionally, NRC inspectors documented one Severity Level IV violation with no associated finding. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement; and the NRC resident inspector at the Arkansas Nuclear One.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; and the NRC resident inspector at the Arkansas Nuclear One.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA Mark Haire Acting for/

Neil O'Keefe, Chief
Project Branch D
Division of Reactor Projects

Docket Nos. 50-313 and 50-368
License Nos. DPR-51and NPF-6

Enclosure:
Inspection Report 05000313/2018003 and 05000368/2018003 w/Attachment: Documents Reviewed

U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	05000313, 05000368
License Numbers:	DPR-51, NPF-6
Report Numbers:	05000313/2018003, 05000368/2018003
Enterprise Identifier:	I-2018-003-0005
Licensee:	Entergy Operations, Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	Russellville, Arkansas
Inspection Dates:	July 1, 2018 to September 30, 2018
Inspectors:	C. Henderson, Senior Resident Inspector M. Tobin, Resident Inspector T. Sullivan, Resident Inspector S. Bussey, Senior Reactor Technology Instructor R. Deese, Senior Reactor Analyst P. Elkmann, Senior Emergency Preparedness Inspector M. Hayes, Operations Engineer S. Hedger, Emergency Preparedness Inspector N. Hernandez, Operations Engineer J. Kirkland, Senior Operations Engineer C. Osterholtz, Senior Operations Engineer G. Pick, Senior Reactor Inspector
Approved By:	Neil O'Keefe Chief, Project Branch D Division of Reactor Projects

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Arkansas Nuclear One, Units 1 and 2, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. NRC-identified and self-revealed findings, violations, and additional items, are summarized in the tables below.

List of Findings and Violations

Failure to Translate the Design Requirements into Instructions for Refueling Emergency Diesel Generators			
Cornerstone	Significance	Cross-cutting Aspect	Inspection Procedure
Mitigating Systems	Green NCV 05000313/2018003-01 and 05000368/2018003-01 Closed	None	71111.04 – Equipment Alignment
The inspectors identified a Green finding and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to translate current design into instructions for Unit 1 and Unit 2 diesel fuel oil transfer system. Specifically, the licensee failed to translate the current diesel fuel oil transfer system design into instructions to refuel Unit 1 and Unit 2 safety-related fuel bunkers, T-57 and 2T-57, if the non-safety bulk diesel fuel oil tank T-25 was unavailable following a design basis event (e.g., tornado, external flooding, or earthquake) for which it was not designed to withstand.			

Failure to Implement Welding Standard and Examination Procedure Guidance			
Cornerstone	Significance	Cross-cutting Aspect	Inspection Procedure
Initiating Events	Green NCV 05000313/2018003-02 Closed	H.2 – Human Performance, Field Presence	71111.12 – Maintenance Effectiveness
The inspectors reviewed a self-revealed Green finding and associated non-cited violation of Arkansas Nuclear One, Unit 1, Technical Specification 5.4.1.a, for the licensee's failure to properly preplan maintenance that can affect the performance of safety-related equipment. Specifically, the licensee failed to implement welding standard guidance and examination procedure guidance during the installation of the high pressure injection system drain line containing drain valves MU-1066A and MU-1066B. The drain line weld developed a crack that caused a leak shortly after plant startup that was determined to have been caused by grinding during the welding process, which was not permitted by the welding standard.			

Failure to Provide Complete and Accurate Information in a License Amendment Request to Change Emergency Action Level Requirements			
Cornerstone	Significance	Cross-cutting Aspect	Inspection Procedure
Not Applicable	Severity Level IV NCV 05000313/2018003-03 and 05000368/2018003-03 Closed	Not Applicable	71114.04 – Emergency Action Level and Emergency Plan Changes
The inspectors identified a Severity Level IV non-cited violation because the licensee provided inaccurate information to the NRC in a license amendment request for an emergency action level scheme change. Specifically, the licensee provided information about the availability of the postaccident sampling system building radiation monitor and the Unit 1 level instrumentation necessary to determine entry into an emergency action level that was not accurate.			

Failure to Verify Safety-Related 4160 V Breaker Operability Following Maintenance Activities			
Cornerstone	Significance	Cross-cutting Aspect	Inspection Procedure
Mitigating Systems	Green NCV 05000313/2018003-04 Closed	P.3 – Problem Identification and Resolution, Resolution	71152 – Problem Identification and Resolution
The inspectors reviewed a self-revealed Green finding and associated non-cited violation of Arkansas Nuclear One, Unit 1, Technical Specification 5.4.1.a, for the licensee's failure to properly preplan maintenance that can affect the performance of safety-related equipment. Specifically, the licensee failed to perform post maintenance testing to demonstrate component operability for train A safety-related 4160 V switchgear A-303 breaker that provides power to the swing service water pump B (P-4B) after the breaker was racked in. The breaker subsequently failed to close when attempting to start the pump.			

Failure to Maintain Main Feedwater Pump B Discharge Pressure in Band Caused a Reactor Trip			
Cornerstone	Significance	Cross-cutting Aspect	Inspection Procedure
Initiating Events	Green NCV 05000313/2018003-05 Closed	H.4 – Human Performance, Teamwork	71153 – Follow-up of Events and Notices of Enforcement Discretion
The inspectors reviewed a self-revealed Green finding and associated non-cited violation of Arkansas Nuclear One, Unit 1, Technical Specifications 5.4.1.a, for the licensee's failure to implement Procedure OP-1102.002, "Plant Startup," Revision 106. Specifically, control room operators failed to maintain main feedwater pump discharge pressure in the required band to control flow to the steam generators during a plant startup. As a result, the only operating main feedwater pump tripped on high discharge pressure causing an automatic reactor trip.			

Reactor Power Transient Caused by the Turbine Bypass Valve Failing Open			
Cornerstone	Significance	Cross-cutting Aspect	Inspection Procedure
Initiating Events	Green NCV 05000313/2018003-06 Closed	P.3 – Problem Identification and Resolution, Resolution	71153 – Follow-up of Events and Notices of Enforcement Discretion
The inspectors reviewed a self-revealed Green finding and associated non-cited violation of Arkansas Nuclear One, Unit 1, Technical Specifications 5.4.1.a, for the licensee's failure to properly preplan maintenance that affected the performance of safety-related equipment. Specifically, the licensee failed to properly preplan maintenance for the replacement of airline tubing for turbine bypass valve CV-6687, which resulted in the vibration-induced failure of the air tubing causing valve CV-6687 to fail open, resulting in a manual reactor trip and a subsequent loss of the main condenser.			

Additional Tracking Items

Type	Issue number	Title	Inspection Procedure	Status
LER	05000313/2018-001-00	Automatic Reactor Trip due to Loss of Main Feedwater Pump	71153	Closed
LER	05000313/2018-002-00	Leak in Class 1 Reactor Coolant System Pressure Boundary Piping due to Cyclic Fatigue Failure on a High Pressure Injection Line Drain Tap Weld	71153	Closed
LER	05000313/2018-003-00	Manual Trip due to Turbine Bypass Valve Failing Open	71153	Closed

PLANT STATUS

Unit 1 began the inspection period at full power. On July 23, 2018, power was lowered to 64 percent as requested by transmission system operator to facilitate repairs on the 500 kV Mabelvale Line. Unit 1 was returned to full power on August 8, 2018.

Unit 2 began the inspection period at full power. On September 15, 2018, power was reduced to 49 percent to address a tube leak in the B north main condenser.

On September 16, 2018, Unit 2 was shutdown to correct leakage from a feedwater system drain line. The unit remained shut down to transition to Refueling Outage 2R26 on September 29, 2018.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public Web site at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors performed plant status activities described in Inspection Manual Chapter 2515, Appendix D, "Plant Status," and conducted routine reviews using IP 71152, "Problem Identification and Resolution." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

71111.01—Adverse Weather Protection

Summer Readiness (1 Sample)

The inspectors evaluated summer readiness of offsite and onsite alternating current (ac) power systems on August 6, 2018.

71111.04—Equipment Alignment

Partial Walkdown (4 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) Unit 2 train A steam drive emergency feedwater system on July 13, 2018

(2) Unit 1 and Unit 2 alternate ac diesel generator fuel oil system temporary modification on August 1, 2018

(3) Unit 2 emergency diesel generator 2 on August 1, 2018

(4) Unit 1 emergency diesel generator 1 starting air system on August 8, 2018

71111.05Q—Fire Protection Quarterly

Quarterly Inspection (6 Samples)

The inspectors evaluated fire protection program implementation in the following selected areas:

(1) Unit 2 turbine-driven emergency feedwater (EFW) pump room during motor driven EFW pump unavailability, Fire Area CC, Fire Zone 2024-JJ, on July 24, 2018

(2) Unit 1 diesel-drive fire water pump temporary fuel oil system, Fire Area N, on August 15, 2018

(3) Unit 2 fuel oil vault for emergency diesel generator 1 fuel oil bunker Room 253 fire impairment, Fire Area L, on August 15, 2018

(4) Unit 1 high pressure injection pump B room degraded fire barrier FB-55-1, Fire Area C, Fire Zone 20-Y on September 19, 2018

(5) Unit 2 reactor building north, Fire Zone 2033K on September 25, 2018

(6) Unit 2 reactor building south, Fire Zone 2032K on September 25, 2018

71111.11—Licensed Operator Requalification Program and Licensed Operator Performance

Operator Requalification (1 Sample)

(1) The inspectors observed and evaluated Unit 1 biannual requalification exam on July 26, 2018.

Operator Performance (2 Samples)

(1) The inspectors observed and evaluated Unit 1 power reduction to 64 percent for grid conditions on July 23, 2018, and power ascension to full power on August 7, 2018.

(2) The inspectors observed and evaluated Unit 2 power reduction to 49 percent to correct a tube leak in the B north main condenser on September 15, 2018.

Operator Exams (2 Samples)

(1) The inspectors reviewed and evaluated Unit 1 requalification examination results on September 6, 2018.

(2) The inspectors reviewed and evaluated Unit 2 requalification examination results on September 6, 2018.

Operator Requalification Program (2 Samples)

(1) The inspectors evaluated the Unit 1 operator requalification program from July 9 to July 13, 2018.

(2) The inspectors evaluated the Unit 2 operator requalification program from July 9 to July 13, 2018.

71111.12—Maintenance Effectiveness

Routine Maintenance Effectiveness (2 Samples)

The inspectors evaluated the effectiveness of routine maintenance activities associated with the following equipment and/or safety significant functions:

(1) Unit 1 and Unit 2 emergency feedwater steam admission check valves on July 31, 2018

(2) Unit 1 high pressure injection; ¾-inch drain line through-wall lead between Sockolet and valve MU-1066A on September 24, 2018

71111.13—Maintenance Risk Assessments and Emergent Work Control (7 Samples)

The inspectors evaluated the risk assessments for the following planned and emergent work activities:

(1) Unit 2 turbine-driven emergency feedwater pump 2P-7A maintained available during planned maintenance of hazard barrier door on July 16, 2018

(2) Unit 1 train B emergency feedwater control vector valve CV-2648 relay replacement on July 19, 2018

(3) Unit 2 motor driven emergency feedwater pump unavailability for planned maintenance on July 24, 2018

(4) Unit 1 and Unit 2 switchyard battery testing while 500 kV breaker B5106 was open on July 31, 2018

(5) Unit 1 reactor protection system channel C main feedwater pump trip bypass bistable emergent work when reactor power greater than 9 percent on August 10, 2018

(6) Unit 2 emergency diesel generator 1 planned 2-year and 10-year preventative maintenance outage on August 15, 2018

(7) Unit 1 Yellow risk window when performing 18-month train A high pressure injection flow instruments calibration on September 12, 2018

71111.15—Operability Determinations and Functionality Assessments (8 Samples)

The inspectors evaluated the following operability determinations and functionality assessments:

(1) Unit 1 turbine-driven emergency feedwater steam generator A degraded admission check valve MS-272 on July 9, 2018

(2) Unit 1 turbine-driven emergency feedwater steam generator A admission check valve MS-271 and trip and throttle valve on July 31, 2018

(3) Unit 1 emergency diesel generator 1 degraded flywheel teeth on August 1, 2018

(4) Unit 1 bus A1 feeder breaker A-111 from startup transformer 2 with no post-maintenance test performed following breaker rack up on August 21, 2018

(5) Unit 2 emergency diesel generator 1 for identified deficiencies during maintenance window on September 10, 2018

(6) Unit 2 train B containment spray pump seal cooler shell side wall thickness below acceptance criteria on September 10, 2018

(7) Unit 1 train A high pressure injection system operability and entry into required technical specifications during 18-month injection flow instrument calibration surveillance on September 14, 2018

(8) Unit 2 turbine-driven emergency feedwater pump degraded floor drain check valve on September 20, 2018

71111.18—Plant Modifications (1 Sample)

The inspectors evaluated the following permanent modifications:

(1) Unit 1 and Unit 2 emergency feedwater steam emission check valves on July 31, 2018

71111.19—Post Maintenance Testing (6 Samples)

The inspectors evaluated the following post maintenance tests:

(1) Unit 1 emergency feedwater initiation and control Channel A vector test, steam generator B vector permissive relay 42X2-M088 replacement on July 17, 2018

(2) Unit 2 motor driven emergency feedwater pump maintenance on July 25, 2018

(3) Unit 2 motor driven emergency feedwater pump room cooler maintenance on July 25, 2018

(4) Unit 2 motor driven emergency feedwater pump room hazard barrier door maintenance on July 25, 2018

(5) Unit 2 emergency diesel generator 1 potential current transformer 3 replacement on August 22, 2018

(6) Unit 1 and Unit 2 alternate ac diesel generator temporary fuel oil system restoration following bulk fuel oil tank T-25 maintenance on September 25, 2018

71111.20—Refueling and Other Outage Activities (1 Sample and a partial sample)

(1) The inspectors evaluated Unit 2 Forced Outage 2018-001 activities from September 16 to September 29, 2018.

(2) The inspectors evaluated Unit 2 Refueling Outage 2R26 activities from September 29 to September 30, 2018. The inspectors completed inspection procedure Sections 03.01.a and 03.01.c.

71111.22—Surveillance Testing

The inspectors evaluated the following surveillance tests:

Routine (3 Samples)

(1) Unit 1 emergency diesel generator 2 24-hour surveillance run on July 16, 2018

(2) Unit 2 A excore instrument monthly surveillance with D excore instrument in trip on September 13, 2018

(3) Unit 1 elevated reactor coolant system unidentified and identified leakage rate on September 17, 2018

In-service (1 Sample)

(1) Unit 2 motor driven emergency feedwater pump on August 23, 2018

71114.01—Exercise Evaluation (1 Sample)

The inspectors evaluated the biennial emergency plan exercise, conducted July 17, 2018. The exercise scenario simulated a loss of offsite power with an emergency diesel generator out of service and a diesel generator trip, resulting in a loss of all ac power onsite, a reactor coolant system leak, isolation valve failures, and a pipe break in a containment penetration between the inboard and outboard isolation valves. The inspectors discussed exercise performance with staff at Federal Emergency Management Agency (FEMA) Region VI.

71114.04—Emergency Action Level and Emergency Plan Changes (1 Sample)

The licensee submitted a summary of Emergency Action Level classification procedure changes (Revision 56) to the NRC on June 28, 2018. The inspectors conducted both in-office and onsite review of the changes from July 10, 2018, to September 13, 2018. This evaluation does not constitute NRC approval.

71114.06—Drill Evaluation

Drill/Training Evolution (2 Samples)

(1) The inspectors observed and evaluated Unit 1 control room simulator training for internal flooding in the turbine building with an overheating event on August 2, 2018.

(2) The inspectors observed and evaluated Unit 1 control room simulator training for fire in emergency diesel generator 1 room and steam generator tube rupture event on September 5, 2018.

71114.08 - Exercise Evaluation – Scenario Review (1 Sample)

The inspectors reviewed and evaluated the proposed scenario for the July 17, 2018, biennial emergency plan exercise on June 21, 2018. The inspectors discussed the proposed exercise scenario with staff at FEMA Region VI.

OTHER ACTIVITIES – BASELINE

71151—Performance Indicator Verification (9 Samples)

The inspectors verified licensee performance indicators submittals listed below:

(1) MS06: Unit 1 and Unit 2 Emergency AC Power Systems (07/01/2017 – 06/30/2018)

(2) MS07: Unit 1 and Unit 2 High Pressure Injection Systems (07/01/2017 – 06/30/2018)

(3) MS08: Unit 1 and Unit 2 Heat Removal Systems (07/01/2017 – 06/30/2018)

(4) EP01: Drill/Exercise Performance Sample (04/01/2017-06/30/2018)

(5) EP02: Emergency Response Organization Drill Participation Sample (04/01/2017-06/30/2018)

(6) EP03: Alert And Notification System Reliability Sample (04/01/2017-06/30/2018)

<u>71152—Problem Identification and Resolution</u>

<u>Annual Follow-up of Selected Issues</u> (2 Samples)

The inspectors reviewed the licensee's implementation of its corrective action program related to the following issues:

(1) Unit 1 axial power shaping rod 8-1 not coupled on July 26, 2018

(2) Unit 1 train A service water pump B breaker A-303 failure to close on July 31, 2018

<u>71153—Follow-up of Events and Notices of Enforcement Discretion</u>

<u>Licensee Event Reports</u> (3 Samples)

The inspectors evaluated the following licensee event reports which can be accessed at https://lersearch.inl.gov/LERSearchCriteria.aspx:

(1) Unit 1 Licensee Event Report 05000313/2018-001-00, Automatic Reactor Trip Due to Loss of Main Feedwater Pump, on July 27, 2018

(2) Unit 1 Licensee Event Report 05000313/2018-002-00, Leak in Class 1 Reactor Coolant System Pressure Boundary Piping Due to Cyclic Fatigue Failure on a High Pressure Injection Line Drain Tap Weld, on September 17, 2018

(3) Unit 1 Licensee Event Report 05000313/2018-003-00, Manual Trip Due to Turbine Bypass Valve Failing Open, on September 17, 2018

INSPECTION RESULTS

Failure to Translate the Design Requirements into Instructions for Refueling Emergency Diesel Generators			
Cornerstone	Significance	Cross-cutting Aspect	Inspection Procedure
Mitigating Systems	Green NCV 05000313/2018003-01 and 05000368/2018003-01 Closed	None	71111.04 – Equipment Alignment

The inspectors identified a Green finding and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to translate current design into instructions for Unit 1 and Unit 2 diesel fuel oil transfer system. Specifically, the licensee failed to translate the current diesel fuel oil transfer system design into instructions to refuel Unit 1 and Unit 2 safety-related fuel bunkers, T-57 and 2T-57, if the non-safety bulk diesel fuel oil tank T-25 was unavailable following a design basis event (e.g., tornado, external flooding, or earthquake) for which it was not designed to withstand.

Description: The inspectors reviewed Unit 1 and Unit 2 Procedure OP-1104.23, "Diesel Oil Transfer Procedure," Revision 37, and the current licensing and design basis for the diesel fuel oil transfer system prior to the licensee installing a temporary diesel fuel oil system to support maintenance on non-safety bulk diesel fuel oil tank T-25. From this review, the inspectors identified the following issue:

The Unit 1 and Unit 2 safety analysis reports state that emergency diesel generators are required to be able to operate during and following severe natural phenomenon. It further describes how long the onsite supply of fuel in protected safety-related tanks will last before refueling is required. Design requirements and commitments stipulate that the licensee be able to refuel the safety-related tanks within that time period so the emergency diesel generators continue to run uninterrupted.

The normal method for refueling the safety-related tanks is to supply fuel from the onsite non-safety bulk storage tank. The inspectors noted that the bulk storage tank is not designed to withstand a safe shutdown earthquake, the maximum flood conditions, or the design basis tornado. The Unit 1 and Unit 2 final safety analysis reports both state that, "…additional fuel could be delivered to the plant site by any one of three methods: truck delivery, rail car delivery or delivery by barge from the river." In the highly unlikely event that all three of these normal supply routes are unavailable because of the earthquake, fuel could be airlifted to the plant site via helicopter. However, the inspectors were unable to locate procedures, instructions, or drawings that provided a method of refueling the protected safety-related tanks in the event that a natural disaster disables the bulk storage tank. When questioned, the licensee concluded that there were no such instructions, procedures, or drawings. The licensee entered this deficiency into the corrective action program as Condition Reports CR-ANO-C-2018-03210 and CR-ANO-C-2018-03735.

From the above information, the inspectors determined the licensee failed to translate into instructions the design requirement to refuel T-57 and 2T-57 if T-25 was unavailable following a design basis event.

Corrective Actions: The immediate actions were to develop a standing order to route 150 feet of hose from the roof of the diesel fuel oil bunker building to the manways on top of T-57 and 2T-57 to refuel T-57 and 2T-57 when T-25 is unavailable, and to initiate Procedure

Improvement Forms 1-18-0482 and 2-18-0326 to update Unit 1 and Unit 2 Procedures OP-1203.025, “Natural Emergency,” and OP-2203.008, “Natural Emergency,” to incorporate the standing order guidance. Additionally, the licensee initiated actions to determine a long-term corrective action for this issue. Corrective Action References: Condition Reports CR-ANO-C-2018-03210 and CR-ANO-C-2018-03735
<u>Performance Assessment</u>: Performance Deficiency: The licensee’s failure to translate diesel fuel oil transfer system design requirements to provide a method for refueling the emergency diesel generators if the non-safety bulk storage tank was not available into instructions, procedures, or drawings is a performance deficiency. Screening: The performance deficiency was more than minor, and therefore a finding, because it was associated with the procedural quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. Specifically, the licensee’s failure to translate the current diesel fuel oil transfer system design into instructions to refuel T-57 and 2T-57 would have complicated operator response during a design basis event to maintain continuous operation of the emergency diesels. Significance: Using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because it: (1) was not a design deficiency; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and (4) did not result in the loss of a high safety-significant, nontechnical specification train. Cross-cutting Aspect: A cross-cutting aspect was not assigned to this finding because the performance deficiency occurred during initial construction and, therefore, is not indicative of current licensee performance.
<u>Enforcement</u>: Violation: As required by 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in 10 CFR 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies, are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, from initial construction to September 2018, the licensee failed to establish measures to assure that applicable regulatory requirements and the design basis, as defined in 10 CFR 50.2, and as specified in the license application for the Unit 1 and Unit 2 diesel fuel oil transfer system, were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to translate the diesel fuel oil transfer system design requirements to provide a method for refueling the emergency diesel generators if the non-safety bulk storage tank was not available. Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was very low safety significance (Green)

and was entered into the licensee's corrective action program as Condition Reports CR-ANO-C-2018-03210 and CR-ANO-C-2018-03735.

Failure to Implement Welding Standard Guidance and Examination Procedures			
Cornerstone	Significance	Cross-cutting Aspect	Inspection Procedure
Initiating Events	Green NCV 05000313/2018003-02 Closed	H.2 – Human Performance, Field Presence	71111.12 – Maintenance Effectiveness
The inspectors reviewed a self-revealed Green finding and associated non-cited violation of Arkansas Nuclear One, Unit 1, Technical Specification 5.4.1.a, for the licensee's failure to properly preplan maintenance that can affect the performance of safety-related equipment. Specifically, the licensee failed to implement welding standard guidance and examination procedure guidance during the installation of the high pressure injection system drain line containing drain valves MU-1066A and MU-1066B. The drain line weld developed a crack that caused a leak shortly after plant startup that was determined to have been caused by grinding during the welding process, which was not permitted by the welding standard.			

<u>Description</u>: During Arkansas Nuclear One, Unit 1, Refueling Outage 1R27, the licensee replaced the ¾ inch drain line containing drain valves MU-1066A and MU-1066B located on the 2.5 inch C high pressure injection header. On June 5, 2018, with the plant operating, the licensee discovered the reactor building sump fill rate was rising. Upon further investigation and direct inspection, the licensee identified the source of the elevated reactor building sump fill rate as coming from a recently welded connection at the drain piping between the drain valves MU-1066A/B and the C high pressure injection header. This section of piping degraded the reactor coolant system American Society of Mechanical Engineers (ASME) Class 1 boundary causing an unplanned reactor plant shutdown. Repairs included removing the drain valve line and the installation of a welded piping plug. The failed piping/weld component was cut out of the high pressure injection system and transferred to a vendor for metallurgical evaluation. The vendor determined the direct cause of the event was a cyclic fatigue failure originating from grinding marks on the axial weld toe of the weld joint between the pipe stub and the Sockolet. The licensee entered this issue into their corrective action program as Condition Report CR-ANO-1-2018-03567 and performed an adverse condition analysis to investigate this event.

The apparent cause analysis identified four causal factors (CF) for the event:

- (CF-1): Less than adequate procedure use and adherence for welding. The welders did not perform the welds in accordance with the Entergy Nuclear Fleet general weld standard CEP-WP-GWS-1, "General Welding Standard ASME/ANSI," Revision 3, or the instructions provided in the welder administrative training provided by the site welding program engineer. Additionally, as part of the welding documentation in Work Order 462301, the socket weld enhancement instruction sheet provides the welder technique necessary to perform a quality 2T enhanced socket weld. These instructions provided the guidance to ensure the weld is left in the as-welded condition. Specifically, weld grinding or polishing on the toe of the weld should not be performed.

- (CF-2): Less than adequate procedure use and adherence for weld inspections. The visual examination of the completed weld was not completed in accordance with the

Entergy Nuclear Fleet visual examination procedure for ASME piping weld joints. Procedure CEP-NDE-0965, "Visual Welding Inspection ASME, ANSI B31.1," Revision 5, provided the acceptance criteria for weld geometry and the presence of grinding marks on the axial toe of the weld, which should have caused the rejection of the weld. This procedure was a reference use procedure; however, supplemental employees are required to be aware of the specific qualification procedure as part of Procedure EN-OM-126, "Management and Oversight of Supplemental Personnel," Revision 6, briefing. This resulted in the examiners not following the visual examination procedure for 2T enhanced socket welds.

- (CF-3): Less than adequate field oversight. A tie between CF-1 and CF-2 was identified to exist in that, in both cases, supplemental employees performed the weld and weld examinations with no direct licensee personnel present to ensure the welding and examination instructions were implemented correctly.

  (CF-4): System induced vibrations. The final causal factor was the system vibrations which caused the welding defect to propagate through the wall in a relatively short period of time. While the initiation site for the weld crack was found to be at a weld grinding mark, the system vibrations contributed to the through wall growth of the crack.

Corrective Action: The licensee installed a welded plug at the MU-1066A/B high pressure injection drain line. They also performed an extent of condition review and examined all of the welds made by contractors on jobs associated with this outage, and identified no further issues.

Corrective Action Reference: Condition Report CR-ANO-1-2018-03729

Performance Assessment:

Performance Deficiency: The licensee's failure to implement welding standard and examination procedure guidance is a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and affected the associated cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to implement welding standard and examination procedure guidance resulted in a leak in the high pressure injection system valve drain line MU-1066A/B, which degraded the reactor coolant system ASME Class 1 boundary causing an unplanned reactor plant shutdown.

Significance: Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because the finding: (1) did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident and (2) did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function.

Cross-cutting Aspect: This finding had a cross-cutting aspect in the area of human performance associated with field presence because leaders failed to be commonly seen in the work areas of the plant observing, coaching, and reinforcing standards and expectation

and promptly correcting deviations from standards and expectations. Senior managers did not ensure supervisory and management oversight of work activities by contractors and supplement personnel. Specifically, the licensee failed to provide adequate supervisory and management oversight of contractors performing welding and inspection activities for replacing the ¾ inch high pressure injection drain line containing drain valves MU-1066A and MU-1066B.
<u>Enforcement</u>: Violation: Technical Specification 5.4.1.a for Unit 1 requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures listed in Appendix A to Regulatory Guide 1.33, "Quality Assurance Program Requirements," Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 9.a, states, in part, that maintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures. The licensee established Procedure CEP-WP-GWS-1, Revision 3, to perform welding maintenance activities on safety-related piping, and Procedure CEP-NDE-0965 to perform visual inspections of safety-related pipe welds. Procedure CEP-WP-GWS-1, Attachment 5.8, step 4, states, in part, "unless otherwise directed by the senior welding engineer, the completed weld should be maintained in an "As-welded" condition." Procedure CEP-NDE-0965, step 5.6, states, in part, visual examination (VT) criteria for welds are contained in Attachment 9.2. Attachment 9.2, states, in part, "the completed weld should be maintained in an "as-welded" condition. Excessive or inadvertent grinding marks and/or notches at or near the toes of the weld should be avoided." Contrary to the above, on April 12, 2018, the licensee failed to implement Procedure CEP-WP-GWS-1, Attachment 5.8, step 4, and Procedure CEP-NDE-0965, Attachment 9.2, which affected the performance of safety-related equipment. Specifically, the licensee failed to preplan welding activities and provide adequate oversight of contractors in a manner that maintained the axial toe weld for valve drain line MU-1066A/B in the as-welded condition. This resulted in a leak in the high pressure injection system valve drain line MU-1066A/B degrading the reactor coolant system ASME Class 1 boundary and causing an unplanned reactor plant shutdown to effect repairs. Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-ANO-1-2018-03567.

Failure to Provide Complete and Accurate Information in a License Amendment Request to Change Emergency Action Level Requirements			
Cornerstone	Severity	Cross-cutting Aspect	Inspection Procedure
Not Applicable	Severity Level IV NCV 05000313/2018003-03 and 05000368/2018003-03 Closed	Not Applicable	71114.04 – Emergency Action Level and Emergency Plan Changes
The inspectors identified a Severity Level IV non-cited violation because the licensee provided inaccurate information to the NRC in a license amendment request for an emergency action level scheme change. Specifically, the licensee provided information about			

the availability of the postaccident sampling system building radiation monitor and the Unit 1 level instrumentation that was material to the licensing decision, but not accurate.

Description: The NRC approved an emergency action level scheme change on November 9, 2012 (ADAMS Accession No. ML12269A455) to allow Arkansas Nuclear One to adopt the Nuclear Energy Institute (NEI) 99-01, Revision 5, scheme. Subsequently, the licensee identified that two of their current emergency action level thresholds could not be implemented in accordance with their emergency classification procedure:

- On May 26, 2017, Condition Report CR-ANO-2-2017-03161 documented that postaccident sampling system building radiation monitor 2RX-9840 should be removed from all regulatory commitments because the postaccident sampling system had been removed from service, and its building would not be monitored for radiological releases. Radiation monitor 2RX-9840 was being used as a means to evaluate emergency action levels AU1, AA1, AS1, and AG1. In addition, it was used in the loss/potential loss of containment (CNB6) for fission product emergency action levels. The condition report noted that requirements for the postaccident sampling system had been removed from Arkansas Nuclear One licenses in August 2000 and the licensee had abandoned the system's valves (March 2003, EC-ANO-1779), removed power from the postaccident sampling system ventilation system (January 2004), and made radiation monitor 2RX-9840 nonfunctional (May 2008, Condition Report CR-ANO-2-2008-01439 and Work Order 150817).

- On March 15, 2018, Condition Report CR-ANO-C-2018-01121 documented that the Unit 1 level instrumentation set point used in emergency action level CA1 was below the indicating range of the instrument. The emergency action level indicated that a loss of Unit 1's reactor vessel inventory was shown by an indicated level less than 368 feet, 0 inches. Therefore, the lowest level indicated on the instrument would be higher than the level used in making the emergency classification decision.

The inspectors reviewed the licensee's license amendment request, dated December 1, 2011 (ADAMS Accession No. ML113350317), "Proposed Emergency Action Levels Using NEI 99-01, Revision 5, Scheme," and the licensee's response to a request for additional information dated July 9, 2012, (ADAMS Accession No. ML12192A090) to determine whether the conditions identified in the corrective action program existed at the time the licensee requested the license amendment and whether the request correctly described the instruments. The inspectors identified:

- The December 1, 2011, submittal incorrectly indicated that radiation monitor 2RX-9840 was a viable means of classifying emergency action levels AU1, AA1, AS1, and AG1, as well as providing input for the evaluation of fission product barrier emergency action levels. In the response to NRC's request for additional information (RAI) dated July 9, 2012, the licensee provided additional details about the super particulate iodine noble gas (SPING) radiation monitors used in this application. Response to Question 3 associated with emergency action levels AA1, AS1, and AG1 stated: "Each SPING is associated with a particular ventilation pathway and provides continuous monitoring of air discharged via the respective release pathway." The license reviewer concluded that all of the SPING monitors included in the license amendment request were operable and continuously monitoring the specified release pathways, thereby being capable of measuring the radiation levels described in the proposed emergency action levels.

- The December 1, 2011, submittal indicated that loss of Unit 1 reactor vessel inventory for emergency action level CA1 was a vessel level less than 368 feet, 0 inches.

This issue was NRC-identified because when the licensee identified the emergency action level errors, they took action to correct the errors, but failed to address the failure to ensure that technical information provided to the NRC in support of the license amendment request was complete and accurate in all material respects.

Corrective Actions: To correct the Unit 1 reactor vessel level emergency action level threshold error, the licensee issued communications regarding correct application of the emergency action level on March 15, 2018, followed by implementation of a change to Procedure OP-1903.010, "Emergency Action Level Classification," Revision 56, dated June 26, 2018, with the corrected level. The use of radiation monitor 2RX-9840 is being removed from the emergency action levels as part of an emergency action level scheme change submitted to the NRC on March 29, 2018 (ADAMS Accession No. ML18088B412 and ML18094A155). In the interim, the licensee issued communications to emergency director-qualified staff members to ensure they are aware of the error, how to address it if implementing emergency action levels, and to inform them of the corrective actions in progress. Additionally, the licensee issued Condition Report CR-ANO-C-2018-03597, dated September 13, 2018, for the incomplete and inaccurate emergency action level submission examples to address the completeness and accuracy issues identified by the inspectors.

Corrective Action References: Condition Reports CR-ANO-2-2017-03161, CR-ANO-C-2018-01121, and CR-ANO-C-2018-03597

Performance Assessment: The inspectors determined this violation was associated with a reactor oversight program performance deficiency of minor significance. Specifically, in both examples, it was determined that the licensee failed to maintain the effectiveness of the emergency plan; however, they were minor performance deficiencies due to the continued effectiveness of the emergency action levels despite the errors.

Enforcement: The ROP's significance determination process does not specifically consider the regulatory process impact in its assessment of licensee performance. Therefore, it was necessary to address this violation using traditional enforcement. This issue was determined to be a Severity Level IV violation using the NRC Enforcement Policy, dated May 15, 2018, Section 2.3.11, "Inaccurate and Incomplete Information," and Section 6.9, "Inaccurate and Incomplete Information or Failure to Make a Required Report."

Violation: Section 50.9(a) of 10 CFR states, in part, that information provided to the Commission by a licensee shall be complete and accurate in all material respects.

Contrary to the above, on December 1, 2011, and July 9, 2012, information was provided to the Commission by the licensee that was not complete and accurate in all material respects. Specifically, the licensee's emergency action level scheme change submittal documents contained emergency action level declaration threshold values (i.e., setpoints) that could not be indicated by the specified plant equipment and/or referenced instrumentation that was no longer in service. The information was material to the NRC's decision whether to approve a license amendment request. The NRC approved a license amendment for an emergency action level scheme on November 9, 2012, which included emergency action levels which could not be implemented; the approval of those emergency action levels was material to the licensing action because it was based on the incorrect information submitted by the licensee.

Severity: This issue was determined to be more than minor because by providing inaccurate information in support of a license amendment request, the licensee impeded the regulatory process of reviewing and approving the license amendment request. The Enforcement Policy, Section 6.9(c)(1), provides that a violation is characterized as Severity Level III if the accurate information would have caused the NRC to reconsider a regulatory position or undertake further inquiry. There are no corresponding Severity Level IV examples. Through discussion with the Office of Nuclear Security and Incident Response (NSIR), it was determined that had accurate information been provided (or had the NRC known the information was inaccurate), the NRC license reviewer would have used the request for additional information process to address these problems with the license amendment request. Specifically, the licensee would have been required to revise their proposed emergency action levels so they could be implemented before the emergency action scheme change was approved. Because the request for additional information is a routine NRC process, it was concluded that the failure to provide accurate information to the NRC would not have caused the NRC to undertake substantial further inquiry (a threshold for Severity Level III), and therefore the violation was appropriately characterized as Severity Level IV. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Verify Safety-Related 4160 V Breaker Operability Following Maintenance Activities			
Cornerstone	Significance	Cross-cutting Aspect	Inspection Procedure
Mitigating Systems	Green NCV 05000313/2018003-04 Closed	P.3 – Problem Identification and Resolution, Resolution	71152 – Problem Identification and Resolution
The inspectors reviewed a self-revealed Green finding and associated non-cited violation of Arkansas Nuclear One, Unit 1, Technical Specification 5.4.1.a, for the licensee's failure to properly preplan maintenance that can affect the performance of safety-related equipment. Specifically, the licensee failed to perform post-maintenance testing to demonstrate component operability for the train A safety-related 4160 V switchgear A-303 breaker that provides power to the swing service water pump B (P-4B) after the breaker was racked in. The breaker subsequently failed to close when attempting to start the pump.			
<u>Description</u>: On May 6, 2018, train A safety-related 4160 V switchgear A-303 breaker, which provides power to the swing service water pump P-4B, was racked out to support maintenance activities for hand switch replacement during Refueling Outage 1R27. After the maintenance was completed, the licensee racked in the breaker and did not perform a post-maintenance test in accordance with Procedure COPD-001, "Operations Expectations and Standards," Revision 75. Specifically, Procedure COPD-001 states, in part, "When a 480 V load center, 4160 V or 6900 V breaker has been restored from being racked out/down, operate the associated component to confirm component operations." Additionally, the licensee personnel did not obtain permission from the Senior Manager, Operations, Operations Manager, Support or Operations Manager, Shift to waive the post-maintenance testing requirement in accordance with Procedure COPD-001. On June 22, 2018, the licensee attempted to start service water pump P-4B by closing breaker A-303 for a quarterly pump surveillance. When the hand switch for the P-4B was taken to start, the associated feeder breaker A-303 failed to close. The licensee's			

troubleshooting verified the A-303 breaker operated appropriately outside the cubicle, but failed to operate (close) when in the racked in position. The licensee replaced the breaker with a spare breaker and declared it operable following successful post-maintenance testing. The licensee determined the most probable cause of the previously installed A-303 breaker failure was improper mechanical alignment of the breaker within the cubicle when it was racked-in following the hand switch work. The licensee entered this issue into their corrective action program as Condition Reports CR-ANO-1-2018-03729 and CR-ANO-1-2018-03754.

The inspectors noted a previous similar occurrence, documented in Condition Report CR-ANO-1-2017-01764, where post maintenance testing was not performed after racking in safety-related 4160 V breakers. Specifically, on May 11, 2017, the licensee did not perform a post maintenance test after racking in the train A high pressure injection pump A P-36A breaker. Procedure COPD-001, Revision 74, provided the shift manager with the latitude to waive the requirement to operate P-36A subsequent to the breaker rack in evolution. The corrective action from that issue was to reinforce the standards to perform post maintenance testing in accordance with COPD-001, and to revise the procedure to require performing a post maintenance test unless senior operations managers agreed to waive the requirement to operate equipment after a breaker has been racked in.

The inspectors noted that, similar to the earlier problem, operators appeared to have focused on the planned work that caused the breaker to be racked out, rather than considering the breaker removal to be work that impacted the operability of the associated pump. In each case, the work was performed during a period when the pump was not required to be operable by technical specifications, and Arkansas Nuclear One did not require that operators track the impact to operability (commonly called a “tracking LCO”) when it was rendered non-functional so that operators would follow the formal process to consider actions needed to declare the pump functional or operable when it was being restored to service. Licensee management had identified the lack of a tracking LCO process during the NRC exit meeting for the 2017 violation, but had failed to take action to address it.

The inspectors determined that this failure did not result in a technical specification violation because P-36B is a swing pump; during the period when it could not be started from the train A bus, the train A pump P-36A was in service and credited for technical specification compliance.

Corrective Actions: The licensee performed an extent of condition evaluation to identify any additionally cases of post maintenance testing nonperformance following the racking in evolution of 4160 V safety-related breakers to verify operability during Refueling Outage 1R27. Additionally, the licensee sent the nonfunctional A-303 breaker to the vendor for further analysis.

Corrective Action Reference: Condition Reports CR-ANO-1-2018-03729 and CR-ANO-1-2018-03754

<u>Performance Assessment</u>:

Performance Deficiency: The licensee’s failure to perform post maintenance testing to demonstrate component operability is a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the train A safety-related 4160 V breaker A-303 was nonfunctional following hand switch maintenance activities because it was not properly racked into the cubicle.

Significance: Using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, the inspectors determined that the finding had very low safety significance (Green) because the finding: (1) was not a design deficiency; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and (4) did not result in the loss of a high safety-significant, nontechnical specification train. Specifically, the pump was credited for technical specification compliance only during the period of the failed test and subsequent breaker replacement.

Cross-cutting Aspect: The finding had a cross-cutting aspect in the area of problem identification and resolution associated with resolution because the licensee failed to take effective corrective actions to address issues in a timely manner commensurate with their safety significance. Specifically, the licensee’s corrective actions for the previous event of not performing post-maintenance testing following the racking in of a safety-related 4160 V breaker in 2017 did not resolve the performance problem.

<u>Enforcement</u>:

Violation: Technical Specification 5.4.1.a for Unit 1 requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures listed in Appendix A to Regulatory Guide 1.33, “Quality Assurance Program Requirements,” Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 9.a, states, in part, that maintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures. The licensee established Procedure COPD-001, “Operations Expectations and Standards,” Revision 75, to perform post-maintenance testing of safely-related 4160 V breakers after maintenance that requires racking a breaker out. Procedure COPD-001, step 5.13.1.C, states, in part, “When a 480 V load center, 4160 V, or 6900 V breaker has been restored from being racked out/down, operate the associated component to confirm component operations.”

Contrary to the above, on May 7, 2018, the licensee failed to implement Procedure COPD-001, step 5.13.1.C. Specifically, operators failed to operate the train A safety-related 4160 V A-303 breaker after it was restored from being racked out by operating the swing service water pump P-4B.

Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was very low safety significance (Green) and was entered into the licensee’s corrective action program as Condition Reports CR-ANO-1-2018-03729 and CR-ANO-1-2018-03725.

<table>
<tr><td colspan="4">Failure to Maintain Main Feedwater Pump B Discharge Pressure in Band Caused a Reactor Trip</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-cutting Aspect</td><td>Inspection Procedure</td></tr>
<tr><td>Initiating Events</td><td>Green
NCV 05000313/2018003-05
Closed</td><td>H.4 – Human Performance, Teamwork</td><td>71153 – Follow-up of Events and Notices of Enforcement Discretion</td></tr>
<tr><td colspan="4">The inspectors reviewed a self-revealed, Green finding and associated non-cited violation of Arkansas Nuclear One, Unit 1, Technical Specifications 5.4.1.a, for the licensee’s failure to implement Procedure OP-1102.002, “Plant Startup,” Revision 106. Specifically, control room operators failed to maintain main feedwater pump discharge pressure in the required band to control flow to the steam generators during a plant startup. As a result, the only operating main feedwater pump tripped on high discharge pressure, causing an automatic reactor trip.</td></tr>
<tr><td colspan="4"><u>Description</u>: On May 16, 2018, Arkansas Nuclear One, Unit 1, experienced an automatic reactor trip from approximately 10 percent reactor power. The reactor trip was caused by the reactor protection system trip due to tripping the only operating main feedwater (MFW) pump.

Operators were in the process of raising reactor power to a band of 12 to 15 percent power, when main feedwater pump B (MFW-B) tripped on high discharge pressure. At the time of the trip, MFW-B hand/automatic (H/A) station was in hand, both main feedwater block valves were closed, and the startup feedwater control valve was maintaining level in steam generator A and B. Procedure OP-1102.002 required operations personnel in this condition to adjust MFW-B speed to maintain discharge pressure between 1025 to 1075 psig. The procedure also provided guidance on the appropriate method for operators to monitor the MFW pump discharge pressures. Specifically, computer points P2833 and P2835 provided sufficient upper range for proper MFW-B discharge pressure monitoring. These points were displayed on a monitor in the control room for the ease of monitoring as a single point, instead of as a trend. The at-the-controls operator received the correct pressure band from the control room supervisor, wrote the correct values on a placard, attached it to the control board near the pump controller, and received a peer check to be used to adjust the controller. Computer points P2833 and P2835 were identified by the at-the-controls operator as being out of band (P2833 and P2835 indicated 1139 psig and 1116 psig, respectively), and during MFW-B speed adjustments, these points did not respond. At this time, the at-the-controls operator decided without any discussion with the control room supervisor, nor was it communicated to the other control room operators, to use the discharge pressure indication on the MFW-B operator interface touchscreen that was indicating in the prescribed band. Procedure OP-1102.002 did not provide any procedural guidance as to if it was appropriate for operators to use the operator interface touchscreen indications to monitor MFW pump discharge pressure, nor did the operators verify that the operator interface touchscreen indications provided appropriate MFW-B discharge pressure.

Using operator interface touchscreen indications the at-the-controls operator continued to adjust MFW-B speed, until actual MFW-B discharge pressure reached the high-pressure setback setpoint. The MFW high-pressure setback reduced MFW-B speed and then released once discharge pressure was less than the setpoint for 10 seconds. This happened three times; however, the at-the-controls operator continued to raise the MFW-B demand because it was not recognized that the controller was reducing MFW-B speed because the</td></tr>
</table>

discharge pressure was too high. The final time the discharge pressure setback was released, the operator increased MFW-B discharge pressure sufficiently to reach the high discharge pressure trip setpoint, causing it to trip. This resulted in reactor protection system actuation and emergency feedwater actuation because there were no longer any MFW pumps running. All control rods inserted into the core and the reactor was verified shutdown. The licensee entered this issue into the corrective action program as Condition Report CR-ANO-1-2018-03238 and performed a root cause evaluation to investigate this event.

The root cause evaluation identified the root cause was that Procedure OP-1102.002 did not identify MFW pump discharge pressure as a critical parameter in accordance with Procedure EN-OP-115, “Conduct of Operations,” Revision 25. The contributing cause was operations management and crew leaders did not effectively meet their responsibilities to provide optimal crew composition, maintain command and control, and oversee control room evolutions. The root cause evaluation identified the key factors of the root and contributing causes:

- Crew Composition:

  (1) The scheduled duty control room supervisor had called in sick prior to the watch and a relief control room supervisor assumed the watch. However, the relief control room supervisor was designated as the team lead for placing main feedwater pump A (MFW-A) in service later in the shift. Therefore, another control room supervisor needed to be identified to support the watch and the scheduled evolutions during the shift. With limited relief capabilities available at the time, a shift manager who was supporting the Outage Control Center became the best available candidate to relieve the control room supervisor. The shift manager assumed the control room supervisor position, but had not received the just-in-time training for the startup and had not served in the control room supervisor’s role in a year. Additionally, assignment of the control room supervisor that had not received just-in-time training did not have the concurrence of the senior operations manager and training manager as required by Procedure EN-TQ-114, “Licensed Operator Requalification Training Program,” Revision 11.

  (2) The shift technical assistant on watch during the event also had not received just-in-time training.

  (3) The control room team did not designate a dedicated reactivity senior reactor operator as required by Procedure EN-OP-115 and COPD-30, “ANO Reactivity Management Program,” Revision 9.

  (4) Overall crew composition was not reviewed in advance for individual performance and weaknesses.

- Command and Control:

  (1) The control room supervisor did not challenge the basis of why the procedurally identified feedwater pump discharge pressure monitoring points were out of band nor why the at-the-controls operator’s alternate monitoring method of using the operator interface touchscreen discharge pressure while manually operating in

hand at the integrated control system MFW-B H/A station was appropriate, nor did he request any updates from the at-the-controls operator during the evolution regarding where the discharge pressure was in relation to the monitoring band.

(2) The at-the-controls operator extrapolated the differences between the two indications and assumed that as long as the P2833 and P2835 indications remained constant, then MFW-B discharge pressure was being controlled within the band per the earlier identified operator interface touchscreen indication.

(3) The at-the-controls operator did not communicate with the control room supervisor or anyone on the crew that they would be monitoring the operator interface touchscreen indication of MFW-B discharge pressure to maintain it in the acceptable band. Therefore, crew members were unaware that the computer points P2833 and P2835 failed to appropriately monitor MFW-B discharge pressure, which impacted the team's ability to challenge MFW pump discharge pressure monitoring and control.

(4) The control board operator turbine performed a component verification versus a peer check as required by Arkansas Nuclear One operations standards.

Corrective Actions: The licensee's interim actions included: (1) implementing a standing order for reactivity control oversight during Level 1 reactivity manipulations following the guidance of Procedure EN-OP-115, and not the site specific COPD-030, due to conflicting requirements; (2) performing hourly update brief/discussion during Level 1 reactivity manipulations; (3) control room personnel participated in a new startup just-in-time training that covered this event, knowledge objectives for lower power feedwater control, and the behavior gaps evident in the event. The just-in-time training also covered other operator risk activities not previously covered in just-in-time training that were determined to be risk significant by the operations management team; (4) creating procedure use and adherence affirmation sheets to be signed by operations personnel; (5) creating roles and responsibilities sheets specific for reactivity senior reactor operator, control room supervisor, shift manager, at-the-controls operator, and management oversight roles to be signed by all licensed operators who stand those positions for the startup; (6) briefing all oversight personnel (senior management) from the general manager of plant operations or senior operations manager prior to being placed in the oversight role to ensure alignment on behaviors to observe. The corrective action to prevent reoccurrence was to revise Procedures OP-1102.002 and OP-1106.016, "Condensate, Feedwater, and Steam System Operation," Revision 76, to designate MFW pump discharge pressure band as a critical parameter in accordance with Procedure EN-OP-115 when MFW pumps are operated in manual.

Corrective Action Reference: Condition Report CR-ANO-1-2018-03238

<u>Performance Assessment</u>:

Performance Deficiency: The licensee's failure to maintain MFW pump discharge pressure in the required band is a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor, and therefore a finding, because it was associated with the human performance attribute of the Initiating Events Cornerstone and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Specifically, operators repeatedly raised the speed of the only operating MFW pump until it tripped on high discharge pressure, causing an automatic reactor trip.

Significance: The inspectors assessed the significance of the finding using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, and determined that the finding required a detailed risk evaluation because it caused a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to stable shutdown conditions (i.e., loss of main feedwater).

The senior reactor analyst performed the detailed risk evaluation by treating the finding as an initiating event and using the conditional core damage probability for a loss of main feedwater, as called for in Section 8.0, “Initiating Event Analyses,” of the Risk Assessment of Operational Events Handbook. The analyst used Arkansas Nuclear One, Unit 1, SPAR model, Version 8.55, run on SAPHIRE, software Version 8.1.8, for the evaluation.

The model was modified to reflect probabilistic recovery of a main feedwater pump. To accomplish this, the analyst adjusted Basic Event MFW-XHE-NOREC, “Operator Fails to Recover Main Feedwater,” used in Fault Tree MFW, “Main Feedwater System,” from TRUE to a SPAR-H human reliability model derived value of 6.0E-2. In this human reliability analysis, the analyst assigned high stress and moderate complexity to both diagnosis and action for the recovery. After reviewing normal operating, annunciator response, and abnormal operating procedures for main feedwater pumps, the analyst classified the action procedures to be available but poor. All other performance shaping factors were set to nominal. Performance of the initiating event analysis with this basic event adjustment yielded an estimate in the increase of core damage frequency of 6.9E-7 per year from internal events.

The analyst noted that this detailed risk assessment evaluates an actual event in which no external events occurred. Additionally, the period of time that the events impacted plant equipment was small enough that the probability of an external initiator occurring during this time would be negligible. Therefore, the analyst assumed that the risk from external events, given the subject performance deficiency was essentially zero. This resulted in a total estimate in the increase of core damage frequency of 6.9E-7 per year, making the finding of very low safety significance (Green).

The analyst noted that the licensee recently completed installation and acceptance of an additional train of feedwater, the common feedwater system, as a fire protection modification. This common feedwater system had not been incorporated or credited into the Arkansas Nuclear One, Unit 1, SPAR model. The analyst considered the system, which could have been used to aid in mitigation of losses of main feedwater, as a qualitative consideration which would further lower the increase in core damage frequency.

The loss of main feedwater events were the dominant sequences and were mitigated by the emergency and auxiliary feedwater systems.

The increase in large early release frequency from this finding was determined to be of very low safety significance (Green) using Inspection Manual Chapter 0609, Appendix H, “Containment Integrity Significance Determination Process,” because loss of main feedwater sequences screen as having low safety significance in pressurized water reactors with large dry containments.

Cross-cutting Aspect: This finding had a cross-cutting aspect in the area of human performance associated with teamwork because individuals failed to communicate and coordinate activities within organizational boundaries to ensure nuclear safety is maintained. Specifically, individuals did not work as a team to provide peer-checks and verify proper indication of pump MFW-B discharge pressure, to verify certifications and training, to ensure detailed safety practices, to actively peer coach personnel, and to share tools and publications.
<u>Enforcement</u>: Violation: Technical Specifications 5.4.1.a for Unit 1 requires, in part, that written procedures be established, implemented, and maintained covering applicable procedures in Appendix A to Regulatory Guide 1.33, "Quality Assurance Program Requirements," Revision 2, dated February 1978. Regulatory Guide 1.33, Appendix A, Section 2.b, requires general plant operating procedures for hot standby to minimum load (nuclear start-up). The licensee established Procedure OP-1102.002, "Plant Startup," Revision 106, to meet the Regulatory Guide 1.33 requirements. Procedure OP-1102.002, step 17.16.13, required with a main feedwater pump H/A station in hand and both main feedwater block valves closed, to adjust main feedwater pump speed to maintain discharge pressure between 1025 and 1075 psig on computer points P2833 and P2835. Contrary to the above, on May 16, 2018, the licensee failed to implement Procedure OP-1102.002, step 17.16.13, with a main feedwater pump B H/A station in hand and both main feedwater block valves closed, to maintain main feedwater pump B discharge pressure between 1025 to 1075 psig. This resulted in operators raising the speed of the only operating main feedwater pump until it tripped on high discharge pressure, causing an automatic reactor trip. Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-ANO-1-2018-03238.

Reactor Power Transient Caused by the Turbine Bypass Valve Failing Open			
Cornerstone	Significance	Cross-cutting Aspect	Inspection Procedure
Initiating Events	Green NCV 05000313/2018003-06 Closed	P.3 – Problem Identification and Resolution, Resolution	71153 – Follow-up of Events and Notices of Enforcement Discretion
The inspectors reviewed a self-revealed Green finding and associated non-cited violation of Arkansas Nuclear One, Unit 1, Technical Specifications 5.4.1.a, for the licensee's failure to properly preplan maintenance that can affect the performance of safety-related equipment. Specifically, the licensee failed to properly pre-plan maintenance for the replacement of air supply tubing for turbine bypass valve CV-6687, which resulted in the failure of the air tubing, causing valve CV-6687 to fail open, which led to a manual reactor trip and a subsequent loss of the main condenser.			
<u>Description</u>: On April 28, 2018, maintenance technicians repaired a section of leaking air tubing off of the turbine bypass valve air regulator vent line. This maintenance was done in			

accordance with Procedure EN-AM-156, “Compression Fitting Installation, Disassembly, Inspection, and Reassembly,” Revision 0, which allows for re-routing of tubing as deemed necessary by the technician. Although unknown at the time, the re-routed tubing was more susceptible to harmonic vibration.

On June 16, 2018, during a Unit 1 reactor startup at approximately 4 percent power, high vibrations caused a section of air supply tubing to fail. The localized loss of air pressure in turn caused turbine bypass valve CV-6687 to fail open. This bypass valve is a large, fast-acting steam valve that bypasses the normal lineup to the main turbine and dumps steam directly into the main condenser. When it failed open it caused a sudden increase in steam flow which caused a resulting reactor power increase and corresponding temperature decrease. Operators responded to the power increase and resulting cooldown in the reactor coolant system by inserting a manual reactor trip due to pressurizer level dropping below 100 inches. The event was terminated when steam generator B received an automatic main steam line isolation. This signal was caused by the difference in pressure between the two steam generators, caused by the increased steam flow only on the B steam generator because the failed open turbine bypass valve was supplied by that steam line. This automatic actuation isolated the B steam line, causing a loss of condenser vacuum when steam was lost to turbine gland sealing steam. The control room manually actuated main steam line isolation for the A steam line, and terminated the overcooling event and initiated emergency feedwater to remove decay heat.

The licensee entered this issue into their corrective action program as Condition Report CR-ANO-1-2018-03632 and performed a root cause evaluation. This root cause evaluation concluded that the corrective action was to replace the copper instrument air tubing with stainless steel and flex lines, which are more robust and appropriate for high vibration systems. The root cause also documented that a contributing cause was the insufficient controls over re-routing of tubing.

The inspectors noted a similar previous occurrence, documented in Condition Report CR-ANO-1-2016-00276, where air supply tubing failed causing a turbine bypass valve to fail open at 100 percent reactor power. In January 2016, air tubing came loose from the turbine bypass valve air regulator vent line causing it to fail open. During startup, these valves are designed to gradually open as reactor power increases until the point where the turbine/generator is connected to the grid, when the bypass valves will shut and remain shut for the duration of the cycle. When the valve failed open in 2016, it caused a slight increase in reactor power, but was ultimately controlled by operators taking manual action. The licensee determined at the time that this line was susceptible to high vibrations and an analysis was completed to determine if the vibrations had caused the failure. The licensee ultimately concluded after this analysis that it failed due to a poor fitting rather than vibration. The corrective actions for this event focused on training maintenance personnel on the standards for replacing turbine bypass valve air tubing.

Corrective Action: The licensee subsequently replaced the copper air tubing with a more robust stainless steel tubing prior to restarting the reactor.

Corrective Action Reference: Condition Report CR-ANO-1-2018-03632

Performance Assessment:

Performance Deficiency: The licensee's failure to have adequate work order instruction for air supply tubing replacement is a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor, and therefore a finding, because it was associated with the procedural quality attribute of the Initiating Events Cornerstone and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Specifically, the failed open turbine bypass valve resulted in a manual reactor trip and subsequent loss of the main condenser as a heat sink.
Significance: The inspectors assessed the significance of the finding using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, and determined that the finding required a detailed risk evaluation because it caused a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to stable shutdown conditions (i.e., loss of main feedwater and loss of main condenser).

A regional senior reactor analyst performed a detailed risk evaluation and determined that the finding associated with the main steam line isolation event was of very low safety significance (Green).

The analyst performed an initiating event analysis as called for in Section 8.0, "Initiating Event Analyses," of Volume 1, "Internal Events," of the RASP Handbook. The analyst chose to run this analysis as a Loss of Condenser Heat Sink Event since the main feedwater pumps and ability to dump steam to the condenser had been lost due to the event. The standard plant analysis risk (SPAR) model for Arkansas Nuclear One, Unit 1, did not credit the startup auxiliary feedwater pump P-75 for loss of condenser heat sink events. The analyst walked down this pump and its controls, interviewed operators and read design information and plant procedures to determine that the startup auxiliary pump would have been available if needed for the event. After sharing this information with Idaho National Laboratory, the analyst modified the SPAR model to credit the startup auxiliary feedwater pump for the event.

In support of crediting the pump, the analyst performed a human reliability analysis for starting and aligning the startup feedwater pump. The analyst assumed high stress and moderate complexity as performance drivers for both diagnosis and action attributes to derive a failure probability of 4.4E-2 in a SPAR-H human reliability analysis. This SPAR-H information was used to modify basic event AFW-XHE-XM-P75, "Failure to Start and Align AFW (P-75)," in the SPAR model. These modifications resulted in a change in core damage frequency of 6.8E-7/year for the finding. The analyst qualitatively considered that the common feedwater system could have been used to lower the increase in core damage frequency of the event even more, giving confidence that the finding was of very low safety significance (Green). Losses of condenser heat sink events comprised the most dominant core damage sequences. The high pressure injection and emergency feed water systems remained available for mitigation of the dominant sequences.

The analyst assumed that external events would be an insignificant contributor to the increase in core damage frequency because the probability of any external event coinciding with the main steam line isolation event would be extremely low. As a result, only the increase in core damage frequency from the initiating event was used in the final estimate.

After reviewing Inspection Manual Chapter 0609, Appendix H, "Containment Integrity Significance Determination Process," the analyst determined that main steam line isolation and loss of main feedwater sequences were not significant contributors to large early release frequency and screened the finding to Green for large early release frequency. The analyst ran the Arkansas Nuclear One, Unit 1, SPAR model, Revision 8.55, on SAPHIRE, Version 8.1.8, to calculate the conditional core damage probability using a cutset truncation of 1.0E-12. Cross-cutting Aspect: The finding had a cross-cutting aspect in the area of problem identification and resolution associated with resolution because the licensee failed to take effective corrective actions to address issues in a timely manner commensurate with their safety significance. Specifically, the licensee's corrective actions from the July 2016 event did not address the replacement of turbine bypass valve air tubing.
Enforcement: Violation: Technical Specification 5.4.1.a for Unit 1 requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures in Appendix A to Regulatory Guide 1.33, "Quality Assurance Program Requirements," Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 9.a, specifies that maintenance that can affect the performance of safety-related equipment should be properly pre-planned. Contrary to the above, on April 28, 2018, the licensee failed to properly pre-plan maintenance that can affect the performance of safety-related equipment. Specifically, the licensee failed to consider more robust materials for known high vibration situations and detailed instructions for routing the air tubing, resulting in a reactor trip, challenge to safety-related systems, and complicated recovery by operators. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy, because it was very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-ANO-1-2018-03632.

EXIT MEETINGS AND DEBRIEFS

On June 12, June 14, June 22, and June 28, 2018, the inspectors discussed the proposed emergency preparedness exercise scenario with Ms. D. Bordelon, Branch Chief, Technological Hazards Branch, FEMA Region VI, and other members of the FEMA regional staff.

On June 21, 2018, the inspectors discussed the proposed emergency preparedness exercise scenario with Mr. T. Renfroe, Emergency Planner, and other members of the licensee's staff.

On July 12, 2018, the inspectors presented the Unit 2 licensed operator requalification inspection results to Mr. R. Anderson, Site Vice President, and other members of the licensee staff. The inspectors verified no proprietary information was retained or documented in this report.

On July 19, 2018, the inspectors discussed the biennial emergency preparedness exercise with Ms. D. Bordelon, Branch Chief, Technological Hazards Branch, FEMA Region VI, and other members of the FEMA regional staff.

On July 26, 2018, the inspectors presented the results of the biennial emergency preparedness exercise inspection to Mr. R. Anderson, Site Vice President, and other members of the licensee staff. The inspectors verified no proprietary information was retained or documented in this report.

On September 13, 2018, the inspectors presented the Unit 1 licensed operator requalification inspection results to Mr. R. Martin, Training Superintendent, and other members of the licensee staff. The inspectors verified no proprietary information was retained or documented in this report.

On September 13, 2018, the inspectors presented the Unit 2 licensed operator requalification inspection results to Mr. M. Coffman, Acting Training Manager, and other members of the licensee staff. The inspectors verified no proprietary information was retained or documented in this report.

On September 24, 2018, the inspectors presented the results of the review of two emergency action levels to Mr. R. Anderson, Site Vice President, and other members of the licensee staff. The inspectors verified no proprietary information was retained or documented in this report.

On October 2, 2018, the inspectors presented the quarterly resident inspector inspection results to Mr. R. Anderson, Site Vice President, and other members of the licensee staff. The inspectors verified no proprietary information was retained or documented in this report.

DOCUMENTS REVIEWED

71111.01 – Adverse Weather Protection

Miscellaneous Documents

Number	Title	Date
0CAN030601	Response to Generic Letter 2006-02 for ANO-1 and ANO-2	March 29, 2006

Procedures

Number	Title	Revision
EN-FAP-WM-015	Unit Generation Forecasting for EMO/MISO	1
ENS-DC-201	ENS Transmission Grid Monitoring	7
OP-1015.033	ANO Switchyard and Transformer Yard Controls	28
OP-1107.001	Electrical System Operations	119
OP-1203.037	Abnormal ES Bus Voltage and Degraded Offsite Power	14
OP-2107.001	Electrical System Operations	126

71111.04 – Equipment Alignment

Condition Reports (CR-ANO-)

1-2018-04071	1-2018-04104	1-2018-04142	1-2018-04304
C-2018-02988			

Drawings

Number	Title	Revision
M-204	EFW Pump Turbine	
M-217, Sheet 4	P&ID Emergency Diesel Generator K-4A/K-4B Starting Air	9

Miscellaneous Documents

Number	Title	Revision
CALC-85-S-00002-01	ANO-2 Diesel Generator #1 (2K-4A) and #2 (2K-4B) Loading	22
CALC-91-E-0107-04	Emergency Diesel Generator (EDG) Fuel Consumption	4

Miscellaneous Documents

Number	Title	Revision
EC 71719	Temp Bulk Diesel Fuel Oil Storage for T-25 10 year Clean/Inspect	0

Procedures

Number	Title	Revision
EN-DC-136	Temporary Modifications	17
OP-1000.113	Diesel Fuel Monitoring Program	15
OP-1104.023	Diesel Oil Transfer Procedure	37
OP-2104.036	Emergency Diesel Generator Operations	96
OP-2107.002	ESF Electrical System Operation	38
OP-2202.007	Loss of Offsite Power	16
OP-220 2.010	Standard Attachments	26
ULD-1-SYS-01	Emergency Diesel Generator System	7

Work Orders

50236465	52697896

71111.05 – Fire Protection

Condition Reports (CR-ANO-)

1-2018-04487

Drawings

Number	Title	Revision
PFP-U2	FZ-2015 Fire Zone Detail Containment Building, Zone 2032K (South)	3
PFP-U2	FZ-2015 Fire Zone Detail Containment Building, Zone 2033K (North)	3

Miscellaneous Documents

Number	Title	Revision
7469	Fire Impairment	
CALC-95-R-0024-01	Basic Requirements for the Component Database on Station Doors and Hatches	14

Miscellaneous Documents

Number	Title	Revision
EC 71719	Temp Bulk Diesel Fuel Oil Storage for T-25 10 Year Clean/Inspect	0
PFP-U1	Unit 1 Prefire Plans	21
PFP-U2	Unit 2 Prefire Plans	17

Procedures

Number	Title	Revision
EN-DC-161	Control of Combustibles	18
EN-DC-330	Fire Protection Program	5
OP-1000.120	ANO Fire Impairment Program	25
OP-1003.014	ANO Fire Protection Program	9
OP-1015.052	Passive Barrier Breach Permit	1
OP-1104.32	Fire Protection Systems	89
OP-1405.016	Unit 1 Penetration Fire Barrier Visual Inspection	24
OP-2305.018	Underground Emergency Diesel Generator F.O. Tank 2T-57A/B Recirculation and Cleanup	16

Work Orders

50236465	52697896	52749902

71111.11 – Licensed Operator Requalification Program

Condition Reports (CR-ANO-)

1-2014-01062	1-2015-02327	1-2016-02615	1-2017-00164
1-2017-00387	1-2017-01567	1-2017-01750	1-2017-01764
1-2017-02073	1-2017-02166	1-2017-02195	1-2017-02518
1-2017-02709	1-2017-20169	1-2018-03238	2-2015-01544
2-2016-01666	2-2016-02614	2-2017-05397	C-2017-04438
C-2018-00285	C-2018-00785	C-2018-00989	C-2018-02348
C-2018-03067			

Miscellaneous Documents

Number	Title	Revision/ or Date
	ANO Unit 2 2018 RO Biennial Requalification Exam Week 2	
	ANO Unit 2 2018 SRO Biennial Requalification Exam Week 2	
	Arkansas Nuclear One Unit 2 Operations Training, Licensed Operator Requalification Cycle 2017-2018, Unit 2 Exam Sample Plan, 2018 Annual Requalification Exam	
	Operating Test	Week of June 4, 2018
	Operating Test	Week of July 9, 2018
	Unit 1 Licensed Operator 2017-2018 Requalification Cycle Report	
A1JPM-RO-AOP13	Perform RO #2 Follow-up Actions for Remote Shutdown without AFW Pump	9
A1JPM-RO-AOP40	Perform Compensatory Actions for Fires in Safety Related Areas	0
A1JPM-RO-EDG11	Reset Emergency Diesel Generator #2 Overspeed Trip Mechanism	10
A2JPM-RO-AOP02	Reset CIAS and Establish Cooling Water to Containment	1
A2JPM-RO-CCW02	Shift Running CCW Pumps	10
A2JPM-RO-CEA06	Recover a Dropped CEA	0
A2JPM-RO-EDDCB	Startup a Diesel Generator without DC Control Power (2K-4B)	3
A2JPM-RO-EFWRS	Reset the EFW Pump Trip Throttle Valve	16
A2JPM-RO-SDBC1	Perform a Restart/Reset of SDBCS after a Power Interruption	16
A2JPM-RO-SFPBMS	Make Up to SFP from BMS	0
A2JPM-RO-SIT06	Isolate SITs Following SIAS Actuation. SIAS has been reset	2
A2JPM-RO-SW02	Shift SW Pump 2P4A Suction & Discharge to ECP	8

Miscellaneous Documents

Number	Title	Revision/ or Date
A2JPM-RO-TLOF	Perform Local Actions to start ‘D’ Condensate Pump during a Loss of Feedwater	3
A2JPM-SRO-EAL17	Classify an Emergency Event	0
A2JPM-SRO-EAL21	Classify an Emergency Event	0
SES-1-010		
SES-1-036		
SES-2-007	Week 4 Scenario 1	7
SES-2-013	Week 4 Scenario 2	8
SES-2-031	Week 5 Scenario 1	8
SES-2-039	Week 5 Scenario 2	6

Procedures

Number	Title	Revision
1015.001	Conduct of Operations	117
1015.050	Time Critical Operator Actions Program	8
1063.008	Operations Training Sequence	43, 57
1102.004	Power Operation	70
COPD-032	Transient Conduct of Operations	9
DG-TRNA-015-CORETEST	Simulator Core Reload Acceptance Test	4
DG-TRNA-015-EXAMSEC	Simulator Exam Security Guidelines	15
DG-TRNA-015-SIMCONTROL	Simulator Modification Control	8
DG-TRNA-217-EXAMSECURITY	Exam Security	5
EN-NS-102	Fitness for Duty Program	20
EN-NS-112	Medical Program	18
EN-TQ-106	Training and Qualification of Training Personnel	19
EN-TQ-114	Licensed Operator Requalification Training Program Description	11
EN-TQ-202	Simulator Configuration Control	9

Procedures

Number	Title	Revision
EN-TQ-210	Conduct of Simulator Training and Evaluation	13, 14
EN-TQ-217	Examination Security	7
OP-1202.001	Reactor Trip	39
OP-1202.003	Overcooling	11
OP-1202.012	Repetitive Tasks	19
TQF-201-IM05	Remedial Training Plan	9

Scenario Based Testing

Number	Revision
SES-1-003	14
SES-1-010	1
SES-1-022	1
SES-1-027	11
SES-1-036	4
SES-1-046	0

71111.12 – Maintenance Effectiveness

Condition Reports (CR-ANO-)

1-2002-01147	1-2016-00097	1-2016-04925	1-2018-03567

Miscellaneous Documents

Number	Title
Item 63	ISO 17-MU-30, Sheet 2
ER-ANO-2003-0237-000	Reactor Coolant System Vent/Drain Vibration Reliability Enhancements

71111.13 – Maintenance Risk Assessments and Emergent Work Control

Condition Reports (CR-ANO-)

1-2018-04160	1-2018-04171	1-2018-04508	2-2018-01221
2-2018-01513	C-2018-02949	C-2018-03210	

Drawings

Number	Title	Revision
M-217, Sheet 1	P&ID Emergency Diesel Generator Fuel Oil Storage	90
M-2217, Sheet 1	P&ID Emergency Diesel Generator Fuel Oil Storage	64

Miscellaneous Documents

Number	Title	Revision
CALC-95-R-0024-01	Basic Requirements for the Component Database on Station Doors and Hatches	14
ER-963555-E202	Door 306, Watertight Door for Pump 2P-7A	0
ER-ANO-2004-0735-000	Risk Associated with Opening a HELB Door	0

Procedures

Number	Title	Revision
COPD-013	Operations Maintenance Interface	59
COPD-024	Risk Assessment Guidelines	66
EN-MA-125	Troubleshooting Control of Maintenance Activities	22
EN-OP-119	Protective Equipment Posting	9
EN-WM-104	On Line Risk Assessment	16
OP-1104.002	Makeup and Purification System Operation	94
OP-1202.012	Repetitive Tasks	19
OP-1203.012K	Annunciator K12 Corrective Action	49
OP-1203.043	Unit 1 Reactor Protection System Channel C Calibration	56
OP-1304.188	Unit 1 Red Channel High Pressure Injection Flow Instrument Calibration	10
OP-2104.036	Emergency Diesel Generator Operations	57
OP-2107.001	Off-Site Power Availability Check for #1 Emergency Diesel Generator Outage	126
OP-2305.018	Underground Emergency Diesel Generator F.O. Tank 2T-57A/B Recirculation and Cleanup	16

Work Orders

490354	490410	506897	50240475
52727213	52749902	52757471	52761176
52761178	52761344	52765620	52771820

71111.15 – Operability Determinations and Functionality Assessments

Condition Reports (CR-ANO-)

1-2008-00171	1-2009-00997	1-2016-00327	1-2018-03832
1-2018-04071	1-2018-04223	1-2018-04517	2-2018-01690
2-2018-01689	2-2018-01691	2-2018-01696	2-2018-01697
2-2018-01958			

Drawings

Number	Revision
B-27033	3
B-27033-F	3
MD20143	

Miscellaneous Documents

Number	Title	Revision
CEP-NDE-0100		10
EC 79057	2M-114A/B Gasket Material 2K-4A	0
ER-2004-0373-000		
ER974714R101	ECCS Flow Instrument Evaluation	0, 1
LBDCR-09-31		
STM 2-08	CS System	20
TD W180.0050	Instructions for Installing and Operating Seal Injection Water Coolers	2

Procedures

Number	Title	Revision
OP-1015.045	Unit 1 Safety Function Determination Program	2
OP-1104.002	Makeup and Purification System Operation	61, 65, 94
OP-1107.001	Electrical System Operations	119

Procedures

Number	Title	Revision
OP-1304.188	Unit 1 Red Channel High Pressure Injection Flow Instrument Calibration	10

Work Orders

52753544	52753544-01	52757471

71111.18 – Plant Modifications

Condition Reports (CR-ANO-)

1-2004-00802	1-2011-02615	1-2016-00327	1-2016-00944
1-2016-03465	1-2016-03502	1-2016-03550	1-2016-03593
1-2016-03614	1-2018-00406	1-2018-03832	2-2012-02083
2-2012-02336	C-2013-01885		

Miscellaneous Documents

Number	Title
EC-51099	
EC-54698	
EC-71145	
EC-72421	
ECN-76825	
ER99164N101	ANO-1 EFW Steam Supply Check Valve Replacement

Procedures

Number	Title	Revision
OP-1106.009	Turbine Startup (Warmup and Roll)	57
OP-1402.192	Static Load Test	9

Work Orders

436364

71111.19 – Post Maintenance Testing

Condition Reports (CR-ANO-)

2-2018-01513	2-2018-01690	2-2018-01693

Miscellaneous Documents

Number	Title
HIC-2648-ICNTRL	

Procedures

Number	Title	Revision
EN-DC-117	Post Modification Testing and Special Instructions	10
ER-ANO-2002-0335-000	2K4A CPT Mounting Feet	0
OP-1304.205	Unit 1 EFIC Channel A Monthly Test, SG Pressure Greater Than 750 PSIG	30
OP-2104.036	Emergency Diesel Generator Operations	96
OP-2106.006	Emergency Feedwater System Operation	98
OP-6030.110	Termination, Splicing and Soldering of Cable and Wire	23

Work Orders

490354-03	5269496-11	52727213	52744844
52761176	52761178	52761344	52765620
52771820			

71111.20 – Refueling and Other Outage Activities

Condition Reports (CR-ANO-)

2-2018-02183

Miscellaneous Documents

Number	Title	Revision
M-2204, Sheet 4	P&ID Emergency Feedwater	73
M-2232, Sheet 1	P&ID Safety Injection	122
M-2236, Sheet 1	P&ID Containment Spray	95

Procedures

Number	Title	Revision
OP-1015.008	Unit 2 SDC Control	59
OP-2104.004	Shutdown Cooling System	63
OP-2106.006	Emergency Feedwater System Operations	98
OP-2202.011	Lower Mode Functional Recovery	14
OP-2504.038	Hawke Seal Maintenance	8

71111.22 – Surveillance Testing

Condition Reports (CR-ANO-)

2-2018-01827

Miscellaneous Documents

Number	Title	Revision
CALC-92E-0078-04	Unit 2 EFW System Pump Performance Requirements	4
SEP-ANO-2-IST-1	ANO Unit 2 Inservice Testing Basis Document	3
SEP-ANO-2-IST-2	ANO Unit 2 Inservice Testing Plan	3
SEP-ANO-2-IST-3	ANO Unit 2 Inservice Testing Cross Reference Document	3

Procedures

Number	Title	Revision
OP-1104.036	Emergency Diesel Generator Operation	78
OP-1107.001	Verification of Two Offsite Circuit Power Sources	119
OP-2104.007	Control Room Emergency Air Conditioning and Ventilation	75
OP-2106.006	Emergency Feedwater System Operation	98

Work Orders

52725207

71114.01 – Exercise Evaluation

Condition Reports (CR-ANO-)

2-2016-02764	C-2016-03017	C-2016-03023	C-2016-03320
C-2016-05375	C-2017-01326	C-2017-01622	C-2018-02808
C-2018-02809	C-2018-02810	C-2018-02811	C-2018-02812

Miscellaneous Documents

Number	Title	Revision or Date
	Arkansas Nuclear One Emergency Plan	42
	Emergency Response Organization – Yellow Team Drill Report	March 1, 2017
	Emergency Response Organization – Red Team Drill Report	August 2, 2017
	Emergency Response Organization – Blue Team Drill Report	December 13, 2017
	Emergency Response Organization – Blue Team Drill Report	May 30, 2018
LO-ALO-2016-00121, CA		
LO-ALO-2017-00026	Pre-NRC Program Inspection Assessment	June 22, 2017
QA-7-2017-ANO-1	Quality Assurance Audit Report – Emergency Preparedness	July 11, 2017
QA-7-2018-ANO-1	Quality Assurance Audit Report – Emergency Preparedness	May 14, 2018

Procedures

Number	Title	Revision
1903.010	Emergency Action Level Classification	54, 55
1903.011	Emergency Response Notifications	55
1903.011-Z	Actions for Follow Up Notification	46
1903.030	Evacuation	33
1903.033	Protective Action Guidelines for Rescue and Repair and Damage Control Teams	26
1903.043	Duties of the Emergency Radiation Team	21
1903.064	Emergency Response Facility Control Room	16

Procedures

Number	Title	Revision
1903.080	Emergency Operations Facility Activation	2
1903.081	Technical Support Center Activation	0
1903.082	Operations Support Center Activation	1
1904.002	Offsite Dose Projection	43
1904.001	Emergency Radiological Controls	21
EN-EP-306	Drills and Exercises	9
EN-EP-308	Emergency Planning Critiques	5
EN-EP-310	Emergency Response Organization Notification System	5
EN-EP-609	Emergency Operations Facility Operations	4
EN-EP-610	Technical Support Center Operations	4
EN-EP-611	Operations Support Center Operations	5
EN-EP-801	Emergency Response Organization	15
EN-LI-102	Corrective Action Program	32, 33
EN-LI-104	Self-Assessment and Benchmark Process	13

71114.04 – Emergency Action Level and Emergency Plan Changes

Condition Reports (CR-ANO-)

2-2008-01439	C-2012-00749	C-2017-03161	C-2018-01121
C-2018-03597			

Miscellaneous Documents

Number	Title	Revision or Date
0CAN061802	Emergency Plan Implementing Procedure, Arkansas Nuclear One – Units 1 and 2, Docket Nos. 50-313, 50-368, and 72-13; License Nos. DPR-51 and NPF-6	June 28, 2018
0CAN071203	Response to Request for Additional Information Related to Proposed Emergency Action Levels Using NEI 99-01 Revision 5 Scheme, Arkansas Nuclear One – Units 1 and 2, Docket Nos. 50-313 and 50-368, License Nos. DPR-51 and NPF-6	July 9, 2012

Miscellaneous Documents

Number	Title	Revision or Date
0CAN121102	Proposed Emergency Action Levels Using NEI 99-01 Revision 5 Scheme, Arkansas Nuclear One – Units 1 and 2, Docket Nos. 50-313 and 50-368, License Nos. DPR-51 and NPF-6	December 1, 2011
A1LP-RO-RMS	Radiation Monitoring System	11
A2LP-RO-RMON	Radiation Monitoring	15
EC 18780	ANO-2 PASS Boundary Valve Permanent Abandonment	January 5, 2011
EC 18779	ANO-1 PASS Boundary Valve Permanent Abandonment	January 5, 2011
EN-EP-305, Attachment 9.2, 10 CFR 50.54(q)(3) Screening	Procedure/Document Number: 1903.010, Revision: 056, Title: Emergency Action Level Classification	June 13, 2018
EN-EP-305, Attachment 9.3, 10 CFR 50.54(q)(3) Evaluation	Procedure/Document Number: 1903.010, Revision: 056, Title: Emergency Action Level Classification	June 13, 2018
EN-EP-305, Attachment 9.3, 10 CFR 50.54(q)(3) Evaluation	Procedure/Document Number: 1903.069, Revision: 007, Title: Equipment Important to Emergency Response	September 13, 2017
ER-ANO-2003-0221-000	Isolation of the Unit 1 and Unit 2 PASS Systems	0
Form No. 1604.051B	Unit 2 SPING Monitor Log	25
FP-2112, Sheet 1	Arkansas Nuclear One, Unit 1, Fire Zones, Post Accident Sampling Facilities	8
M-237, Sheet 1	Arkansas Nuclear One, Unit 1, Piping & Instrumentation Diagram; Sampling System	59
M-2152	Arkansas Power & Light Company, Arkansas Nuclear One, Unit 2, Heating, Ventilation & Air Conditioning; Post Accident Sampling Facility; Air Flow Diagram	0
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C-2017-00716	C-2017-00743	C-2017-01851	C-2017-02910
C-2018-02783	C-2018-02829		

Procedures

Number	Title	Date
	Guidelines for Siren Warning System	
EP-02	Early Warning System	May 29, 2018

71152 – Problem Identification and Resolution

Condition Reports (CR-ANO-)

1-2017-03646	1-2018-03228	1-2018-03396	1-2018-03729
1-2018-03754			

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71153 – Follow-up of Events and Notices of Enforcement Discretion

Condition Reports (CR-ANO-)

1-2018-03228	1-2018-03238	1-2018-03567	1-2018-03632
1-2018-03633	1-2018-03634	1-2018-03636	1-2018-03643
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