UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

July 30, 2001

EA-01-201

Craig G. Anderson, Vice President,
Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, Arkansas 72801-0967

SUBJECT: ARKANSAS NUCLEAR ONE - NRC INSPECTION REPORT 50-313/01-02; 50-368/01-02

Dear Mr. Anderson:

On June 30, 2001, the NRC completed an inspection at your Arkansas Nuclear One, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on May 31 and July 10, 2001, with you and other members of your staff as described in Section 4OA6.

This inspection examined activities conducted under your licenses as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your licenses. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC determined that three violations occurred. The inspection identified one Severity Level IV violation of 10 CFR 50.59, one violation that did not affect a cornerstone (No Color), and one violation that was evaluated by the significance determination process and determined to have very low safety significance (Green). An additional finding was also evaluated using the significance determination process and was also determined to be of very low safety significance (Green). All of the violations and the finding were entered into your corrective action program. The violations are being treated as noncited violations (NCVs), consistent with Section VI.A.1 of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violation 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, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas

76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspectors at the Arkansas Nuclear One, Units 1 and 2 facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Linda Joy Smith, Chief
Project Branch D
Division of Reactor Projects

Dockets: 50-313
50-368
Licenses: DPR-51
NPF-6

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NRC Inspection Report
50-313/01-02; 50-368/01-02

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-313 50-368
Licenses:	DPR-51 NPF-6
Report:	50-313/01-02 50-368/01-02
Licensee:	Entergy Operations, Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	Junction of Hwy. 64W and Hwy. 333 South Russellville, Arkansas
Dates:	April 1 through June 30, 2001
Inspectors:	R. Bywater, P.E., Senior Resident Inspector K. Weaver, Resident Inspector M. Shannon, Senior Health Physicist
Approved By:	Linda Joy Smith, Chief, Project Branch D Division of Reactor Projects

SUMMARY OF FINDINGS

Arkansas Nuclear One, Units 1 and 2
NRC Inspection Report 50-313/01-02; 50-368/01-02

IR 05000313-01-02, IR 05000368-01-02; on 04/01-06/30/2001; Entergy Operations, Inc., Arkansas Nuclear One, Units 1 & 2. Equipment Alignment, Operability Evaluations, Surveillance Testing, ALARA Planning & Controls

The inspection was conducted by resident inspectors and a region-based senior health physicist. The inspection identified one Severity Level IV noncited violation of 10 CFR 50.59, one finding of very low safety significance (Green) that was a noncited violation, one noncited violation that did not affect a cornerstone (No Color), and one additional finding of very low safety significance. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html. Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

- Green. The inspectors determined that a violation of 10 CFR 50.59 occurred in that the licensee failed to adequately evaluate whether a change to the Updated Final Safety Analysis Report (UFSAR) was an unreviewed safety question (EA-01-201). Specifically, on September 2, 1998, the licensee completed a 10 CFR 50.59 screening determination which concluded that adding a statement to the Unit 2 UFSAR allowing Valve 2EFW-706 (emergency feedwater suction from the startup/blowdown demineralizer system) to be opened during surveillance testing did not require a safety evaluation per 10 CFR 50.59, because the change was inconsequential. However, the evaluation was inadequate because it did not recognize that the practice of opening Valve 2EFW-706 at greater than 10 percent power introduced a common mode failure potential not previously evaluated as acceptable by the NRC. Pump cavitation would occur after a loss of condensate pumps, making both pumps inoperable each time Valve 2EFW-706 was opened. This violation is being treated as a noncited violation and is in the licensee's corrective action program as Condition Reports (CRs) ANO-2-2001-349 and ANO-2-2001-440.

  Having both trains of emergency feedwater (EFW) inoperable during surveillance testing was evaluated using the SDP. The condition was determined to be of very low safety significance (Green) because: the cumulative time that Valve 2EFW-706 was open was only about 1 day per year; core damage required a concurrent loss of condensate pumps, failure to recover the EFW pumps, and failure to initiate once-through core cooling; and credit was given for operator recovery actions to close Valve 2EFW-706 on a loss of condensate pumps and for operator recovery actions to vent the EFW pumps (Section 1R22).

- Green. The inspectors identified that the Unit 1 decay heat vault purge ventilation isolation dampers had no leak testing requirements or controls during maintenance to ensure the accident analysis assumption of no leakage from the vaults was met. Also, reactor building penetration leakage in the decay heat vaults was not evaluated as a contributor to postaccident dose consequences. This failure to monitor damper degradation through testing or have positive controls on damper position during maintenance could have resulted in increased postaccident dose consequences. When dose consequences were considered for unsealed decay heat vaults, a more than minimal increase in control room thyroid dose resulted but was still within GDC 19 limits. This finding is in the licensee's corrective action program as CR ANO-1-2001-656.

This issue was characterized as having a very low safety significance (Green) using the SDP. The finding only represented a potential degradation of the radiological barrier function provided for the auxiliary building (Section 1R04).

- Green. The inspectors identified that a violation of 10 CFR Part 50, Appendix B, Criterion V, occurred in that the licensee failed to implement a program for trending boric acid corrosion evaluations as required by their procedure. The finding was determined to suggest a programmatic problem that had a credible potential to impact safety because any cumulative effects of repetitive boric acid leaks on reactor coolant system (RCS) barrier integrity could be missed. This violation is being treated as a noncited violation and is in the licensee's corrective action program as CR ANO-C-2001-050.

  The finding was found to have very low safety significance using the SDP because the RCS barrier was not in a degraded condition (Section 4OA5).

- No Color. On May 31, 2001, the inspectors identified three examples in which the licensee increased the person-rem exposure estimates on two radiation work permits (RWPs) without documenting the reasons why the additional exposure was necessary. The failure to document the reasons for the additional exposure is a violation of Technical Specification 6.8.1.(a). This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR ANO-C-2001-0297.

  The safety significance of this finding was determined to be more than minor because documenting the reasons for changing dose estimates was used, in part, to evaluate whether the task was being performed ALARA and to determine if additional ALARA controls were necessary. As a result, the failure to document the reasons for changing the person-rem estimates could cause additional unnecessary worker dose, resulting in a credible impact on safety. However, this issue did not affect the Occupational Radiation Safety cornerstone since there were no unplanned or unintended doses that resulted from actions contrary to Technical Specifications requirements (Section 2OS2).

B. <u>Licensee Identified Findings</u>

None.

Report Details

Summary of Plant Status

Unit 1 began the inspection period shut down in Refueling Outage 1R16. On April 8, 2001, Unit 1 operators made the reactor critical and commenced a power increase following completion of Refueling Outage 1R16. Unit 1 achieved 100 percent power on April 10. On May 5, Unit 1 main feedwater Pump P-1B tripped on low bearing oil pressure due to a lube oil pump failure. The integrated control system feedwater pump trip automatic runback to 40 percent power did not function as expected and plant power instead was automatically reduced by integrated control system cross-limits to approximately 59 percent power. The Unit 1 operators then manually commenced a power reduction to approximately 37 percent power. On May 6, following repairs to the failed lube oil pump, operators commenced a reactor power increase to 100 percent power. The unit operated at or near 100 percent power for the remainder of the inspection period.

Unit 2 operated at or near 100 percent power throughout the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R04 Equipment Alignment - Routine Inspection (71111.04)

.1 Unit 2 High Pressure Safety Injection System

a. Inspection Scope

Inspectors performed a partial walkdown of the Unit 2 Train A high pressure safety injection system to verify equipment operability during postmaintenance testing and temporary alteration of the Train B high pressure safety injection system.

In addition, the inspectors performed a partial walkdown of the Unit 2 Train B high pressure safety injection system to verify that proper system alignment had been restored following the temporary alteration and testing activities. The inspectors used Procedure 2104.039, Revision 40, "HPSI System Operation," and Procedure 2409.711, Revision 0, "2P89B Full Flow Test."

b. Findings

No findings of significance were identified.

.2 Startup Transformer 2

a. Inspection Scope

The inspectors performed a partial walkdown of Units 1 and 2 electrical systems to verify vital power availability during troubleshooting efforts for the indicated pressure increase on the Startup Transformer 2 sudden pressure relay. In addition, the inspectors also verified that danger flagging had been placed in the transformer yard and other work in close proximity to the transformer had been suspended during the troubleshooting efforts.

b. Findings

No findings of significance were identified.

.3 Unit 1 Penetration Room Ventilation System

a. Inspection Scope

The inspectors performed a partial system walkdown of the Unit 1 penetration room ventilation system. The inspection focused on ensuring that equipment used to collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from postaccident reactor building leaks met accident analysis assumptions. The following references were used:

- Procedure 1104.043, Revision 16, "Penetration Room Ventilation System"
- Procedure 1203.028, Revision 16, "Loss of Decay Heat Removal"
- Procedure 1104.035, Revision 21, "Fuel Handling and Rad Waste Ventilation"
- Procedure 1202.010, Revision 5, "ESAS"
- Procedure 1104.004, Revision 68, "Decay Heat Removal Operating Procedure"
- Procedure 1413.035, Revision 6, "Bettis Damper Actuator Cleaning and Inspection"
- Drawing M-262, Revision 43, Sheet 1, "HVAC Auxiliary Building and Rad Waste Areas"; Revision 15, Sheet 2; Revision 9, Sheet 3; Revision 3, Sheet 4
- System Training Manual 1-11, Revision 2, "Auxiliary Building, Spent Fuel, Penetration Room Ventilation Systems"
- UFSAR - Unit 1, Amendment 15
- Preventive Maintenance Task 4789, "CV-7635 Valve Operator for Decay Heat Room Purge"
- Maintenance Action Item 19647, "Corrective Maintenance on CV-7635 Valve Operator for Decay Heat Room Purge"

b. Findings

Upon review of the design, maintenance, and testing of the Unit 1 decay heat vault purge ventilation system, the inspectors found that purge isolation dampers had no leak testing program or positive controls during maintenance activities to ensure that accident analysis assumptions for decay heat vault leakage would not be exceeded. This was determined to be a finding of very low safety significance (Green).

The inspectors noted that the Unit 1 UFSAR, Section 6.5, provided a list of reactor building penetrations that did not pass through a penetration room serviced by the penetration room ventilation system. These penetrations were individually evaluated for acceptability. The inspectors identified that the list did not include reactor building penetrations located in the decay heat vaults (where the reactor building spray and low pressure injection pumps were located) and noted that neither of the vaults was serviced by the penetration room ventilation system.

The inspectors also noted that each decay heat vault was provided by a normal ventilation purge system described in Section 9.7.2.1 of the UFSAR. The UFSAR further stated that the purge systems are in use normally and are isolated if an emergency safeguards actuation occurs. Isolation of the purge system was not an automatic function, but was a proceduralized manual operator action to be taken after an emergency safeguards actuation had occurred, prior to initializing reactor building sump recirculation. The safety analysis of dose consequences following an accident described in Section 14.2 of the UFSAR identified that the reactor building spray and low pressure injection pumps were located in sealed rooms of the auxiliary building.

Upon review of the maintenance and testing program for the decay heat vault purge ventilation system, the inspectors determined that the licensee did not have a test program for leak tightness of the purge isolation dampers. The inspectors also determined that the licensee did not have a method of establishing configuration control of the dampers during actuator maintenance to ensure the dampers could be placed and maintained in their isolated position following an accident. The inspectors were concerned that, without an isolation damper testing program or configuration control during actuator maintenance, reactor building leakage from the penetrations in the decay heat vaults would not be contained and that this leakage was unaddressed by the UFSAR, Section 14, "Safety Analysis."

The licensee initiated CRs ANO-1-2001-622 and -656 in response to these findings. Safety Analysis Report Discrepancy Forms were initiated to address the UFSAR discrepancies. An operability evaluation was performed to determine the impact of low pressure injection system and reactor building spray system leakage from unsealed decay heat vaults. The results indicated that the increased exclusion area boundary thyroid dose consequences used 1.9 percent of the available safety analysis margin to the 300 rem acceptance criterion and that the increased control room thyroid dose consequences used 15.7 percent of the available margin to the 30 rem acceptance criterion. Based on these results, the inspectors concluded that the increase in consequences to control room operators for uncontained decay heat vault leakage was more than minimal (greater than 10 percent of the margin available in the safety analysis).

The licensee's corrective action plan included provisions to evaluate the necessity and feasibility of performing periodic leak rate testing on the subject dampers, initiate steps to provide positive control of damper position during maintenance activities, and evaluate and document the acceptability of installing a gagging device on damper actuators during maintenance activities.

This finding had a credible impact on safety. Undetected purge damper degradation or inability to close purge dampers when necessary following an accident could have resulted in a more than minimal increase in dose consequences to control room operators following an accident. The finding was of very low safety significance (Green), because the finding only represented a potential degradation of the radiological barrier function provided for the auxiliary building. This finding was in the barrier integrity cornerstone and the licensee has included this finding in their corrective action program as CR ANO-1-2001-656.

.4 Unit 2 EFW System

a. Inspection Scope

The inspectors performed partial walkdowns on three separate occasions of the Unit 2 EFW system to verify proper system alignment for operability. The inspectors used Procedure 2106.006, Revisions 52-3, 52-4, and 53, "Emergency Feedwater System Operations." The inspectors also referred to Procedure 2102.004, Revision 27, "Power Operation"; System Training Manual 2-19-2, Revision 8, "EFW & AFW Systems"; Calculation 94-E-0047-01, Revision 1, "Unit 2 EFW Pumps Net Positive Suction Head Evaluation"; and Drawing M-2204, Revision 62, Sheet 4, "Emergency Feedwater."

b. Findings

Findings associated with the EFW system are discussed in Section 1R22.

1R05 Fire Protection - Routine Inspection (71111.05)

a. Inspection Scope

The inspectors conducted tours of the areas listed below that are important to reactor safety to evaluate conditions related to licensee control of transient combustibles and ignition sources; the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and the fire barriers used to prevent fire propagation.

- Unit 2 lower north piping penetration room
- Unit 2 engineered safety features rooms

- Unit 1 Train A decay heat removal vault
- Unit 1 control room

b Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the UFSAR and other licensee flood protection documents. The inspectors conducted walkdowns of flood protection features of the Units 1 and 2 auxiliary buildings susceptible to internal and external flooding to verify that risk-significant equipment was adequately protected. The inspectors also inspected flood mitigation equipment located in these areas and the switchyard to verify that the licensee's flooding mitigation plans and equipment were consistent with design requirements.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On June 27, 2001, the inspectors observed two Unit 2 licensed operator simulator requalification annual examination scenarios and evaluator grading sessions. The inspectors evaluated licensed operator performance for adherence to principles of sound reactor plant operation. The inspectors evaluated the evaluator grading sessions to determine the depth of the evaluation and critical analysis of operator performance. The inspectors evaluated the simulator scenarios to determine if they were challenging and appropriate for the requalification program.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

Throughout the inspection period, the inspectors reviewed daily and weekly work schedules to determine when risk significant activities were scheduled. The inspectors reviewed selected activities regarding risk evaluations and overall plant configuration control to ensure that work was adequately planned, controlled, and executed. The inspectors also discussed emergent work issues with personnel and reviewed the potential risk impact of these activities as well. The specific activities reviewed were associated with planned and emergent maintenance on:

- Unit 1 EFW Pump P-7A speed control problems
- Startup Transformer 2 sudden pressure relay problems
- Unit 2 Emergency Diesel Generator 1 synchronization problem
- Review of impact of seismic event
- Unit 1 integrated control system anomaly identified following main feedwater pump trip
- Unit 2 high pressure safety injection Pump 2P-89B overhaul

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Evolutions (71111.14)

a. Inspection Scope

The inspectors observed, reviewed, and evaluated operator response to the following nonroutine plant evolutions and events to verify that operator response was appropriate and in accordance with procedures:

- Unit 1 reduced inventory conditions during Refueling Outage 1R16 and plant startup at the conclusion of the outage
- Unit 2 operator response to spent fuel pool level transient on May 3
- Unit 1 response to May 5, 2001, main feedwater Pump P-1B trip and power escalation following repairs (CRs ANO-1-2001-610 and ANO-1-2001-611)

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following operability evaluations for technical adequacy and assessed the impact of the condition on continued plant operation:

- CR ANO-1-2001-611, "Failure of Unit 1 Integrated Control System to Provide Runback on Main Feedwater Pump Trip"

- CR ANO-1-2001-656, "Unit 1 Decay Heat Vault Ventilation Issues"

- CR ANO-1-2001-675, "Unit 1 Steam Driven Emergency Feedwater Pump Turbine Speed Out of Tolerance"

- CR-ANO-2-2001-349, "Unit 2 Emergency Feedwater System Operability with Valve 2EFW-0706 Open During Surveillance Testing"

- CR ANO-2-2001-491, "Unit 2 Emergency Diesel Generator Test Terminated Due to Fluctuations in Reactive Load"

- CR ANO-1-2001-041, "Loss of Unit 1 Boric Acid Corrosion Evaluation Records from Refueling Outage 1R15"

- CR-ANO-C-2001-050, "Failure to Have and Implement a Program for Trending Boric Acid Corrosion Evaluations"

b. Findings

Findings of very low safety significance associated with CRs ANO-1-2001-656, ANO-2-2001-349, and ANO-C-2001-050 are discussed elsewhere in this report. No other findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed and/or evaluated the results from the following postmaintenance tests to determine whether the test adequately confirmed equipment operability.

- Full flow performance test of Unit 2 high pressure safety injection Pump 2P-89B in accordance with Work Plan 2409.711 following rotating assembly replacement and pump rebuild

- Operability testing of Unit 1 steam-driven EFW Pump P-7A in accordance with Procedure 1106.006, Supplement 12, Revision 61, "Steam Driven Emergency Feedwater Pump Quarterly Test," following turbine speed control maintenance

- Operability testing of Unit 1 reactor building spray Pump P-35B in accordance with Procedure 1104.005, Supplement 5, Revision 41, "Reactor Building Spray Pump P-35B Quarterly Test," following a system outage for routine preventive maintenance

- Operability testing of Unit 2 Emergency Diesel Generator 1 in accordance with Procedure 2104.036, Supplement 1B, Revision 45, "2DG1 Monthly Test (Slow Start)," following troubleshooting of voltage regulator problem

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed plant conditions and observed selected refueling outage activities throughout the outage to verify that the licensee maintained the plant in a configuration consistent with the requirements of Technical Specifications and with the assumptions of the Shutdown Operations Protection Plan. The inspectors verified that emergent issues were properly assessed for their impact on plant risk.

Electrical power availability was periodically verified to meet Technical Specification requirements and outage risk assessment recommendations. Control room operators were interviewed to determine if they were cognizant of plant conditions. The inspectors reviewed equipment tagout activities, controls for reactivity management, and RCS inventory. The inspectors also conducted tours of the reactor building prior to plant startup to verify adequate containment cleanliness for emergency core cooling system sump operability existed.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed the performance of and/or reviewed documentation for the following surveillance tests. Applicable test data was reviewed to verify whether they met Technical Specification, UFSAR, and licensee procedure requirements. Also, the inspectors verified that the testing effectively demonstrated that the systems were operationally ready and capable of performing their intended safety functions and that identified problems were entered into the corrective action program for resolution.

- Procedure 2106.006, Revision 52-03, "Emergency Feedwater System Operations," Supplement 1, "2P-7A Quarterly Surveillance," and Supplement 2, "2P7-B Quarterly Surveillance"

- Procedure 2106.006, Revision 52-04, "Emergency Feedwater System Operations," Supplement 1, "2P-7A Quarterly Surveillance," and Supplement 2, "2P7-B Quarterly Surveillance"

- Procedure 2106.006, Revision 53-00, "Emergency Feedwater System Operations," Supplement 1, "2P-7A Quarterly Surveillance," and Supplement 2, "2P7-B Quarterly Surveillance"
- Procedure 2104.039, Revision 40, "HPSI System Operation," Supplement 2, "2P-89B Quarterly Test"
- Procedure 1104.036, Revision 39, "Emergency Diesel Generator Operation," Supplement 2, "DG2 Monthly Test"
- Procedure 1106.006, Revision 61, "Emergency Feedwater System Operations," Supplement 12, "Steam Driven EFW Pump Quarterly Test"

b. Findings

A Severity Level IV violation (noncited) of 10 CFR 50.59(b)(1) was identified for not performing an adequate evaluation in 1998 of a change made to the UFSAR to determine if the change involved an unreviewed safety question (EA-01-201). 10 CFR 50.59 was recently revised and the NRC Enforcement Policy allows for discretion in cases involving violations of the old rule that are not violations of the new rule that went into effect after June 4, 2001. The inspectors determined that discretion was not appropriate in this case, because an evaluation of this change to the UFSAR would also be required by the new rule and because a license amendment would be required for the change that was made. The issue associated with this finding was evaluated using the SDP and was determined to be a finding of very low safety significance (Green).

ANO Unit 2 has two trains of safety-related EFW (one motor-driven pump and one turbine-driven pump) and one nonsafety-related auxiliary feedwater (AFW) pump, which was added in the early 1990s. All three pumps currently share a common condensate supply pipe which is fed from the condensate storage tanks or the startup/blowdown demineralizer system.

The inspectors identified that the licensee has an NRC commitment to maintain startup/blowdown supply Valve 2EFW-706 (EFW suction from the startup/blowdown demineralizer system) locked closed when at greater than 10 percent power. This commitment was in response to an event that occurred on April 7, 1980, when a tornado caused a loss of offsite power to Units 1 and 2. The event was reported to the NRC in Licensee Event Report 50-368/80-18. Both of the Unit 2 EFW pumps (i.e., steam-driven Pump 2P-7A and motor-driven Pump 2P-7B) experienced flow oscillations due to cavitation and were sequentially stopped and manually vented during the posttrip natural circulation RCS cooldown.

Prior to the 1980 event, Valve 2EFW-706 was a motor-operated, normally open valve providing a source of water to the safety-related EFW pumps from the startup/blowdown demineralizer system. Following the 1980 event, the licensee replaced the valve's motor actuator with a manually operated actuator with remote position indication and documented in the original UFSAR the commitment to maintain the valve closed when at greater than 5 percent power. In UFSAR, Amendment 4, the licensee changed the commitment to maintain the valve closed when at greater than 10 percent power. With Valve 2EFW-706 closed, the normal source of water to the pumps was the nonsafety-related condensate storage tanks.

The inspectors identified that the current Unit 2 UFSAR, Amendment 15, stated that Valve 2EFW-706 may be temporarily opened during surveillance testing. Procedure 2106.006 required an operator to maintain communication with the control room to close the valve on a loss of condensate pumps, but no Technical Specification limiting condition for operation was entered when Valve 2EFW-706 was opened. The licensee stated that it would not be uncommon for this designated operator to have other concurrent duties and to not be stationed directly at Valve 2EFW-706.

The licensee had identified a design bases document discrepancy open item in 1992 concerning the practice of unlocking and opening Valve 2EFW-706 for surveillance testing of the EFW and AFW pumps given the NRC commitment following the 1980 loss of offsite power event. An Engineering Request 974539 and 10 CFR 50.59 screening, performed to support a UFSAR change to reconcile the discrepancy, was completed September 2, 1998, and the UFSAR was updated in 1999 (Amendment 15). A 10 CFR 50.59 safety evaluation was not performed because the licensee concluded that the proposed change was an “inconsequential change” to the SAR for which a safety evaluation was not required per their program document (Procedure 1000.131, Revision 3, "10 CFR 50.59 Review Program"). Therefore, the condition was not evaluated to determine if an unreviewed safety question existed.

The resident inspectors concluded that a scenario in which a loss of offsite power during the time that the EFW pump suction was in this alignment could result in steam formation in both EFW pumps and the nonsafety AFW pump. Similar to the 1980 event, operators would be required to vent each pump in order to recover the system. Certain events involving loss of the power conversion system (i.e., feedwater and condensate) could also result in the loss of EFW when the system was in this test alignment. The practice of opening Valve 2EFW-706 at greater than 10 percent power introduced a common mode failure potential which created the possibility of a malfunction of equipment important to safety of a different type than previously evaluated. The inspectors concluded that opening Valve 2EFW-706 when at greater than 10 percent power was an unreviewed safety question. This position is consistent with the current rule because creating a possibility for an accident of a different type than any previously evaluated in the SAR is a condition that would require the licensee to obtain a license amendment prior to the change [refer to 10 CFR 50.59 (c)(2)(v)].

The inspectors also concluded, with support of technical staff in the NRC Office of Nuclear Reactor Regulation, that opening Valve 2EFW-706 resulted in both EFW system trains being inoperable. The EFW system was inoperable during surveillance testing when the SU/BD DI line was unisolated and a designated watch was not established locally at the valve. Using the time response calculation methodologies in ANSI/ANS-58.8-1984 and 1994, "Time Response design Criteria for Nuclear Safety Related Operator Actions," at least 14 minutes should be needed to reliably respond locally to isolate the line. Licensee analysis stated that after 5.2 minutes steam would start to degrade the operation of the EFW pumps. In addition, the licensee's compensatory measures to maintain an operator in communication with the control room when Valve 2EFW-706 was opened did not meet guidance for crediting operator actions to maintain operability of the EFW system described in Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," or Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times." To address this finding, the licensee initiated a condition report and revised their surveillance procedure to prelude opening Valve 2EFW-706 at power.

This issue had a credible impact on plant safety and affected the operability of both trains of the EFW system, which affected the mitigating systems cornerstone. Using the Phase 1 worksheet of the SDP in NRC Manual Chapter 0609, the inspectors concluded that the finding represented an actual loss of safety function of the EFW system because both trains of EFW were inoperable during surveillance testing. This conclusion required a Phase 2 evaluation.

With the assistance of an NRC Region IV senior reactor analyst and technical staff of the NRC Office of Nuclear Reactor Regulation, and with information provided by the licensee's staff documented in CR 2-2001-349, the inspectors performed the evaluation using the SDP Phase 2 site-specific worksheets for Unit 2. The initiating event chosen was a loss of offsite power (with sensitivity evaluation for other causes of a loss of condensate pumps). The exposure time was estimated at 24 hours per year, resulting in an estimated likelihood rating of D in Table 1 of the "Risk-Informed Inspection Notebook," for Arkansas Nuclear One, Unit 2.

The licensee's Individual Plant Evaluation (1992) identified that operators had 55 minutes following an initiating event to initiate once-through core cooling if feedwater was not available. Upon loss of condensate, the licensee determined that steam formation in the EFW pump suction would occur in approximately 5.2 minutes, but EFW flow termination or pump damage would not occur immediately. The inspectors reviewed the licensee's human recovery analysis and determined that recovery of at least a single train of EFW by closing Valve 2EFW-706 and venting the EFW pumps using guidance contained in Procedure 2106.006 was reasonable. Crediting recovery of a failed train of EFW in each of the affected core damage sequences resulted in none of the sequences having a total remaining mitigation capability rating that resulted in greater than Green in the Risk Significance Estimation Matrix (Table 4) of Manual Chapter 0609. In summary, the issue was determined to be of very low safety significance because, while both trains of EFW were inoperable during the time that Valve 2EFW-706 was open, the cumulative exposure time was only about 1 day per year and core damage required a concurrent loss of condensate pumps, failure to recover the EFW pumps, and failure to initiate once-through core cooling. Operator recovery actions to close Valve 2EFW-706 on a loss of condensate pumps and to vent the EFW pumps were credited (Green).

The inspectors concluded that the EFW system should have been considered inoperable during surveillance testing and the licensee was required to enter Technical Specification 3.0.3 and initiate action to shut down the facility. However, duration of the test alignment condition was less than 2 hours. Therefore, the inspectors concluded that the licensee would have never exceeded the Technical Specification 3.0.3 action statement to be in hot standby within 7 hours and that no Technical Specification 3.0.3 violation had occurred.

However, in 1998 10 CFR 50.59 (b)(1) required, in part, that the licensee shall maintain records of the changes to the facility and that these records must include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question. 10 CFR 50.59(a) stated, in part, that a proposed change, test, or experiment shall be deemed to involve an unreviewed safety question if a possibility for malfunction of a different type than any evaluated previously in the safety analysis report may be created.

The inspectors identified a violation of 10 CFR 50.59 (b)(1) for the inadequate screening evaluation performed September 2, 1998. Specifically, the evaluation did not recognize that opening Valve 2EFW-706 constituted an unreviewed safety question. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (50-368/0102-01). This violation is in the licensee's corrective action program as CRs ANO-2-2001-349 and -440.

The circumstances in this case would also be in violation of the current rule on 10 CFR 50.59 in that an evaluation would also be required and a license amendment would be required for this change.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Alteration 01-2-001 that was installed to provide a mechanical bypass around the Unit 2 Pump 2P-89B high pressure safety injection miniflow recirculation orifice to allow full flow performance testing of the pump following maintenance. The inspectors evaluated this modification and associated 10 CFR 50.59 safety evaluation against the system design basis documentation and verified that the modification did not adversely affect system operability when not in use for surveillance testing. The mechanical bypass was required to be isolated for Pump 2P-89B to be considered operable. The inspectors verified that the licensee had established acceptable administrative controls to ensure the mechanical bypass was isolated prior to operators declaring Pump 2P89-B operable.

b. Findings

No findings of significance were identified.

Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed portions of the off-hours unannounced emergency preparedness drill conducted on June 7, 2001, to evaluate emergency response organization performance and adequacy of the licensee's critique. The drill was conducted in the Unit 1 simulator, and all onsite emergency facilities (emergency operations facility, technical support center, and the operations support center) were activated.

The inspectors also observed an off-hours unannounced emergency response organization staffing drill on June 27, 2001.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors interviewed radiation workers and radiation protection personnel throughout the radiologically controlled access area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- ALARA program procedures
- Radiation Protection Department Fourth Quarter Assessment Year 2000
- Processes used to estimate and track exposures
- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information
- Three RWP packages for refueling outage work activities which resulted in the highest personnel collective exposures during Refueling Outage 2R-14 (RWP 2000-2004, "Routine Maintenance Activities"; RWP 2000-2048, "Remove and Replace Scaffolding and Insulation"; and RWP 2000-2206, "Steam Generator RCS Piping Work")

- Use of engineering controls to achieve dose reductions, including temporary shielding
- Hot spot tracking and reduction program
- Radiological work planning
- A summary of ALARA and radiological worker performance related to corrective action reports written since October 1, 2000 (12 corrective action reports were reviewed in detail: CRs ANO-1-2000-0468, ANO-1-2001-0330, ANO-1-2001-0481, ANO-2-2000-0565, ANO-2-2001-0275, ANO-C-2000-0336, ANO-C-2000-0354, ANO-C-2001-0054, ANO-C-2001-0067, ANO-C-2001-0111, ANO-C-2001-0179, and ANO-C-2001-0194)
- Declared pregnant worker dose monitoring controls

No work was performed in high exposure or high radiation areas during this inspection. Therefore, this aspect of the above procedure could not be evaluated.

b. Findings

Three examples of a noncited violation, that did not affect the occupational radiation safety cornerstone (No Color), were identified for the failure to document the reasons for changing person-rem estimates for two RWPs. On May 31, 2001, the inspectors identified that on two occasions (November 21 and December 3, 2000) the licensee increased the person-rem job exposure estimate for RWP 2000-2004, Revisions 4 and 5 (7.2 to 7.8 person-rem and from 7.8 to 8.3 person-rem, respectively), without documenting the reasons why the additional exposure was necessary. Additionally, on November 22, 2000, the RWP 2000-2048, Revision 3, person-rem estimate was increased (4.6 to 5.2 person-rem) without documenting the reasons why the additional exposure was necessary. During discussions with radiation protection management, the inspector was informed that documenting the reasons for the increased exposure was necessary to: (1) help ensure proper dose tracking, (2) ensure that the task was being performed ALARA, and (3) determine if additional ALARA controls were needed.

The significance of this finding was determined to be more than minor because documenting the reasons for changing dose estimates was used, in part, to evaluate whether the task was being performed ALARA and to determine if additional ALARA controls were necessary. As a result, the failure to document the reasons for changing the person-rem estimates could cause additional unnecessary worker dose, resulting in a credible impact on safety. However, this issue did not affect the occupational radiation safety cornerstone since there were no unplanned or unintended doses that resulted from actions contrary to Technical Specifications requirements.

Technical Specification 6.8.1.(a) requires procedures for the ALARA program. Section 6.4.2 of Procedure 1012.019, Revision 6-3, “Radiological Work Permits,” states “when an RWP is revised to change the total man-rem or man-hour estimate, document the reasons for this change on Form 1012.019C, "ALARA RWP Supplemental Job History." The failure to document the reasons for changing the person-rem estimates on the above RWPs represents three examples of a violation of Technical Specification 6.8.1.(a). This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR ANO-C-2001-0297 (NCV 50-313/0102-02; 50-368/0102-02).

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Initiating Events Cornerstone

a. Inspection Scope

The inspectors reviewed unit logs, plant thermal performance records, monthly operating reports, and licensee event reports for the first quarter of 2001 to verify the accuracy and completeness of data used to calculate and report the following performance indicators:

Unit 1 unplanned scrams per 7000 critical hours
Unit 2 unplanned scrams per 7000 critical hours
Unit 1 unplanned power changes per 7000 critical hours

b. Findings

No findings of significance were identified. The performance indicators all remained in the licensee response band (Green).

4OA3 Event Followup (71111.14, 71153)

.1 Seismic Event

a. Inspection Scope

On May 4, 2001, at 1:42 a.m., a 4.4 magnitude earthquake occurred with an epicenter located approximately 55 miles east of the plant near Greenbrier, Arkansas. The earthquake was felt by personnel outside of the plant and in the turbine building, but not in Seismic Category I structures. No damage occurred and the earthquake was of insufficient magnitude to actuate any seismic alarms. The inspectors reviewed the licensee's followup of the event to confirm that no damage occurred to safety-related equipment and that no emergency action level declarations were required. The inspectors reviewed Procedures 1203.025, Revision 18, "Natural Emergencies"; 2203.008, Revision 8, "Natural Emergencies"; 1903.010, "Emergency Action Level

Classification"; and 1203.012M, Revision 28, "Annunciator K15 Corrective Action," and confirmed that CR ANO-C-2001-0237 adequately addressed a procedure weakness that was identified by the inspector.

b. Findings

No findings of significance were identified.

.2 Unit 1 Main Feedwater Pump Trip

a. Inspection Scope

The inspectors reviewed plant response and the licensee's posttransient review report following the May 5, 2001, trip of main feedwater Pump P-1B. This included review of Procedure 1105.004, Revision 13, "Integrated Control System," and CRs ANO-1-2000-492, ANO-1-2001-610, and ANO-1-2001-611.

b. Findings

No findings of significance were identified.

4OA5 Other

.1 License Renewal Inspection Followup

a. Inspection Scope (71111.15)

The inspectors reviewed issues identified during an NRC inspection of the licensee's aging management programs necessary for Unit 1 license renewal and documented them in NRC Inspection Report 50-313/01-03; 50-368/01-03. The issues involved implementation of the licensee's boric acid corrosion prevention program. The inspectors reviewed Procedure 5000.005, Revision 1, "Boric Acid Corrosion Prevention Program Administration," and CRs ANO-1-2001-0041 and ANO-C-2001-0050 to confirm adequate implementation of the licensee's boric acid corrosion prevention program.

b. Findings

The inspectors identified that the licensee failed to implement a program for trending boric acid corrosion evaluations as required by their procedure. This was a violation of 10 CFR Part 50, Appendix B, Criterion V, which had very low safety significance (Green).

Procedure 5000.005 required that "System Engineering shall trend and maintain all boric acid evaluations and determine repetitive problem areas that can be addressed during each refueling outage." As discussed in NRC Inspection Report 50-313/01-03; 50-368/01-03, a program for trending boric acid corrosion evaluations had not been implemented. The inspectors did not, however, identify any concern that boric acid corrosion evaluations were not being performed as leaks were identified.

The finding had a credible potential to impact safety because any cumulative effects of repetitive boric acid leaks on RCS barrier integrity could be missed. The finding was found to have very low safety significance (Green) because the RCS barrier was not in a degraded condition.

10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Trending of boric acid corrosion evaluations was an activity affecting quality prescribed by Procedure 5000.005. Failure to follow the requirements of Procedure 5000.005 is a violation of 10 CFR Part 50, Appendix B, Criterion V. This issue was captured in the licensee's corrective action program as CR ANO-C-2001-0050. As a result, the violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-313/0102-03).

4OA6 Exit Meeting Summary

The regional senior health physicist presented the ALARA planning and controls inspection results to Mr. Bob Bement, General Manager, and other members of licensee management on May 31, 2001. The licensee acknowledged the findings presented.

The resident inspectors presented the inspection results to Mr. Craig Anderson, Vice President, and other members of licensee management on July 10, 2001. The licensee acknowledged the findings presented.

The inspectors asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

KEY POINTS OF CONTACT

Licensee

C. Anderson, General Manager, Plant Operations
B. Bement, General Manager Plant Operations
S. Bennett, Licensing
E. Blackard , Mechanical/Civil/Structural Design Engineering
M. Chisum, Manager, Unit 2 System Engineering Manager
B. Converse, Quality Assurance
M. Cooper, Licensing Specialist
B. Day, Unit 1 Acting Director, Engineering
G. Dobbs, Electrical and Instrumentation Control Design Engineering Supervisor
N. Eggemeyer, Technical Support Manager
C. Eubanks, Project Manager Power Uprate
D. Fouts, Supervisor, Nuclear Safety Analysis
D. Fowler, Supervisor, Quality Assurance
B. Gordon, Unit 2 Outage Manager
J. Hoffpauir, Unit 2 Plant Manager
B. James, Maintenance Manager
D. James, Licensing Manager
J. Kowalewski, Unit 1 System Engineering Manager
R. Lane, Director, Design Engineering
M. Little, Unit 1 Assistant Operations Manager
D. MacPhee, Mechanical/Civil/Structural Design Engineering
R. Martin, Unit 2 Assistant Operations Manager
T. Nickels, Superintendent, Radiation Protection
S. Pyle, Licensing Specialist
D. Sealock, Supervisor, Training
B. Smart, Unit 2 Instrumentation and Control Coordinator
D. Stoltz, Supervisor, Radiation Protection
M. Stroud, Design Engineering Manager
J. Vandergrift, Director, Nuclear Safety
C. Zimmerman, Unit 1 Plant Manager

ITEMS OPENED AND CLOSED

Opened and Closed During this Inspection

50-368/0102-01	NCV	Failure to perform a 10 CFR 50.59 evaluation for a change to the Unit 2 SAR that caused both trains of EFW to be inoperable during surveillance testing (Section 1R22)
50-313/368/0102-02	NCV	Failure to document and evaluate the reasons for increasing dose estimates (Section 2OS2)
50-313/0102-03	NCV	Failure to implement boric acid corrosion evaluation trending program (Section 4OA5)

ACRONYMS

AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
CR	condition report
EFW	emergency feedwater
RCS	reactor coolant system
RWP	radiation work permit
SDP	significance determination process
UFSAR	Updated Final Safety Analysis Report