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

October 6, 2000

EA-00-202

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

SUBJECT: NRC INSPECTION REPORT NO. 50-313/00-16; 50-368/00-16

Dear Mr. Anderson:

On September 15, 2000, the NRC completed a safety inspection at your Arkansas Nuclear One, Units 1 and 2, facility. The enclosed report presents the results of this inspection.

The inspection was an in-office examination of Unresolved Item 50-368/00-08-01. The finding involved a fire door, separating the Unit 2 vital switchgear rooms, that was found inoperable. As documented in NRC Inspection Report 50-313/00-08; 50-368/00-08, this finding was unresolved pending completion of an NRC review of the risk significance. Based on this review, as described in the enclosed inspection report, this finding is now being characterized as an apparent violation of NRC requirements.

This issue was assessed using the applicable significance determination process as an apparent significant finding that was preliminarily determined to be white. The enclosed report provides further amplification to support the NRC's preliminary significance determination. This is an issue of some increased importance to safety, which may require additional NRC inspection.

Although we believe that we have sufficient information to make our final significance determination for the issue, we are giving you the opportunity to provide us your position on the finding's risk significance and the apparent violation. If you choose to do so, you may provide your position either at a regulatory conference or in writing. If you chose to provide your position in writing, you should do so within 30 days of the date of this letter. Should you request a regulatory conference, it will be open for public observation. In addition, should you request a regulatory conference, we request that you submit a written response at least one week prior to the conference, as this would allow for a clearer understanding of any significance evaluation or regulatory differences and thus a more productive conference. Should you agree with our characterization of the significance of the issue and the apparent violation, a written response or a regulatory conference is not necessary. Please contact Mr. Phil Harrell at 817/860-8250 within 7 days of the date of this letter to notify the NRC of your intended response.

Please be advised that the number and characterization of the apparent violation described in the enclosed report may change as a result of further NRC review. You will be advised by separate correspondence of the results of our deliberations on this matter.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be 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).

Sincerely,

E. E. Collins for

Ken E. Brockman, Director
Division of Reactor Projects

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

Enclosure:
NRC Inspection Report No.
50-313/00-16; 50-368/00-16

cc w/enclosure:
Executive Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

Vice President
Operations Support
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

Manager, Washington Nuclear Operations
ABB Combustion Engineering Nuclear
Power
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

County Judge of Pope County
Pope County Courthouse
100 West Main Street
Russellville, Arkansas 72801

Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

David D. Snellings, Jr., Director
Division of Radiation Control and
  Emergency Management
Arkansas Department of Health
4815 West Markham Street, Mail Slot 30
Little Rock, Arkansas 72205-3867

Manager
Rockville Nuclear Licensing
Framatome Technologies
1700 Rockville Pike, Suite 525
Rockville, Maryland 20852

Electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(RLB3)**
Branch Chief, DRP/D **(PHH)**
Senior Project Engineer, DRP/D **(KMK)**
Branch Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**

Only inspection reports to the following:
David Diec **(DTD)**
NRR Event Tracking System **(IPAS)**
ANO Site Secretary **(VLH)**
Dale Thatcher **(DFT)**
G. F. Sanborn, D:ACES **(GFS)**
K. D. Smith, RC **(KDS1)**
R. W. Borchardt, OE **(RWB1)**
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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-313; 50-368
License Nos.:	DPR-51; NPF-6
Report No.:	50-313/00-16; 50-368/00-16
Licensee:	Entergy Operations, Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	1448 S. R. 333 Russellville, Arkansas 72801
Dates:	June 26 through September 15, 2000
Inspector:	R. Bywater, Senior Resident Inspector
Approved by:	P. Harrell, Chief, Project Branch D Division of Reactor Projects

ATTACHMENTS:

Attachment 1: Supplemental Information
Attachment 2: NRC's Revised Reactor Oversight Process

SUMMARY OF FINDINGS

Arkansas Nuclear One, Units 1 and 2
NRC Inspection Report 50-313/00-16; 50-368/00-16

IR 05000313-00-16, IR 05000368-00-16, on 6/26 - 9/15/2000; Entergy Operations, Inc.; Arkansas Nuclear One; Units 1 & 2. Special Report. Fire Prot.

This report documents the in-office review of a previously identified unresolved item.

The significance of issues is indicated by their color (green, white, yellow, or red) and was determined by the significance determination process in Inspection Manual Chapter 0609. The body of the report is organized under the broad categories of Reactor Safety and Other Activities as reflected in the summary below.

Mitigating Systems

- White. On June 26, 2000, the inspectors identified that Fire Door 269, which separates the North and South Unit 2 vital switchgear rooms, was not operable. Fire Door 269 was a 3-hour rated fire barrier that separated redundant trains of equipment necessary for safe shutdown of the reactor following a fire. It consisted of a double door with one side normally latched at the top and bottom in a stationary position. This stationary door was found not latched and both doors could be pushed open with slight pressure. This condition existed for approximately 3.5 days. The failure to maintain Fire Door 269 operable was identified as an apparent violation of ANO Unit 2 License Condition 2.C.(3)(b), "Fire Protection."

  This apparent violation was preliminarily determined to be white using the significance determination process. It was identified as an apparent significant finding based on the fact that a postulated fire in the south switchgear room could damage one train of safe shutdown equipment and could propagate through the inoperable fire door into the north switchgear room and damage the redundant train of safe shutdown equipment. The simultaneous unavailability of both trains of safe shutdown equipment could result in the loss of a mitigation function necessary to prevent core damage in the event of a fire-induced transient (Section 1R05.2).

Report Details

Summary of Plant Status: Unit 2 was operating at 100 percent power when Fire Door 269 was inoperable from June 22-26, 2000.

1. REACTOR SAFETY
Cornerstone: Mitigating Systems

1R05 Fire Protection

.1 (Closed) Unresolved Item 50-368/00-08-01: Inoperability of Unit 2 Fire Door 269

a. Inspection Scope

The inspector performed an in-office examination of Unresolved Item 50-368/00-08-01 to determine the risk significance of the condition using the significance determination process.

b. Issues and Findings

Brief Overview

The inspector identified that the inoperability of Fire Door 269 was an apparent violation of the licensee's fire protection program, as required by ANO Unit 2 Operating License Condition 2.C.(3)(b). The NRC staff determined that this issue was within the increased regulatory response band based on fire-induced transient exposure time and credit for operator recovery of the turbine-driven emergency feedwater pump.

Background

At ANO Unit 2, Fire Door 269 is a 3-hour rated fire barrier consisting of a double leaf door. The active leaf of the door provided normal ingress and egress and the inactive leaf is normally latched with two internal latches at the top and bottom of the leaf. Both leafs had door closure mechanisms and the active leaf used a standard door knob/throw mechanism to latch into the inactive leaf. The acceptable tested configuration for this 3-hour rated fire barrier was with the inactive leaf closed with both latches engaged, and the active leaf closed with its latch engaged in the inactive leaf.

Fire Door 269 separates the north and south vital switchgear rooms and opens from the south room into the north room. The north vital switchgear room contains Train A vital electrical equipment including: (1) 4.16-kV engineered safety features Bus 2A3, (2) 480-volt engineered safety features Load Center 2B5, (3) and 480-volt engineered safety features Motor Control Center 2B54. The south vital switchgear room contains Train B vital electrical equipment including: (1) 4.16-kV engineered safety features Bus 2A4, (2) 480-volt engineered safety features Load Center 2B6, and (3) 480-volt engineered safety features Motor Control Center 2B64. With Fire Door 269 open, there was less than 10 feet of separation between Buses 2A3 and 2A4.

Neither the north nor the south vital switchgear rooms had an automatic suppression system; however, both rooms had ionization type smoke detectors that alarmed in the control room.

On June 26, 2000, the inspector identified that the internal latches in the inactive leaf of Fire Door 269 were not latched and that a slight push on either leaf would open both doors. The inspector reported the finding to control room personnel and the condition was promptly corrected. The licensee initiated Condition Report 2-2000-225 to enter the issue in the corrective action program.

During the licensee's investigation of the issue, it was determined that security personnel had conducted a security drill in the area on June 22, 2000, and had propped the doors open to facilitate their drill. The investigation also determined that two operators toured the area approximately 20 minutes after the doors had been propped open and identified the unusual condition. The operators removed the device that propped the doors open but apparently did not close the latches in the inactive leaf. The licensee initiated Condition Report 2-2000-411 to document this issue in the corrective action program.

Based on the above, the inspector concluded that Fire Door 269 was inoperable for approximately 3.5 days, including a period of approximately 20 minutes when it was propped open in an unattended condition.

<u>Risk Determination</u>

The inspector reviewed this issue with the assistance of an NRC senior reactor analyst using NRC Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the April 21, 2000, revision. The inspector also referred to the licensee's Fire Hazards Analysis, Revision 6; ANO Unit 2 PreFire Plan, Revision 0; and Individual Plant Examination of External Events (IPEEE), dated May 1996.

Using the fire protection risk significance screening methodology, the inspector determined that the issue required a Phase 2 evaluation because the inoperable fire door was an affected defense-in-depth element that provided a 3-hour fire barrier separation between redundant trains of safe shutdown equipment. The north and south vital switchgear rooms contained the offsite and emergency power supply switchgear and power distribution switchgear to both trains of engineered safety features equipment. A loss of all equipment in these rooms would result in the loss of power to all motor-driven engineered safety features equipment. According to the licensee's prefire plan, a fire in these rooms could prevent the main feedwater system from operating and could also result in the loss of motor-operated emergency feedwater supply and discharge valves and atmospheric dump valves due to cable damage.

The inspector limited consideration of fire ignition to the south vital switchgear room. This was because Fire Door 269 opened from the south switchgear room into the north switchgear room, and it was assumed that, for the significant portion of time under consideration, the door remained an effective fire barrier from the north to the south switchgear room. Using guidance provided by the NRC Office of Nuclear Regulation, the inspector developed a fire scenario that assumed that: (1) plant electrical equipment (e.g., motor control centers, switchgear, relay panels, termination cabinets, motors, motor-generator sets, transformers) can be a source of ignition and part of the fuel load; (2) fires in electrical cabinets that have ventilation openings or are unsealed at the top can expose and ignite cables above the cabinets; (3) faults in high- and medium-voltage switchgear can breach a metal cabinet and cause faults in adjacent switchgear; (4) hydrocarbon fuels, when burning, can give off dense smoke within a short period of time (fill room or rooms from floor to ceiling with smoke) and smoke transport may impact fire brigade and operator actions; and (5) unprotected (no fire resistive barrier) safe shutdown equipment and recovery equipment/components that are in the fire's plume or located in the ceiling region are damaged. Additionally, the inspector used guidance for fire barriers to evaluate the condition of Fire Door 269. Examples of observed conditions that may represent a highly degraded fire barrier included: (1) fire barrier system design that was misapplied or had an indeterminate fire resistive rating; (2) an inoperable fire door in a fire barrier wall; and (3) a blocked open fire door.

The licensee's IPEEE identified that the fire ignition frequency in the south vital switchgear room was 1.09E-3/year. The fire hazards analysis identified that the fire duration in the room was moderate and that combustibles consisted of flame resistant cable insulation and transients.

Based on the above, the inspector considered a possible fire scenario involving a fire originating in the south vital switchgear room that damaged the safe shutdown equipment located in that room. Also, fire extension occurred into the north vital switchgear room through the inoperable Fire Door 269, damaging redundant safe shutdown equipment.

Potential Mitigation Strategies

As stated previously, each switchgear room had ionization-type smoke detectors that were operable during this time period and provided an alarm in the control room. An evaluation of the design adequacy of the fire detection system was not performed as part of this review.

The licensee's fire brigade consisted of a 5-member team, comprised of operations and security personnel trained in firefighting. The prefire plan provided the guidelines for fire brigade member fire attack for the south vital switchgear room. The guidelines included: (1) entry by two brigade members with fire extinguishers (unless fire was obviously beyond the extinguishers capability) to determine the extent of the fire and extinguish fire, if able; (2) a brigade member should extend a hoseline and standby at the staging area with a second brigade member standing by at the hose station to charge the hoseline and verify no hoseline obstructions; (3) if the brigade members who made entry were unable to extinguish the fire, they should return to the staging area for the hoseline, a brigade member should consider extending a secondary hoseline, and an additional member should provide reconnaissance of exterior zone boundaries for fire extension; and (4) if the fire was not controlled at this point, additional manpower and resources may be necessary. If fire were extinguished prior to extension into the north vital switchgear room, then a redundant train of safe shutdown equipment would be available to achieve and maintain a safe shutdown condition.

If fire damaged equipment in both vital switchgear rooms, no remaining mitigation capability would be available; however, the turbine-driven emergency feedwater pump would be a recoverable failed train. Operation of this pump and necessary valves would require manual operator action under a high stress condition.

Significance Determination Process

Using Phase 2 of the fire protection risk significance screening methodology, the inspector calculated the fire mitigation frequency (FMF) as follows:

FMF=IF+FB+MS+AS+CC

where IF = Fire Ignition Frequency
FB = Fire Barrier Degradation
MS = Manual Suppression
AS = Automatic Suppression
CC = Dependencies/Common Cause

AS and CC were not applicable because there was no automatic fire suppression system. MS was assigned a value of -1 (normal operating state) based on the operability of the fire detection system and standard readiness of the licensee's fire brigade. FB was 0 based on the previously addressed assumption that the barrier was highly degraded. Therefore, only the "double room term" addressed in Appendix F was applicable. From the IPEEE, IF=1.09E-3/year and log(IF)= -2.96. No credit was given in the determination of FB for fire brigade member recovery of the degraded fire barrier. Fire brigade activities were accounted for in the MS term.

Therefore, FMF= -3.96, which resulted in an approximate frequency in Table 5.7 of Appendix F of 1 per 1000 to 10,000 years. Using this approximate frequency and an exposure time of 3.5 days resulted in an estimated likelihood rating of E in Table 5.8.

The inspector and the senior reactor analyst used the "Site-Specific Worksheets for Use in the Nuclear Regulatory Commission's Significance Determination Process," dated January 24, 2000, and referred to the transients worksheet for ANO Unit 2. For each of the three sequences identified, credit of 1 was given for recovery of the turbine-driven emergency feedwater pump with no remaining mitigation capability.

Table 5.8 of the significance determination process indicates, for an estimated likelihood rating of E and with recovery of a failed train, the result is "white next to yellow" for each of the three sequences. The accounting rules of the significance determination process indicated that this result was a degraded cornerstone (yellow). However, because the duration of the condition was known to be approximately 3.5 days, this resulted in an estimated likelihood rating more appropriately categorized as F. Using Table 5.8 resulted in "green next to white" for each of the sequences. This indicates that the result is within the increased regulatory response band (white). The assumptions used by the inspectors and senior reactor analyst, and the results of the significance determination process, were discussed with licensee personnel.

Requirements

ANO Unit 2 Operating License Condition 2.C.(3)(b) requires that the licensee implement and maintain in effect all provisions of the approved fire protection program, as described in Amendment 9A to the safety analysis report and as approved in the safety evaluation dated March 31, 1992.

Safety Analysis Report Section 9D.5, "Fire Barriers," required that all fire barriers separating safety-related fire areas or separating portions of redundant safe shutdown systems required in the event of a fire shall be fully operable.

Procedure 1000.152, “Unit 1 and Unit 2 Fire Protection System Specifications,” Revision 3, identified Fire Door 269 as a door required to be operable.

Contrary to the above, Fire Door 269 was inoperable from approximately June 22-26, 2000. This is an apparent violation of ANO Unit 2 License Condition 2.C.(3)(b) (APV 50-368/00-16-01).

Corrective Actions to Date

The licensee implemented a practice of performing more frequent checks of the condition of risk-significant fire doors. The licensee also initiated a root cause investigation and informed the inspector that this investigation would address the licensee's past practice of not initiating fire protection impairment permits for fire doors that are made inoperable during drills.

Summary

The inspector identified an apparent violation of the licensee's fire protection program, as required by ANO Unit 2 Operating License Condition 2.C.(3)(b) for Fire Door 269 being inoperable approximately 3.5 days. The inoperable fire door resulted in the potential for redundant trains of safe shutdown equipment being unavailable in the event of a fire, leaving no mitigation capability. Recovery of the failed train of turbine-driven emergency feedwater was credited. Based on Tables 5.7 and 5.8 of the Fire Protection Significance Determination Process, the NRC staff determined that the screening for a fire-induced transient was within the increased regulatory response band (white), low frequency and low likelihood (F) with recovery of one failed train.

4. OTHER ACTIVITIES

4OA6 Management Meetings

.1 Exit Meeting Summary

On September 15, 2000, the inspector conducted a meeting with Mr. C. Anderson, and other members of plant management and presented the inspection results. The managers acknowledged the findings presented and discussed with the inspector the

NRC's significance determination process and enforcement review panel procedures. This included the NRC's process of obtaining licensee perspectives on initial characterization of a finding's significance, the finalization of the NRC's significance determination, and the significance determination process appeal process. No proprietary material was examined during the inspection.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. Anderson, Vice President, Operations
G. Ashley, Technical Assistant to the Vice President
R. Bement, General Manager, Plant Operations
M. Chisum, Manager, Unit 2 System Engineering
M. Cooper, Licensing Specialist
J. Hoffpauir, Plant Manager, Unit 2
B. James, Manager, Maintenance
D. James, Licensing Manager
R. Lane, Director, Engineering
T. Mitchell, Manager, Unit 2 Operations
S. Pyle, Licensing Specialist
J. Smith, Manager, Radiation Protection
M. Smith, Engineering Programs and Components Manager
C. Turk, Manager, Design Engineering
C. Tyrone, Manager, Quality Assurance
J. Vandergrift, Director, Nuclear Safety
H. Williams, Manager, Security

ITEMS OPENED AND CLOSED

Opened

APV	50-368/00-16-01	Apparent violation involving the inoperability of Unit 2 Fire Door No. 269 (Section 1R05)

Closed

URI	50-368/00-08-01	Inoperability of Unit 2 Fire Door No. 269 (Section 1R05)

DOCUMENTS REVIEWED

Procedure 1000.152	Unit 1 and Unit 2 Fire Protection System Specifications	Revision 3
Condition Report ANO-2-2000-225	Fire Door No. 269 found not latched	June 26, 2000
Condition Report ANO-2-2000-411	Investigation identified Fire Door No. 269 propped open	September 12, 2000
Unit 2 Pre Fire Strategies		Revision 0
ANO Fire Hazards Analysis		Revision 6

Summary Report of Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities for ANO-2		May 1996

ATTACHMENT 2

NRC'S REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
•Initiating Events	•Occupational	•Physical Protection
•Mitigating Systems	•Public	
•Barrier Integrity		
•Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC used two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the significance determination process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, or RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http:\\www.nrc.gov\NRR\OVERSIGHT\index.html.