UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

August 3, 2007

Mr. Timothy G. Mitchell
Vice President Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802-0967

SUBJECT: ARKANSAS NUCLEAR ONE - NRC INTEGRATED INSPECTION REPORT 05000313/2007003 AND 05000368/2007003

Dear Mr. Mitchell:

On June 23, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Arkansas Nuclear One, Units 1 and 2, facility. The enclosed integrated report documents the inspection findings, which were discussed on June 26, 2007, with you and other members of your staff.

The 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two NRC identified findings and five self-revealing findings of very low safety significance (Green). Six of these findings were determined to involve violations of NRC requirements. Additionally, two licensee-identified violations which were determined to be of very low safety significance are listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest these noncited violations, 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 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Arkansas Nuclear One, Units 1 and 2, facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/ GDReplogle for

Jeff Clark, P.E.
Chief, Branch E
Division of Reactor Projects

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

Enclosure:
NRC Inspection Report 05000313/2007003 and 05000368/2007003
w/Attachment: Supplemental Information

cc w/Enclosure:

Senior Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P.O. Box 31995
Jackson, MS 39286-1995

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

General Manager Plant Operations
Entergy Operations, Inc.
Arkansas Nuclear One
1448 S. R. 333
Russellville, AR 72802

Director, Nuclear Safety Assurance
Entergy Operations, Inc.
Arkansas Nuclear One
1448 S. R. 333
Russellville, AR 72802

Manager, Licensing
Entergy Operations, Inc.
Arkansas Nuclear One
1448 S. R. 333
Russellville, AR 72802

Director, Nuclear Safety & Licensing
Entergy Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213-8298

Section Chief, Division of Health
Radiation Control Section
Arkansas Department of Health and
Human Services
4815 West Markham Street, Slot 30
Little Rock, AR 72205-3867

Section Chief, Division of Health
Emergency Management Section
Arkansas Department of Health and
Human Services
4815 West Markham Street, Slot 30
Little Rock, AR 72205-3867

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

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

James Mallay
Director, Regulatory Affairs
Framatome ANP
3815 Old Forest Road
Lynchburg, VA 24501

Lisa R. Hammond, Chief
Technological Hazards Branch
National Preparedness Division
FEMA Region VI
800 N. Loop 288
Denton, TX 76209

Electronic distribution by RIV:
Regional Administrator **(BSM1)**
DRP Director **(ATH)**
DRS Director **(DDC)**
DRS Deputy Director **(WBJ)**
DRS Deputy Director **(RJC1)**
Senior Resident Inspector **(RWD)**
Branch Chief, DRP/E **(JAC)**
Senior Project Engineer, DRP/E **(GDR)**
Team Leader, DRP/TSS **(CJP)**
RITS Coordinator **(MSH3)**

Only inspection reports to the following:
DRS STA **(DAP)**
M. Kunowski, OEDO RIV Coordinator **(MAK3)**
D. Pelton, OEDO RIV Coordinator **(DLP)**
ROPreports
ANO Site Secretary **(VLH)**

SUNSI Review Completed: _GDR_____ ADAMS: √ Yes ☐ No Initials: _GDR__
√ Publicly Available ☐ Non-Publicly Available ☐ Sensitive √ Non-Sensitive

R:_REACTORS_ANO\2007\AN2007-03RP-RWD.wpd

RIV:RI:DRP/E	RI:DRP/E	SRI:DRP/E	SPE:DRP/E	C:DRS/OB
CHYoung	JEJosey	RWDeese	JDHanna	ATGody
JEJosey for	E-JDHanna	E-MJSpivey	/RA/	TFStetka for
8/2/07	8/2/07	7/31/07	7/24/07	7/22/07

C:DRS/PSB	C:DRS/EB1	C:DRS/EB2	C:DRP/E
MPShannon	DAPowers	LJSmith	JAClark
BTLarson for	JRLarsen for	/RA/	GDReplogle
7/27/07	7/27/07	7/25/07	8/3/07

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-313, 50-368
Licenses:	DPR-51, NPF-6
Report:	05000313/2007003 and 05000368/2007003
Licensee:	Entergy Operations, Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	Junction of Hwy. 64W and Hwy. 333 South Russellville, Arkansas
Dates:	March 25 through June 23, 2007
Inspectors:	B. Baca, Health Physicist, Plant Support Branch R. Deese, Senior Resident Inspector J. Groom, Project Engineer, Project Branch E J. Josey, Acting Senior Resident Inspector S. Rutenkroger, PhD, Reactor Inspector C. Young, P.E., Resident Inspector
Approved By:	Jeff A. Clark, P.E., Chief, Project Branch E Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS . . . . -1-

REACTOR SAFETY . . . . -1-
1R01 Adverse Weather Protection . . . . -1-
1R04 Equipment Alignment . . . . -2-
1R05 Fire Protection . . . . -2-
1R06 Flood Protection Measures . . . . -3-
1R08 Inservice Inspection Activities . . . . -4-
1R11 Licensed Operator Requalification Program . . . . -9-
1R12 Maintenance Effectiveness . . . . -9-
1R13 Maintenance Risk Assessments and Emergent Work Control . . . . -9-
1R15 Operability Evaluations . . . . -10-
1R19 Postmaintenance Testing . . . . -11-
1R20 Refueling and Outage Activities . . . . -12-
1R22 Surveillance Testing . . . . -12-
1R23 Temporary Plant Modifications . . . . -14-
1EP6 Drill Evaluation . . . . -15-

RADIATION SAFETY . . . . -15-
2OS1 Access Control to Radiologically Significant Areas . . . . -15-
2OS2 ALARA Planning and Controls . . . . -18-

OTHER ACTIVITIES . . . . -19-
4OA1 Performance Indicator Verification . . . . -19-
4OA2 Identification and Resolution of Problems . . . . -21-
4OA3 Event Follow-up . . . . -24-
4OA5 Other Activities . . . . -28-
4OA6 Meetings, Including Exit . . . . -28-
4OA7 Licensee-Identified Violations . . . . -29-

SUPPLEMENTAL INFORMATION A-1

KEY POINTS OF CONTACT A-1

LIST OF ITEMS OPENED AND CLOSED A-2

LIST OF DOCUMENTS REVIEWED A-2

LIST OF ACRONYMS A-17

SUMMARY OF FINDINGS

IR 05000313/2007003, 05000368/2007003; 03/25/07 - 06/23/07; Arkansas Nuclear One, Units 1 and 2; Surveillance Testing, Access Control to Radiologically Significant Areas, Identification and Resolution of Problems, and Follow-up of Events and Notices of Enforcement Discretion.

This report covered a 3-month period of inspection by resident inspectors and regional specialist inspectors. The inspection identified seven Green findings, six of which were noncited violations. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after Nuclear Regulatory Commission management's review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing noncited violation of Unit 2 Technical Specification 6.4.1.c, "Fire Protection Program Implementation," was identified for the licensee's failure to provide training and qualification for fire protection designees which resulted in non-routine hot work activities not being adequately evaluated by appropriately trained individuals. Specifically, the roofing contractor working on the auxiliary building roof required that a 2-hour fire watch was to be stationed following roofing activities involving the use of open flames, but the licensee only required the fire watch be stationed for 30 minutes. As a result, on June 7, 2007, following roofing activities on the auxiliary building roof above the spent fuel floor that involved the use of open flames, two fires occurred after approximately one hour from the completion of hot work activities, and there was not an appropriately trained fire watch in the area. This issue was entered into the licensee's corrective action program as Condition Reports ANO-2-2007-0816 and ANO-2-2007-0839.

  The finding was determined to be more than minor because it affected the protection against external factors attribute of the initiating events cornerstone, and it directly affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Additionally, if left uncorrected, the practice of not adequately evaluating nonroutine hot work activities by appropriately trained individuals would become a more significant safety concern in that it could result in a fire in or near other risk important equipment. Using the Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," Phase 1 Worksheet, the finding was determined to have very low safety significance because the condition did not constitute a high degradation of a fire prevention and administrative controls feature. The finding

had crosscutting aspects in the area of human performance associated with decision making (H.1(b)) because the licensee did not use conservative assumptions and failed to verify the validity of the underlying assumptions (Section 4OA3.1).

- Green. A self-revealing finding was identified when Unit 2 experienced a complete loss of component cooling water flow due to the loss of the Train B component cooling water Pump 2P-33B on February 21, 2007. Specifically, the loss of component cooling water occurred when an operator was attempting to pressurize an out-of- service heat exchanger to support maintenance activities. This issue was entered into the licensee's corrective action program as Condition Report ANO-2-2007-0313.

  The finding was determined to be more than minor because it affected the equipment performance attribute of the initiating events and mitigating systems cornerstones. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required.

  The inspectors performed a Phase 2 analysis using Appendix A, "Technical Basis For At Power Significance Determination Process," of Manual Chapter 0609, "Significance Determination Process," and the Phase 2 Worksheets for Arkansas Nuclear One. The inspectors assumed that the duration of the component cooling water system unavailability was very short, approximately 4 hours. Additionally, the inspectors assumed that only the power conversion system was affected and all other mitigating systems were available. Based on the results of the Phase 2 analysis, the finding was determined to have very low safety significance. The finding had crosscutting aspects in the area of human performance associated with resources (H.2(b)) because the training of personnel and procedural guidance available was inadequate (Section 4OA2).

- Green. A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified associated with small flash fires that occurred on the Unit 2 Emergency Diesel Generator 2K-4A on April 15, 2007. Specifically, the licensee failed to verify that the outer protective cover for insulation used on the exhaust manifold was rated for expected temperatures. This issue was entered into the licensee's corrective action program as Condition Report ANO-2-2007-0630.

  The finding was determined to be more than minor because it affected the protection against external factors attribute of the initiating events cornerstone, and it directly affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using the Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process,” Phase 1 Worksheet, the finding was determined to have very low safety significance because the condition did not constitute a high degradation of a fire prevention and administrative controls feature. The finding had crosscutting aspects in the

area of human performance associated with work practices (H.4(a)) because the licensee personnel proceeded with work in the face of uncertainty (Section 4OA3.3).

- <u>Green</u>. A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” was identified associated with the exhaust manifold lagging fire that occurred on Unit 2 Emergency Diesel Generator 2K-4A on May 11, 2007. Specifically, the licensee failed to adequately implement corrective actions from a previous diesel exhaust manifold fire in 2003 and as such, the licensee failed to identify and correct an oil leak from the front cover of the diesel which resulted in a fire during a monthly surveillance run. This issue was entered into the licensee's corrective action program as Condition Report ANO-2-2007-0718.

  The finding was determined to be more than minor because it affected the protection against external factors attribute of the initiating events cornerstone, and it directly affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using the Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process,” Phase 1 Worksheet, the finding was determined to have very low safety significance because the condition did not constitute a high degradation of a fire prevention and administrative controls feature (Section 4OA3.2).

Cornerstone: Barrier Integrity

- <u>Green</u>. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the licensee’s failure to prompty identify and correct a practice of unacceptable preconditioning prior to American Society of Mechanical Engineers Code inservice testing of the Unit 1 reactor building spray pumps. The licensee’s corrective action program (via Condition Report ANO-C-1997-0288) failed to identify and correct the practice of venting the reactor building spray pump casing prior to conducting the quarterly surveillance test, which continued from 1997 through 2007. This issue was entered into the licensee's corrective action program as Condition Report ANO-1-2007-1645.

  The finding was determined to be more than minor because it affected the procedure quality attribute of the barrier integrity cornerstone, and affected the associated cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radio nuclide releases caused by accidents or events. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding was determined to have very low safety significance because it did not involve an actual reduction in defense-in-depth for the atmospheric pressure control function of the reactor containment (Section 1R22).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a noncited violation of 10 CFR 20.1902(a) because the licensee failed to conspicuously post a radiation area. On May 2, 2007, during a tour of the auxiliary building, the inspectors observed that the radiological posting to the entryway of the Unit 1 Decay Heat Vault B was not conspicuously posted. An operations' "protected train" sign obscured the radiological posting. The licensee's immediate corrective action was to re-post the operations' sign to prevent obscuring the radiological posting.

  The finding was greater than minor because it was associated with the occupational radiation safety cornerstone attribute of program and process and affected the cornerstone objective to ensure the adequate protection of a worker's health and safety from exposure to radiation because it could have resulted in workers being exposed to higher radiation levels. When processed through the occupational radiation safety significance determination process, the finding was determined to be of very low safety significance because it was not an as low as reasonably achievable finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. In addition, this finding had a crosscutting aspect associated with the human performance component of work practices (H.4(a)) because personnel failed to use human error prevention techniques such as self-checking or peer checking to verify that the radiation area was conspicuously posted (Section 2OS1).

- Green. The inspectors reviewed a self-revealing noncited violation of Technical Specification 5.4.1.a. because of a failure to use an engineering control as required by a radiation work permit. On April 25, 2007, four workers were unable to clear the personnel contamination monitors after working near the Unit 1 Steam Generator A. The licensee conducted an investigation and determined the steam generator high-efficiency particulate air ventilation (a radiation work permit required engineering control) had been rendered inoperable due to an incorrect line-up. The licensee's immediate corrective actions were to counsel the workers and brief associated personnel on the correct method for verifying high-efficiency particulate air ventilation.

  The finding was greater than minor because it was associated with the occupational radiation safety cornerstone attribute of program and process and affected the cornerstone objective to ensure the adequate protection of a worker's health and safety from exposure to radiation because it resulted in workers being exposed to higher radiation levels. When processed through the occupational radiation safety significance determination process, the finding was determined to be of very low safety significance because it was not an as low as reasonable achievable finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. In addition, this finding had a crosscutting aspect associated with the human performance component of resources (H.2(c)) because the high-efficiency particulate air ventilation verification procedure was not adequate in that it did not have sufficient detail (Section 2OS1).

B. Licensee-Identified Violations

Violations of very low safety significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and their corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at 100 percent rated thermal power (RTP) and operated there until April 10, 2007, when the unit began a coastdown until April 22 when the unit shutdown for Refueling Outage 1R20. The reactor achieved criticality on May 12, the main generator output breakers were closed on May 13 and the plant achieved approximately 100 percent RTP on May 17. Reactor power was lowered to 10 percent RTP on May 25, and the main generator output breakers were opened and the main turbine was tripped to facilitate repairs of the C phase main transformer. The main generator output breakers were closed on May 31 and the plant achieved approximately 100 percent RTP on June 1. The unit remained at 100 percent RTP for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent RTP. Reactor power was lowered to 80 percent RTP on April 28, 2007, at the direction of the load dispatcher and returned to 100 percent RTP on May 1. Reactor power was lowered to 84 percent RTP on May 6, at the direction of the load dispatcher, and returned to 100 percent RTP on May 7. The unit remained at 100 percent RTP for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Seasonal Susceptibilities

a. Inspection Scope

The inspectors completed a review of the licensee's readiness for impending adverse weather involving severe thunderstorms. The inspectors: (1) reviewed plant procedures, the Updated Final Safety Analysis Report (UFSAR), and Technical Specifications (TSs) to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the below listed two systems to ensure that adverse weather protection features (heat tracing, space heaters, weatherized enclosures, temporary chillers) were sufficient to support operability, including the ability to perform safe shutdown functions; (3) reviewed maintenance records to determine that applicable surveillance requirements were current before the anticipated severe thunderstorms developed; and (4) reviewed plant modifications, procedure revisions, and operator work arounds to determine if recent facility changes challenged plant operation.

- June 8, 2007, Units 1 and 2, Emergency Feedwater and 480 Volt AC Safety-Related Electrical Distribution System

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

The inspectors: (1) walked down portions of the three below listed risk important systems and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned, and (2) compared deficiences identified during the walkdown to the licensee's UFSAR and Corrective Action Program (CAP) to ensure problems were being identified and corrected.

- April 26, 2007, Unit 1, Decay Heat Removal Trains A and B
- April 27, 2007, Unit 1, Spent Fuel Pool Cooling
- May 16, 2007, Unit 1, Emergency Feedwater Pump P-7B

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors walked down the six below listed plant areas to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the

compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the UFSAR to determine if the licensee identified and corrected fire protection problems.

- March 28, 2007, Unit 1, Fire Zone 34-Y, North Safeguard Pipeway
- April 30, 2007, Unit 1, Fire Zone 20-Y, Radiological Waste Processing Room
- June 14, 2007, Unit 2, Fire Zone 2091-BB, North Electrical Equipment Room
- June 20, 2007, Unit 1, Fire Zone 149-E, Upper North Electrical Penetration Room
- June 21, 2007, Unit 2, Fire Zone 2063-DD, Sample Room
- June 22, 2007, Unit 1, Fire Zone 97-R, Cable Spreading Room

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Inspection

a. Inspection Scope

On April 16, 2007, the inspectors observed a fire brigade drill to evaluate the readiness of licensee personnel to prevent and fight fires, including the following aspects: (1) the number of personnel assigned to the fire brigade, (2) use of protective clothing, (3) use of breathing apparatuses, (4) use of fire procedures and declarations of emergency action levels, (5) command of the fire brigade, (6) implementation of prefire strategies and briefs, (7) access routes to the fire and the timeliness of the fire brigade response, (8) establishment of communications, (9) effectiveness of radio communications, (10) placement and use of fire hoses, (11) entry into the fire area, (12) use of firefighting equipment, (13) searches for fire victims and fire propagation, (14) smoke removal, (15) use of prefire plans, (16) adherence to the drill scenario, (17) performance of the postdrill critique, and (18) restoration from the fire drill. The licensee simulated a fire in the oil warehouse near the southwest corner of the switchyeard.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

.1 Semi-annual Internal Flooding

a. Inspection Scope

The inspectors: (1) reviewed the UFSAR, the flooding analysis, and plant procedures to assess seasonal susceptibilities involving external flooding; (2) reviewed the UFSAR and CAP to determine if the licensee identified and corrected flooding problems; (3) inspected underground bunkers/manholes to verify the adequacy of (a) sump pumps, (b) level alarm circuits, (c) cable splices subject to submergence, and (d) drainage for bunkers/manholes; (4) verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and (5) walked down the area listed below to verify the adequacy of: (a) equipment seals located below the floodline, (b) floor and wall penetration seals, (c) watertight door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms and control circuits, and (f) temporary or removable flood barriers. Specifically, the inspectors reviewed:

- June 20, 2007, Unit 2, Emergency Safety Features Train B

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

Inspection Procedure 71111.08 requires four samples as identified in Sections 02.01, 02.02, 02.03, and 02.04.

.1 Performance of Nondestructive Examination (NDE) Activities Other than Steam Generator Tube Inspections, Pressurized Water Reactor (PWR) Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control

a. Inspection Scope

The inspection procedure required the review of NDE activities consisting of two or three different types (i.e., volumetric, surface, or visual). The inspectors observed the performance of three liquid penetrant examinations (surface) and two ultrasonic examinations (volumetric) on the preemptive weld overlays performed on the dissimilar metal welds on nozzles of the pressurizer. The inspectors also reviewed three liquid penetrant examinations and four ultrasonic examinations on the remaining weld overlays performed on the pressurizer. (The examinations/welds are identified in the NDE section of the documents reviewed attachment to this report.)

For each of the observed NDE activities, the inspectors verified that the examinations were performed in accordance with the specific site procedures and the applicable American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirements.

During review of each examination, the inspectors verified that appropriate NDE procedures were used, examinations and conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also verified the NDE certifications of the personnel who performed the above examinations. Finally, the inspectors verified that indications identified during the examinations were dispositioned in accordance with the ASME Code-qualified NDE procedures used to perform the examinations.

The inspection procedure required review of one or two examinations with recordable indications that were accepted for continued service to ensure that the disposition was made in accordance with the ASME Code. The inspectors verified that recordable indications revealed in ultrasonic examination of the weld overlay of Nozzle PSV-1001 of the pressurizer were dispositioned in accordance with the ASME Code, including those accepted for continued service.

The inspection procedure further required verification of one to three welds on Class 1 or 2 pressure boundary piping to ensure that the welding process and welding examinations were performed in accordance with the ASME Code. The inspectors observed the welding performed for the six weld overlays on the nozzle-to-piping connections of the pressurizer. The inspectors verified that the welding was performed in accordance with the Safety Evaluation Report granting the licensee's Relief Request, and as referenced therein, Sections IX and XI of the ASME Code. This included review of welding material issue slips to establish that the appropriate welding materials had been used and verification that the welding procedure specifications had been properly qualified.

The inspectors completed the one sample required by Section 02.01.

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Upper Head Penetration Inspection Activities

The inspection requirements for this section paralleled the inspection requirement steps in Section 02.01. The licensee did not perform any of these activities during this refueling outage.

a. Inspection Scope

This sample was not completed because there was no activity to observe.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Inspection Activities (PWRs)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspection procedure requires review of a sample of boric acid corrosion control walkdown visual examination activities through either direct observation or record review. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown. Additionally, the inspectors performed independent observations of piping containing boric acid during walkdowns of the containment building and the auxiliary building and discussed the program's implementation with the licensee's program owner.

The inspection procedure required verification that visual inspections emphasized locations where boric acid leaks could cause degradation of safety significant components. The inspectors verified through direct observation and program/record reviews that the licensee's boric acid corrosion control inspection efforts were directed towards locations where boric acid leaks could cause degradation of safety-related components.

The inspection procedure required both a review of one to three engineering evaluations performed for boric acid leaks found on reactor coolant system piping and components and one to three corrective actions performed for identified boric acid leaks. The inspectors reviewed four evaluations to assess the licensee's analysis and evaluate the assessment of the condition and proposed corrective actions.

The inspectors completed the one sample required by Section 02.03.

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspection procedure specified performance of an assessment of in situ screening criteria to assure consistency between assumed NDE flaw sizing accuracy and data from the Electric Power Research Institute (EPRI) examination technique specification sheets. It further specified assessment of appropriateness of tubes selected for in situ pressure testing, observation of in situ pressure testing, and review of in situ pressure test results.

At the time of this inspection, no conditions had been identified that warranted in situ pressure testing. The inspectors did, however, review the licensee's degradation assessment report, "Steam Generator Pre-Outage Degradation Assessment and Repair Criteria for 1R20," Revision 1, and compared the in situ test screening parameters to the EPRI guidelines. The inspectors determined that the screening parameters were consistent with the EPRI guidelines.

In addition, the inspectors reviewed both the licensee site-validated and qualified acquisition and analysis technique sheets used during this refueling outage and the qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy, tubing, equipment, technique, and analysis had been identified and qualified through demonstration. The inspector-reviewed acquisition technique and analysis technique sheets are identified in the documents reviewed attachment.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess the licensee's prediction capability. The inspectors compared the previous outage operational assessment predictions with the flaws identified during the current steam generator tube inspection effort. Compared to the projected damage mechanisms identified by the licensee, the number of identified indications fell within the range of prediction and were quite consistent with those predictions.

However, an unusual pattern of wear was identified at the eighth tube support plate in Steam Generator A, and the eddy current examinations also identified bowing in seven stay rods, apparently being in contact with the adjacent tubes. Some of the affected tubes also appeared to have secondary, and in one case tertiary, tube-to-tube contact. The licensee made specific commitments to the NRC regarding this issue in a commitment letter. However, since no tube damage was evident from the identified condition, no new damage mechanisms were identified during this inspection.

The inspection procedure specified confirmation that the steam generator tube eddy current test scope and expansion criteria met Technical Specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors evaluated the recommended steam generator tube eddy current test scope established by Technical Specification requirements and the degradation assessment report. The inspectors compared the recommended test scope to the actual test scope and found that the licensee had accounted for all known flaws and had, as a minimum, established a test scope that met Technical Specfication requirements, EPRI guidelines, and commitments made to the NRC.

The inspection procedure specified, if new degradation mechanisms were identified, verification that the licensee fully enveloped the problem in its analysis of extended conditions including operating concerns and had taken appropriate corrective actions before plant startup. As discussed above, the eddy current test results did not identify any new degradation mechanisms in the form of tube wear. However, due to the bowing of the seven stay rods and unusual pattern of wear indications at the eighth tube

support plate, the licensee made appropriate written commitments to the NRC regarding monitoring, providing further information to the NRC, and operational plans for Steam Generator A.

The inspection procedure required confirmation that the licensee inspected all areas of potential degradation, especially areas that were known to represent potential eddy current test challenges (e.g., top-of-tubesheet and tube support plates). The inspectors confirmed that all known areas of potential degradation were included in the scope of inspection and were being inspected.

The inspection procedure further required verification that repair processes being used were approved in the TSs. One tube in Steam Generator B was plugged. The inspectors verified that the mechanical expansion plugging process used was an NRC-approved repair process.

The inspection procedure also required confirmation of adherence to the TSs plugging limit, unless alternate repair criteria have been approved. The inspection procedure further required determination whether depth sizing repair criteria were being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections. The inspectors determined that the Technical Specification plugging limits were being adhered to (i.e., 40 percent maximum through-wall indication).

If steam generator leakage greater than three gallons per day was identified during operations or during post shutdown visual inspections of the tubesheet face, the inspection procedure required verification that the licensee had identified a reasonable cause based on inspection results and that corrective actions were taken or planned to address the cause for the leakage. The inspectors did not conduct any assessments because this condition did not exist.

The inspection procedure required confirmation that the eddy current test probes and equipment were qualified for the expected types of tube degradation and an assessment of the site-specific qualification of one or more techniques. The inspectors reviewed portions of eddy current tests performed on the tubes in Steam Generators A and B. The inspectors verified that: (1) the probes appropriate for identifying the expected types of indications were being used, (2) probe position location verification was performed, (3) calibration requirements were adhered to, and (4) probe travel speed was in accordance with procedural requirements. The inspectors performed a review of site-specific qualifications of the techniques being used. These are identified in the documents reviewed attachment.

If loose parts or foreign material on the secondary side were identified, the inspection procedure specified confirmation that the licensee had taken or planned appropriate repairs of affected steam generator tubes and that they inspected the secondary side to either remove the accessible foreign objects or perform an evaluation of the potential effects of inaccessible object migration and tube fretting damage. During licensee-performed foreign object search and retrieval inspections, one small metal shaving was identified and removed in Steam Generator B. Eddy current examination of tubes in the area of the loose part did not identify any tube damage.

Finally, the inspection procedure specified review of one to five samples of eddy current test data if questions arose regarding the adequacy of eddy current test data analyses. Although the inspectors did not identify any results where the adequacy of eddy current test data analysis was questionable, the inspectors reviewed eddy current test data for the plugged tube in Steam Generator B and two of the tubes affected by the bowed stay rods in Steam Generator A and verified the adequacy of the analysis.

The inspectors completed one sample under Section 02.04.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspection procedure required review of a sample of problems associated with inservice inspections documented by the licensee in the corrective action program for appropriateness of the corrective actions. For this sample the inspectors reviewed three condition reports, which dealt with inservice inspection and welding activities. From this review, the inspectors concluded that the licensee had an appropriate threshold for entering issues into the CAP and had procedures that directed a root cause evaluation when necessary. The licensee also had an effective program for applying industry operating experience.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On April 5, 2007, the inspectors observed testing and training of senior reactor operators and reactor operators in the Unit 2 simulator to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario involved the crew response to a reactor coolant system leak with a failed fuel assembly.

Documents reviewed by the inspectors included:

- Procedure A1SPGOR070401, “Unannounced Casualties”, Revision 0

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the two below listed maintenance activities to: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the Maintenance Rule, 10 CFR Part 50, Appendix B, and TSs.

- June 15, 2007, Unit 1, Emergency Feedwater Initiation Control

- June 22, 2007, Unit 2, Control Room Emergency Ventilation

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

.1 Risk Assessment and Management of Risk

a. Inspection Scope

The inspectors reviewed the five below listed assessment activities to verify: (1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognized, and/or entered as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) that the licensee identified and corrected problems related to maintenance risk assessments.

- April 19, 2007, Unit 1, Emergency Feedwater Pump P-7A Maintenance

- May 16, 2007, Unit 1, Emergency Feedwater Pump P-7A Steam Exhaust Modification

- June 4, 2007, Unit 1, Emergency Diesel Generator (EDG) A Maintenance

- June 12, 2007, Unit 1, Emergency Feedwater Pump P-7B Maintenance

- June 13, 2007, Unit 2, Alternate AC Diesel Generator Maintenance

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plant status documents, such as operator shift logs, emergent work documentation, deferred modifications, and standing orders, to determine if an operability evaluation was warranted for degraded components; (2) referred to the UFSAR and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TSs; (5) used the significance determination process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- March 10, 2007, Unit 1, Reactor Building Spray Pump P-35B
- April 9, 2007, Unit 2, EDG Oil Sump Temperature Switch 2K4B
- April 12, 2007, Unit 1, Control Room Emergency Ventilation Fan VSF-9
- April 23, 2007, Unit 2, Steam Leak on the A Steam Generator Level Detecting Piping
- April 24, 2007, Unit 1, Containment Sump Operability
- April 26, 2007, Unit 2, EDG A Jacket Water Keep Warm Pump
- May 2, 2007, Battery Chargers D-03A, D-04A, and D-04B Supply Breakers

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed seven samples.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the six below listed postmaintenance test activities of risk significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety

functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly realigned, and deficiencies during testing were documented. The inspectors also reviewed the UFSAR to determine if the licensee identified and corrected problems related to postmaintenance testing.

- April 13, 2007, Unit 1, Control Room Emergency Ventilation Fan Damper CV-7910
- April 23, 2007, Unit 1, Emergency Feedwater Pump P-7A
- May 26, 2007, Unit 1, Source Range Nuclear Instrument Channel NI-502
- May 30, 2007, Unit 1, Emergency Feedwater Initiation Control Channel A
- June 4, 2007, Unit 1, Temporary Modification to NI-5
- June 11, 2007, Unit 1, EDG A

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Unit 1 Forced Outage Caused by C Phase Main Transformer High Voltage Bushing Oil Leak

a. Inspection Scope

The inspectors reviewed the following risk significant outage activities to verify defense-in-depth commensurate with the outage risk control plan and compliance with the TSs: (1) the risk control plan, (2) tagging/clearance activities, (3) reactivity control, and (4) restart activities.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Refueling Outage 1R20

a. Inspection Scope

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense-in-depth commensurate with the outage risk control plan, compliance with the TSs, and adherence to commitments in response to Generic Letter 88-17, "Loss of Decay Heat Removal:" (1) the risk control plan, (2) tagging/clearance activities, (3) reactor coolant system instrumentation, (4) electrical power, (5) decay heat removal, (6) spent fuel pool cooling, (7) inventory control, (8) reactivity control, (9) containment closure, (10) reduced inventory conditions, (11) refueling activities, (12) heatup and cooldown activities, (13) restart activities; and (14) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities. The inspectors' containment inspections included observation of the containment sump for damage and debris, supports, braces, and snubbers for evidence of excessive stress, water hammer, or aging.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the UFSAR, procedure requirements, and TSs to ensure that the six below listed surveillance activities demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- March 10, 2007, Unit 1, Reactor Building Spray Pump P-35B
- March 27, 2007, Unit 2, Emergency Feedwater Pump 2P-75
- April 4, 2007, Unit 2, Core Protection Calculator Channel A
- April 11, 2007, Unit 2, Train A Refueling Water Tank Outlet Valve 2CV-5630-1
- June 11, 2007, Unit 1, Reactor Coolant System Leak Detection

- June 20, 2007, Unit 1 Pressurizer Hot Leg Sample Valve SV-1818 Local Leak Rate Test

Documents reviewed by the inspectors are listed in the attachment

The inspectors completed six samples.

b. Findings

Introduction. The inspectors identified a Green noncited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to prompty identify and correct a practice of unacceptable preconditioning prior to ASME Code inservice testing of the Unit 1 reactor building spray pumps.

Description. In April 1997, the NRC published Information Notice (IN) 97-16, "Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing of Technical Specification Surveillance Testing." This IN contained examples of unacceptable preconditioning activities, including the practice of venting the casing of safety related pumps immediately before performing surveillance tests. The IN also cited NUREG-1482 "Guidelines for Inservice Testing at Nuclear Power Plants." Section 3.5.4 of NUREG-1482 also specifically identifies that venting a pump prior to testing without proper controls is an example of unacceptable preconditioning. Proper controls included a technical evaluation to establish that the amount of gas vented would not have otherwise adversely affected pump operation.

The licensee initiated CR ANO-C-1997-0288 in September 1997 in order to evaluate the applicability of IN 97-16 to the licensee's current applicable procedures and practices. Corrective actions assigned from this CR failed to identify that the quarterly surveillance test procedure for the Unit 1 reactor building spray pumps, OP-1104.005 "Reactor Building Spray Pump Operation," Supplement 5, contained a requirement to vent the pump casing just prior to conducting the surveillance test, with no controls in place to limit the quantity of gas vented below an established maximum allowable amount. As a result, this preconditioning activity continued through June of 2007, when the deficiency in the surveillance test procedure was identified by the inspectors. The step requiring the venting of the pumps just prior to performing the surveillance test was incorporated into the procedure with Revision 2 in January of 1974.

Analysis. The performance deficiency associated with this finding involved the licensee's failure to identify that the practice of venting the reactor building spray pump casing prior to conducting the quarterly surveillance test constituted unacceptable preconditioning based on the information published in NRC Information Notice 97-16. The finding was determined to be more than minor because it affected the procedure quality attribute of the barrier integrity cornerstone, and affected the associated cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding was determined to have very low safety significance because it did not involve an actual reduction in defense-in-depth for the atmospheric pressure control function of the reactor containment.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this, between April 1997 and June 2007, the licensee failed to establish measures to assure that a condition adverse to quality was promptly identified and corrected. Specifically, the licensee’s evaluation of the applicability of IN 97-16 failed to promptly identify and correct a practice of unacceptable preconditioning prior to ASME Code inservice testing of the Unit 1 reactor building spray pumps. Because the finding is of very low safety significance and has been entered into the licensee’s CAP as CR ANO-1-2007-1645, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000313/2007003-01, “Failure to Identify the Preconditioning of Reactor Building Spray Pumps.”

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the UFSAR, plant drawings, procedure requirements, and TSs to ensure that the two below listed temporary modifications were properly implemented. The inspectors: (1) verified that the modification did not have an affect on system operability/availability, (2) verified that the installation was consistent with the modification documents, (3) ensured that the postinstallation test results were satisfactory and that the impact of the temporary modification on permanently installed SSC’s were supported by the test, (4) verified that the modifications were identified on control room drawings and that appropriate identification tags were placed on the affected drawings, and (5) verified that appropriate safety evaluations were completed. The inspectors verified that the licensee identified and implemented any needed corrective actions associated with temporary modifications.

- May 10, 2007, Unit 1, Temporary Alteration of P-32C Motor
- May 16, 2007, Unit 1, Temporary Modification to Disable NI-5 Channel A Output to Emergency Feedwater Initiation Control Channel A

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

For the below listed simulator-based training evolution contributing to drill/exercise performance, emergency response organization, and performance indicators (PIs), the inspectors: (1) observed the training evolution to identify any weaknesses and

deficiencies in classification, notification, and protective action requirements development activities; (2) compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee is properly identifying failures; and (3) determined whether licensee performance is in accordance with the guidance of the Nuclear Energy Institute (NEI) 99-02, “Voluntary Submission of Performance Indicator Data,” acceptance criteria.

- April 3, 2007, Unit 2, simulator-based exercise involving the declaration of a notice of unusual event

Documents reviewed by the inspectors included:

- Unit 2 Dynamic Exam Scenario SES-2-021, Revision 4

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess the licensee’s performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspectors used the requirements in 10 CFR Part 20, the TSs, and the licensee’s procedures required by TSs as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the occupational radiation safety cornerstone

- Controls (surveys, posting, and barricades) of radiation, high radiation, or airborne radioactivity areas in the reactor, auxiliary, and spent fuel pool buildings

- Radiation work permits, procedures, engineering controls, and air sampler locations

- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers’ knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms

- Barrier integrity and performance of engineering controls in two airborne radioactivity areas

- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools.

- Self-assessments and audits related to the access control program since the last inspection

- Corrective action documents related to access controls

- Radiation work permit briefings and worker instructions

- Adequacy of radiological controls such as, required surveys, radiation protection job coverage, and contamination controls during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

The inspectors completed 19 of the required 21 samples.

b. Findings

.1 Introduction. The inspectors identified an NCV of 10 CFR 20.1902(a) because the licensee failed to conspicuously post a radiation area.

Description. On May 2, 2007, during a tour of the auxiliary building, the inspectors observed that the radiological posting to the entryway of the Unit 1 Decay Heat Vault B was not conspicuously posted. An operations' "protected train" sign was hung with barrier tape on the same wall fasteners as the radiological posting. This configuration allowed the "protected train" sign to hang in front of the radiological posting, obscuring the radiological posting from view. The licensee's immediate corrective action was to re-post the operation's sign to prevent obscuring the radiological posting.

Analysis. The failure to conspicuously post a radiation area is a performance deficiency. The finding was greater than minor because it was associated with the occupational radiation safety cornerstone attribute of program and process and affected the cornerstone objective to ensure the adequate protection of a worker's health and safety from exposure to radiation because it could have resulted in workers being exposed to

higher radiation levels. When processed through the occupational radiation safety significance determination process, the finding was determined to be of very low safety significance because it was not an as low as reasonably achievable (ALARA) finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. In addition, this finding has a crosscutting aspect associated with the human performance component of work practices (H.4(a)) because personnel failed to use human error prevention techniques such as self-checking or peer-checking to verify that the radiation area was conspicuously posted.

Enforcement. Title 10 CFR 20.1902(a) states, in part, that the licensee shall post each radiation area with a conspicuous sign bearing the radiation symbol and the words “Caution, Radiation Area.” Contrary to this requirement, the area was not conspicuously posted. Because the finding was of very low safety significance and has been entered into the licensee’s corrective action program as CR ANO-1-2007-01044, this violation is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000313/2007003-02, “Failure to Conspicuously Post a Radiation Area.”

.2 Introduction. The inspectors reviewed a self-revealing NCV of Technical Specification 5.4.1.a. because of a failure to use an engineering control as required by a radiation work permit.

Description. On April 25, 2007, two workers were unable to clear the personnel contamination monitors after performing a radiological survey of the Unit 1 Steam Generator A lower channel head. After two additional personnel contaminations occurred related to the same area, the licensee conducted an investigation and determined that the Steam Generator A high-efficiency particulate air (HEPA) ventilation had been rendered inoperable which caused the contamination of personnel and the surrounding area.

Prior to entering the lower steam generator area and performing the radiological survey, a radiation protection technician went to verify that the HEPA unit for the steam generator ventilation was running. This HEPA unit was verified running and a differential pressure noted on the gauge. However, the radiation protection technician did not recognize that the damper to the steam generator HEPA hose port was closed. A second, open port on the HEPA unit was drawing suction and provided the gauge indication. With the HEPA hose port damper closed, the HEPA unit was rendered ineffective for steam generator ventilation and resulted in the loss of an engineered contamination control. The licensee’s immediate corrective actions were to counsel the workers and brief associated personnel on the correct method for verifying HEPA ventilation.

Analysis. The failure to use an engineering control as required by the radiation work permit is a performance deficiency. The finding was greater than minor because it was associated with the occupational radiation safety cornerstone attribute of program and process and affected the cornerstone objective to ensure the adequate protection of a worker’s health and safety from exposure to radiation because it resulted in workers being exposed to higher contamination levels. When processed through the occupational radiation safety significance determination process, the finding was determined to be of very low safety significance because it was not an ALARA finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. In addition, this finding had a crosscutting aspect

associated with the human performance component of resources (H.2(c)) because the HEPA ventilation verification procedure was not complete in that it did not have sufficient detail.

Enforcement. Technical Specification 5.4.1.a requires applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A, Section 7(e)(1) requires procedures for access control and a radiation work permit system. Fleet Procedure EN-RP-105, "Radiation Work Permits," Revision 1, Section 5.3, for radiation work permit planning requires, in part, a documented "RWP Pre-Job Briefing" in which radiation protection determined the radiological hazards and necessary controls associated with the work. Radiation Work Permit 2007-1442 required HEPA ventilation to be in operation on the steam generators after the manway and diaphragms were removed as an engineering control. The "Pre-Job Briefing" for Task 2 required a HEPA be installed for contamination control. Contrary to these requirements, the HEPA ventilation line up and failure of the technician to recognize the deficiency created a loss of an engineered contamination control which resulted in four personnel contaminations. Because the finding was of very low safety significance and has been entered into the licensee's corrective action program as CR ANO-1-2007-00778, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000313/2007003-03, "Failure to Use an Engineering Control as Required by a Radiation Work Permit."

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures ALARA. The inspectors used the requirements in 10 CFR Part 20 and the licensee's procedures required by TSs as criteria for determining compliance. The inspectors interviewed licensee personnel and reviewed:

- Three outage or on-line maintenance work activities scheduled during the inspection period and associated work activity exposure estimates which were likely to result in the highest personnel collective exposures
- Intended versus actual work activity doses and the reasons for any inconsistencies
- Interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling, and engineering groups
- Integration of ALARA requirements into work procedure and radiation work permit documents
- Shielding requests and dose/benefit analyses
- Exposure tracking system
- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding

- Workers use of the low dose waiting areas

- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

- Self-assessments, audits, and special reports related to the ALARA program since the last inspection

- Corrective action documents related to the ALARA program and follow-up activities such as initial problem identification, characterization, and tracking

The inspectors completed 12 of the required 29 samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems and Barrier Integrity Cornerstone

a. Inspection Scope

The inspectors sampled licensee submittals for the three performance indicators listed below for the period from April 1, 2006, through March 31, 2007 for Units 1 and 2. The definitions and guidance of NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the licensee's basis for reporting each data element in order to verify the accuracy of PI data reported during the assessment period. The inspectors reviewed licensee event reports, monthly operating reports, and operating logs as part of the assessment. Licensee performance indicator data were also reviewed against the requirements of EN-LI-114 "Performance Indicator Process," Revision 2.

- Safety System Functional Failures
- Reactor Coolant System Specific Activity
- Reactor Coolant System Leakage

Documents reviewed by the inspectors are listed in the attachent.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety Cornerstone

a. Inspection Scope

The inspectors reviewed licensee documents from January through March 2007. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee's TSs), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

- Occupational Exposure Control Effectiveness

.3 Public Radiation Safety Cornerstone

a. Inspection Scope

The inspectors reviewed licensee documents from January through March 2007. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspectors interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. Performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify the basis in reporting for each data element.

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Identification and Resolution of Problems for Occupational Radiation Safety

a. Inspection Scope

The inspectors evaluated the effectiveness of the licensee's problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings

No findings of significance were identified.

.2 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a daily screening of items entered into the licensee's CAP. This assessment was accomplished by reviewing CRs and attending corrective action review and work control meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by the licensee at an appropriate threshold and that the issues were entered into the CAP; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional follow-up through other baseline inspection procedures.

b. Findings

No findings of significance were identified.

.3 Selected Issue Follow-up Inspection

a. Inspection Scope

In addition to the routine review, the inspectors selected the two below listed issues for a more in-depth review. The inspectors considered the following during the review of the licensee's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

- May 23, 2007, Unit 2, Loss of Component Cooling Water (CCW) Pump 2P-33B
- June 15, 2007, Unit 2, Electrical Equipment Room Heat Load Calculations

Documents reviewed by the inspectors are listed in the attachment.

b. Findings

Introduction. A Green self-revealing finding was identified when Unit 2 experienced a complete loss of CCW flow due to the loss of the Train B CCW Pump 2P-33B on February 21, 2007. Specifically, the loss of CCW occurred when an operator was attempting to pressurize an out-of-service heat exchanger to support maintenance activities.

Description. On February 21, the CCW system was lined up with Pump 2P-33B, the CCW Pump B, supplying both the nuclear and non-nuclear loop of CCW. Pump 2P-33C

was tagged out for maintenance by instrumentation and control technicians, and the Train C CCW heat exchanger was isolated so the mechanical craft could unplug tubes to recover thermal margin.

The normal lineup for the system was for the C pump to be in operation and the B pump in standby. The A pump was for use with the non-nuclear loop when the system was not in a cross-connected lineup, and would not automatically start like the B or C pumps.

To support the maintenance in progress on the C heat exchanger, the mechanics requested that operations pressurize the C heat exchanger so that they could check for leaks from the tubes that had been unplugged. An operator was sent to pressurize the heat exchanger. While the operator was pressurizing the heat exchanger, the B pump tripped, and this resulted in a complete loss of CCW flow. The licensee entered into their abnormal operating procedure for reactor coolant pump emergencies and subsequently restored CCW flow using the A CCW pump. The inspectors noted that the recovery was accomplished after approximately nine minutes because the abnormal operating procedure did not support system restoration using the A CCW pump. Had recovery taken one more minute, the abnormal operating procedure would have required that operators trip the reactor.

The licensee performed a root cause evaluation of this event as documented in CR ANO-2-2007-0313. During this evaluation, the licensee determined that the heat exchanger had been drained because of leaking tubes and was not full as expected at the start of the evolution. As a result, when the operator began filling the heat exchanger, this caused a pressure transient on the system which tripped the running CCW pump. Also, the operator did not shut the heat exchanger inlet valve because he thought that it was maintenance on the C pump that had caused the trip. This prevented restarting the B pump because of the low pressure condition caused by the filling of the heat exchanger. The licensee determined the root cause for this event to be inadequate procedures because the procedures used to control the maintenance were not adequate for the activities performed.

Analysis. The inspectors determined that the licensee’s failure to provide adequate procedures or training that appropriately addressed the pressurization of an out-of-service heat exchanger without causing a plant transient was a performance deficiency. The finding was determined to be more than minor because it affected the equipment performance attribute of the initiating events and mitigating systems cornerstone objectives. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required, because two cornerstones were affected by the finding.

The inspectors performed a Phase 2 analysis using Appendix A, "Technical Basis For At Power Significance Determination Process," of Manual Chapter 0609, "Significance Determination Process," and the Phase 2 Worksheet for Arkansas Nuclear One. The inspectors assumed that the duration of the CCW system unavailability was very short, approximately 4 hours. Additionally, the inspectors assumed that only the power conversion system was affected and all other mitigating systems were available. Based on the results of the Phase 2 analysis, the finding was determined to have very low safety significance. The finding had crosscutting aspects in the area of human performance associated with resources (H.2(b)) because the training of personnel and procedural guidance was inadequate.

Enforcement. While a performance deficiency was identified, there were no violations of NRC requirements identified during the review of this issue, because CCW is not a safety related system. The licensee has entered this issue into the CAP as CR ANO-2-2007-0313: FIN 05000368/2007003-04, "Complete Loss of Component Cooling Water Flow During Maintenance Operations."

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semi-annual trend review of repetitive or closely related issues that were documented in corrective action documents to identify trends that might indicate the existence of more safety significant issues. The inspectors review consisted of the 6-month period of January 1 through June 23, 2007. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors also reviewed CAP items associated with deficiencies in the safety parameter display system. The inspectors compared and contrasted their results with the results contained in the licensee's quarterly trend reports. Corrective actions associated with a sample of these issues identified in the licensee's trend report were reviewed for adequacy.

When evaluating the effectiveness of the licensee's corrective actions for these issues, the following attributes were considered:

- Complete and accurate identification of the problem in a timely manner commensurate with its significance and ease of discovery
- Evaluation and disposition of operability and reportability issues
- Consideration of extent of condition, generic implications, common cause, and previous occurrences
- Classification and prioritization of the resolution of the problem commensurate with its safety significance
- Identification of root and contributing causes of the problem for significant conditions adverse to quality
- Identification of corrective actions which are appropriately focused to correct the problem
- Completion of corrective actions in a timely manner commensurate with the safety significance of the issue

Documents reviewed by the inspectors are listed in the attachment.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

a. Inspection Scope

The inspectors: (1) reviewed operator logs, plant computer data, and/or strip charts for the below listed evolutions to evaluate operator performance in coping with nonroutine events and transients; (2) verified that operator actions were in accordance with the response required by plant procedures and training; and (3) verified that the licensee has identified and implemented appropriate corrective actions associated with personnel performance problems that occurred during the nonroutine evolutions sampled.

- May 5, 2007, Unit 2, EDG A Exhaust Manifold Lagging Fire
- May 25, 2007, Unit 1, C Phase Main Transformer High Voltage Bushing Oil Leak
- June 6, 2007, Fire on the Auxiliary Building Roof Above the Spent Fuel Floor

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

.1 Fire on Auxiliary Building Roof Above Spent Fuel Floor

Introduction. A self-revealing NCV of Unit 2 Technical Specification 6.4.1.C, "Fire Protection Program Implementation," was identified associated with the licensee's failure to provide training and qualification for fire protection designees which resulted in non-routine hot work activities not being adequately evaluated by appropriately trained individuals. Specifically, the roofing contractor working on the auxiliary building roof required that a 2-hour fire watch was to be stationed following roofing activities involving the use of open flames, but the licensee only required the fire watch be stationed for 30 minutes. As a result, on June 7, 2007, following roofing activities on the auxiliary building roof above the spent fuel floor that involved the use of open flames, two fires occurred after approximately one hour from the completion of hot work activities, and there was not an appropriately trained fire watch in the area.

Description. On June 7, roofing contractors were performing roofing repairs on the auxiliary building roof above the spent fuel floor. These activities were classified as hot work because they involved the use of open flames to aid in the roofing process, and had a licensee fire watch assigned to monitor the job and postwork cool down.

The contractor completed hot work at approximately 7:00 pm and commenced taking 30 minute temperature readings in all areas that they had performed hot work on. The baseline temperature in all areas was 100°F. After approximately 30 minutes all area temperatures except for one, designated Area J, had shown a decrease in temperature. At this time the licensee fire watch was secured by the hot work supervisor because the licensee's procedural requirement was to be present for 30 minutes following the completion of hot work activities in accordance with EN-DC-127, "Control of Hot Work and Ignition Sources," Revision 2. At 8:00 pm the contractor took temperature readings again, and all but Area J temperatures had continued to go down. The contractor noted

that Area J temperature had actually increased, and approximately 5 minutes later the contractor took another temperature reading and temperature had increased again. The contractor removed the top layer of roofing material and discovered smoke and a small fire in the material underneath. This was extinguished and during a subsequent check another area was discovered to have a small fire when the top layer of material was removed. This small fire was also extinguished.

Procedure EN-DC-127 "Control of Hot Work and Ignition Sources," Revision 2, Attachment 9.5, states that the designee for fire protection responsibilities is the cognizant supervisor in all cases described in the procedure. This procedure also states in Section 3 [8] that the hot work supervisor is trained and qualified to this procedure. Also, Section 4.2 [5] states that fire protection/designee provides additional guidance when non-routine hot work activities arise that are not bounded by this procedure, and Section 3 [6] states that a fire watch is to be in constant attendance during the activity and for 30 minutes afterwards (cool down period) unless an alternate time is required/approved by fire protection.

During the inspectors' review of this issue, they noted that the roofing contractor required that a 2-hour fire watch was to be stationed following roofing activities involving the use of open flames, but the licensee only stationed a fire watch for 30 minutes. The inspectors questioned the hot work supervisor about this, and discovered that he had known of the contractors requirement for a 2-hour fire watch but based on his understanding of EN-DC-127, "Control of Hot Work and Ignition Sources," Revision 2, they were only required to station a 30 minute fire watch. He also stated that he did not consult with fire protection on this issue because his understanding was that one only involved fire protection to shorten the length of fire watches, and that 30 minutes was the maximum length of time a fire watch was to be stationed. The inspectors questioned fire protection personnel about this and also about how the supervisors were trained and qualified on this procedure. Based on the licensee's review of their training records, it was determined that the station did not have a formal training or qualification program for Procedure EN-DC-127 "Control of Hot Work and Ignition Sources," Revision 2. The inspectors also noted that the hot work supervisor for the roofing job had not received any formal training on this procedure.

Analysis. The inspectors determined that the licensee's failure to provide training and qualification for fire protection designees which resulted in nonroutine hot work activities not being adequately evaluated by appropriately trained individuals, was a performance deficiency. The finding was determined to be more than minor because it affected the protection against external factors attribute of the initiating events cornerstone, and it directly affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Additionally, if left uncorrected, the practice of not adequately evaluating non-routine hot work activities by appropriately trained individuals would become a more significant safety concern in that it could result in a fire in or near other risk important equipment. Using the Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," Phase 1 Worksheet, the finding was determined to have very low safety significance because the condition did not constitute a high degradation of a fire prevention and administrative controls feature. The finding had crosscutting aspects in the area of human performance associated with decision making (H.1(b)) because the licensee did not use conservative assumptions and failed to verify the validity of the underlying assumptions

Enforcement. Unit 2 TS 6.4.1.c, "Procedures," requires that written procedures be established, implemented, and maintained covering fire protection program implementation. Procedure EN-DC-127, "Control of Hot Work and Ignition Sources," is one of those procedures and requires that hot work supervisors be trained and qualified to this procedure. Contrary to this, on June 7, 2007, a hot work supervisor who had received no formal training or qualification improperly implemented this procedure which resulted in nonrountine hot work activities not being adequately evaluated by appropriately trained individuals. Because this finding is of very low safety significance and has been entered into the CAP as CRs ANO-2-2007-0816 and ANO-2-2007-0839, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000313/2007003-05 "Inadequate Evaluation of Non-Routine Hot Work Activities Resulted in a Failure to Maintain Fire Watch for Required Amount of Time."

.2 Fire in EDG Exhaust Manifold Lagging

Introduction. A self-revealing NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified associated with the exhaust manifold lagging fire that occurred on the Unit 2 EDG 2K-4A on May 11, 2007. Specifically, the licensee failed to adequately implement corrective actions from a previous diesel exhaust manifold fire in 2003 and as such, the licensee failed to identify and correct an oil leak from the front cover of the diesel which resulted in a fire during a monthly surveillance run.

Description. On May 11, local operators were performing a monthly surveillance run of EDG 2K-4A. The EDG had been running fully loaded for approximately 10 minutes when the operators observed a small fire on the exhaust manifold that appeared to originate from under the lagging. The operators observed the fire for approximately 20 seconds and noted that it was growing in size. The operators extinguished the fire with a fire extinguisher. Control room operators unloaded and secured the EDG. The licensee removed the insulation on the exhaust manifold and discovered a small section, approximately 16 square inches, was saturated with oil.

The licensee performed a root cause evaluation of this as documented in CR ANO-2-2007-0718. During this evaluation, the licensee determined the source of the oil to be from the front cover of the EDG. The licensee also determined the root cause of this event to be equipment condition, specifically uncorrected equipment problems.

The inspectors reviewed the licensee's root cause evaluation for this event as well as the corrective actions documented in CR ANO-2-2003-1158 for a previous diesel exhaust manifold fire that was caused by oil leaks in 2003. The inspectors noted that approved corrective actions from the previous event in 2003 included requiring inspections by operators and system engineers for the purpose of identifying and documenting any oil leaks on the EDGs. The inspectors noted that the operators were performing inspections every 12 hours.

The inspectors determined that the inspections that were being performed by both operations and system engineering were inadequate because the system engineer and operators were not proactively investigating the EDGs to identify oil leaks. Instead, they were only performing cursory inspections looking for any obvious leaks and oil puddles.

Analysis. The inspectors determined that the licensee's failure to adequately implement corrective actions from a previous EDG exhaust manifold fire to identify, document, and correct conditions adverse to quality was a performance deficiency. The finding was determined to be more than minor because it affected the protection against external factors attribute of the initiating events cornerstone, and it directly affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using the Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," Phase 1 Worksheet, the finding was determined to have very low safety significance because the condition did not constitute a high degradation of a fire prevention and administrative controls feature.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Contrary to the above, the licensee failed to adequately implement corrective actions to identify and correct this condition adverse to quality. Because this finding is of very low safety significance and has been entered into the CAP as CR ANO-2-2007-0718, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000368/2007003-06, "Ineffective Corrective Actions Fail to Identify and Correct a Condition Adverse to Quality."

.3 Fire In EDG Exhaust Insulation Cover

Introduction. A self-revealing NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified associated with small flash fires that occurred on the Unit 2 EDG 2K-4A on April 15, 2007. Specifically, the licensee failed to verify that the outer protective cover for insulation used on the exhaust manifold was rated for expected temperatures.

Description. On April 15, while performing a surveillance run of EDG 2K-4A, operators observed smoke coming from underneath the insulation on both sides of the EDG. The operators observed that small flash fires were occurring and causing smoke.

The licensee investigated this issue and determined that the insulation on EDG 2K-4A had been replaced on April 11 and there was no oil on or in the insulation in question. When the licensee inspected the installed insulation they determined that the outer cover of the insulation that was located next to the exhaust piping and was in direct contact with the piping and appeared to be backing away. The licensee also determined that the actual insulation material was not damaged. It was also determined that the same type of insulation had been installed on EDG 2K-4B on April 2.

The licensee determined that the cover material on the installed insulation was only rated to 500°F, and the expected temperature of piping was 1000°F. As a result, the insulation started to burn as the EDG exhaust manifold heated up. As part of their corrective action, the licensee replaced the insulation on both EDGs with insulation with the outer wrap appropriately rated for expected temperatures.

Analysis. The inspectors determined that the licensee's failure to adequately evaluate the acceptability of all parts of replacement insulation was a performance deficiency. The finding was determined to be more than minor because it affected the protection against external factors attribute of the initiating events cornerstone, and it directly affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using the Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," Phase 1 Worksheet, the finding was determined to have very low safety significance because the condition did not constitute a high degradation of a fire prevention and administrative controls feature. The finding had crosscutting aspects in the area of human performance associated with work practices (H.4(a)) because the licensee personnel proceeded with work in the face of uncertainty.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that 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 SSCs to which this appendix applies are correctly translated in specifications, drawings, procedures, and instructions. Contrary to the above, the licensee failed to verify that the cover material of the insulation was appropriately rated for expected exhaust piping temperatures. Because this finding is of very low safety significance and has been entered into the CAP as CR ANO-2-2007-0630, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000368/2007003-07, "Improperly Rated Material Results in Small Flash Fire."

4OA5 Other Activities

(Closed) NRC Temporary Instruction (TI) 2515/166, PWR Containment Sump Blockage

The inspectors reviewed ANO's Unit 1 implementation of plant modifications and procedure changes committed to in their response to Generic Letter 2004-02, "Potential Impact of Debris on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors."

The inspectors observed installation of the containment recirculation sump strainer, and screens with the same perforation size as the strainer on drains from the reactor cavity and other drains that bypass the sump strainer, as designed by the vendor. In addition, the inspectors verified that ANO Unit 1 has implemented specific procedure changes to control qualified and unqualified coatings, and also control tags, labels, tape, and other objects inside the containment building.

At the time of the inspection, industry testing for chemical effects on containment recirculation sumps was not complete. Since the testing was not complete, ANO Unit 1 evaluated the new recirculation sump modifications to the original design basis, Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 0. The licensee's commitment ". . . to resolve chemical effects, downstream effects, and the resulting effects on the head-loss for both of the ANO units" has a scheduled completion date of December 31, 2007. Final review and acceptance of the modification will be performed by the Office of Nuclear Reactor Regulation at a later date.

4OA6 Meetings, Including Exit

On May 4, 2007, the health physicis inspector presented the occupational radiation safety inspection (with focus on access controls) results to Mr. T. Mitchell and other members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection. On May 10, 2007, the inspector re-exited with Mr. D. Moore via telephone to correct a NCV.

On May 10, 2007, an engineering inspector presented the results of the inservice inspection and TI 2515/166 inspection to Mr. J. Kowalewski, General Manager, and other members of his staff who acknowledged the findings. The inspector noted that while proprietary information was reviewed, all such documents had been returned to the licensee, and the information would not be included in this report.

The resident inspectors presented the inspection results of the resident inspections to Mr. T. Mitchell, Vice President, Operations, and other members of the licensee's management staff on June 26, 2007. The licensee acknowledged the findings presented. The inspectors noted that while proprietary information was reviewed, none would be included in this report.

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs..

- 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis for those SSCs are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above requirement, the licensee failed to adequately maintain the specifications for the instantaneous overcurrent trip setpoints for the supply breakers to station battery chargers D-03A, D-04A, and D-04B. Specifically, the as-found setting of the overcurrent trip setpoints were below the designed in-rush current leaving the chargers’ supply breakers vulnerable to tripping when restoring from a loss of normal supply power. This condition was identified during a review of industry operating experience. In accordance with Manual Chapter 0609, Appendix A, this finding was of very low safety significance (Green), because it did not represent an actual loss of safety function and does not screen as potentially risk significant due to external events. This issue was entered into the licensee’s CAP as CR ANO-1-2007-0785.

- 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that 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 SSCs to which this appendix applies are correctly translated in specifications, drawings, procedures, and instructions. Contrary to the above requirement, the licensee failed to evaluate, as an input into design heating, ventilation, and air conditioning calculations, power cable heat losses in safety related electrical equipment rooms on both Units 1 and 2. This was licensee identified because it was identified during a review of design

calculations by licensee personnel. In accordance with Manual Chapter 0609, Appendix A, this finding was of very low safety significance (Green), because it was confirmed not to result in loss of operability per Part 9900, Technical

Guidance, “Operability Determination Process for Operability and Functionality Assessment.” This issue was entered into the licensee’s CAP as CRs ANO-1-2007-0289, ANO-C-2007-0613, and ANO-2-2007-0525.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

B. Acree, WSI Quality Control Supervisor
E. Addison, Technical Specialist, ECT Level III
J. Bacquet, ALARA Supervisor, Radiation Protection
R. Barnes, Manager, Planning, Scheduling, and Outages
S. Bennett, Project Manager, Licensing
B. Berryman, Manager, Operations Unit 1
E. Blackard, Supervisor, Engineering Programs
R. Briley, Level II, SIA
J. Browning, Manager, Operations Unit 2
S. Chandler, System Engineer
S. Cotton, Manager, Training
B. Daiber, Supervisor, System Engineering
D. Edgell, Supervisor, System Engineering
J. Eichenberger, Manager, Corrective Actions and Assessments
N. Finney, Level III, SIA
R. Fowler, Emergency Planner
R. Freeman, Emergency Planner
J. Giles, Manager, Technical Support
M. Ginsberg, Supervisor, Engineering Programs and Components
B. Gordon, Project Manager
B. Greeson, Engineering Supervisor
R. Gresham, Emergency Planner
D. Harris, Emergency Planner
J. Hoffpauir, Manager, Maintenance
R. Holeyfield, Manager, Emergency Planning
M. Huff, Supervisor, Project Engineering
B. James, Project Manager
D. James, Manager, Licensing
W. James, Manager, Engineering Projects
R. Jones, Technical Specialist
J. Kowalewski, Acting General Manager, Plant Operations
J. Looper, Units 1 and 2 Supervisor, Radiation Protection
D. MacPhee, Mechanical Design Engineer
T. Marlow, Director, Nuclear Safety Assurance
D. Meatheany, Steam Generator Technical Specialist
J. Meeker, Senior Lead Engineer
J. Miller, Jr., Manager, System Engineering
T. Mitchell, Vice President, Operations
D. Moore, Manager, Radiation Protection
K. Panther, Nondestructive Examination Site Level III
D. Parker, WSI Director of Projects
G. Parks, NDE Supervisor
C. Reasoner, Manager, Engineering Programs and Components
A. Remer, Project Engineer
R. Scheide, Licensing Specialist

W. Sims, Project Engineer
J. Smith, Jr., Project Manager
B. Starkey, Technical Support Supervisor, Radiation Protection
D. Tucker, Engineering Programs Engineer
C. Tyrone, Manager, Quality Assurance
F. Van Buskirk, Licensing Specialist
D. White, Emergency Planner
P. Williams, Supervisor, System Engineering
R. Woodard, WSI Site Manager
M. Woodby, Manager, Design Engineering

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000313/2007003-01	NCV	Failure to Identify the Preconditioning of Reactor Building Spray Pumps (Section 1R22)
05000313/2007003-02	NCV	Failure to Conspicuously Post a Radiation Area (Section 2OS1.1)
05000313/2007003-03	NCV	Failure to Use an Engineering Control as Required by a Radiation Work Permit (Section 2OS1.2)
05000368/2007003-04	FIN	Complete Loss of Component Cooling Water Flow During Maintenance Operations (Section 4OA2.3)
05000368/2007003-05	NCV	Inadequate Evaluation of Non-Routine Hot Work Activities Resulted in a Failure to Maintain Fire Watch for Required Amount of Time (Section 4OA3.1)
05000368/2007003-06	NCV	Ineffective Corrective Actions Fail to Identify and Correct a Condition Adverse to Quality (Section 4OA3.2)
05000368/2007003-07	NCV	Improperly Rated Material Results in Small Flash Fires (Section 4OA3.3)

LIST OF DOCUMENTS REVIEWED

In addition to the documents referred to in the inspection report, the following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Section 1R01: Adverse Weather Protection

Procedures:

NUMBER	TITLE	REVISION
OP-1203.025	Natural Emergencies	22

OP-2203.008	Natural Emergencies	13

Section 1R04: Equipment Alignment

Procedures:

NUMBER	TITLE	REVISION
OP-1015.002	Decay Heat Removal and LTOP System Control	8
OP-1104.006	Spent Fuel Cooling System	36

Drawings:

M-235 “Piping & Instrument Diagram Spent Fuel Cooling System,” Sheet 1, Rev. 67

Section 1R05: Fire Protection

Procedures:

NUMBER	TITLE	REVISION
	Arkansas Nuclear One Fire Hazards Analysis	11
PFP-U1	ANO Prefire Plan (Unit 1) - Section 1B-357-67-U.doc, Section 1B-354-79-U.doc	2
PFP-U2	ANO Prefire Plan (Unit 2) - Section 2B-335-2040-JJ.doc	2

Drawings:

FZ-1070, Sheet 1, Revision 2

FZ-2027, Sheet 1, Revision 3

FZ-1041, Sheet 1, Revision 2

FZ-1043, Sheet 1, Revision 2

FZ-2038, Sheet 1, Revision 2

FZ-1049, Sheet 1, Revision 2

Crs:

ANO-C-2007-0755
ANO-1-2007-0495
ANO-1-2007-0962
ANO-1-2007-0967
ANO-1-2007-1047
ANO-1-2007-1225

Section 1R06: Flood Protection Measures

Procedures:

NUMBER	TITLE	REVISION
OP-2203.008	Natural Emergencies	13

Miscellaneous Documents

Upper Level Document ULD-0-TOP-17, “ANO Flooding Topical,” Revision 0
Engineering Report 92-R-0024-01 Revision 0
Engineering Report 92-R-0034-01 Revision 1
Engineering Report 92-R-0034-02 Revision 1

Section 1R08: Inservice Inspection

CRs:

ANO-1-2005-01438	ANO-1-2007-00578	ANO-1-2007-00790	ANO-1-2007-00966
ANO-1-2005-02866	ANO-1-2007-00635	ANO-1-2007-00799	ANO-1-2007-01127
ANO-2-2006-01746	ANO-1-2007-00678	ANO-1-2007-00861	ANO-1-2007-01148
ANO-1-2007-00523	ANO-1-2007-00703	ANO-1-2007-00934	
ANO-1-2007-00569	ANO-1-2007-00770	ANO-1-2007-00959	

Boric Acid Engineering Evaluations

07-1-0795
07-1-0810
07-1-0820
07-1-0821

Work Orders

NUMBER	TITLE	REVISION
00082026	Perform Boric Acid Inspections Per OP-1032.037 During 1R	1

Drawings

NUMBER	TITLE	REVISION
ANO-39Q-01	Pressurizer Surge Nozzle Weld Overlay Design	4
ANO-39Q-03	Pressurizer 3" Safety Valve Nozzle Weld Overlay Design	2

NDEs

REPORT	COMPONENT/LOCATION	METHOD
102830-PT-018	PZR PSV-1002 Nozzle Final Overlay Weld, OVL-02	PT
102830-PT-019	PZR Spray Nozzle Final Overlay Weld, OVL-04	PT
102830-PT-020	PZR PSV-1001 Nozzle Final Overlay Weld, OVL-01	PT
102830-PT-021	PZR CV-1000 Nozzle Final Overlay Weld, OVL-03	PT
102830-PT-023	PZR Surge Hot Leg Final Overlay Weld, OVL-05	PT
102830-PT-024	PZR Surge Nozzle Final Overlay Weld, OVL-06	PT
ANO-39Q-LPA-001	PZR PSV-1001 Nozzle Final Overlay Weld, OVL-01	UT
ANO-39Q-LPA-002	PZR PSV-1002 Nozzle Final Overlay Weld, OVL-02	UT
ANO-39Q-LPA-003	PZR CV-1000 Nozzle Final Overlay Weld, OVL-03	UT
ANO-39Q-LPA-004	PZR Spray Nozzle Final Overlay Weld, OVL-04	UT
ANO-39Q-LPA-005	PZR Surge Hot Leg Final Overlay Weld, OVL-05	UT
ANO-39Q-LPA-006	PZR Surge Nozzle Final Overlay Weld, OVL-06	UT

Procedure

PROCEDURE,	TITLE,	REVISION
1032.037	Inspection and Identification of Boric Acid Leaks for ANO-1 & ANO-2	4
CEP-ISI-002	Arkansas Nuclear One Unit 1 Inservice Inspection Plan	7
CEP-NDE-0641	Liquid Penetrant Examination (PT) for ASME Section XI	2
CEP-NDE-0731	Magnetic Particle Examination (MT) for ASME Section XI	1

CEP-NDE-0901	VT-1 Examination	1
CEP-NDE-0902	VT-2 Examination	3
CEP-NDE-0903	VT-3 Examination	2
CEP-NDE-3000	ASME Section XI Flaw Evaluation	0
EN-DC-319	Inspection and Evaluation of Boric Acid Leaks	1
QAP 2.7 (WSI)	Selection, Training, Qualification and Certification of Non-Destructive Examination Personnel	13
QAP 9.1 (WSI)	Welding Procedure and Performance Qualification	13
QAP 9.21 (WSI)	Liquid Penetrant Inspection Procedure Solvent Removable Visible Dye for Alloy 690 Weld Overlay	0
QAP 12.0 (WSI)	Control of Measuring and Test Equipment	12
SI-NDE-08 (SIA)	Qualification and Certification of NDE Personnel for Nuclear Applications	1
SI-UT-126 (SIA)	Procedure for the Phased Array Ultrasonic Examination of Weld Overlaid Similar and Dissimilar Metal Welds	3
SI-UT-123 (SIA)	Ultrasonic Wall Thickness Measurement of Components	3

<u>Welding Procedures/Qualification Records</u>

NUMBER	TITLE	REVISION/DATE
PQR 01-01-T-802	Procedure Qualification Record	2, 04/16/07
PQR 01-08-T-032	Procedure Qualification Record	0, 04/14/05
PQR 08-08-T-009	Procedure Qualification Record	0, 08/27/02
PQR 08-08-TS-001	Procedure Qualification Record	0, 01/26/99
PQR 08-08-TS-002	Procedure Qualification Record	0, 08/15/00
PQR 8.8.6-OKG	Procedure Qualification Record	06/03/98
PQR A0143-F43	Procedure Qualification Record	06/12/95
PQR A08202.3-3	Procedure Qualification Record	01/08/91
PQR A843256-52	Procedure Qualification Record	1, 03/30/93

WPQ 2006.04.076 (WSI)	ASME IX Process Expiration Report	04/10/07
WPQ 02636 (WSI)	ASME Section IX - Welder Performance Qualification (WPQ)	02/12/03
WPQ 00910 (WSI)	ASME Section IX - Welder Performance Qualification (WPQ)	07/12/02
WPQ 00904 (WSI)	ASME Section IX - Welder Performance Qualification (WPQ)	07/12/02
WPQ 05629 (WSI)	ASME Section IX - Welder Performance Qualification (WPQ)	08/15/05
WPS 01-08-T-8301-Surge-102830	Welding Procedure Specification, Weld Overlay	1, 04/16/07
WPS 01-08-T-8303-SRV-ERV-102830	Welding Procedure Specification, Weld Overlay	1, 04/16/07
WPS 01-43-08-T-8303-Spray-102830	Welding Procedure Specification, Weld Overlay	1, 04/16/07
WPS 08-08-T-001	Welding Procedure Specification	1, 12/05/02
WPS 08-08-T-031-102830	Welding Procedure Specification	0, 04/16/07
WSI Traveler No. 102830-301-1	Work Traveler For Pressurizer Nozzle WOL Repair (PSV-1001 Nozzle)	0, 04/19/07
WSI Traveler No. 102830-310	Work Traveler For Pressurizer Nozzle WOL Repair Installation of ER308L Buffer Layer (Surge Nozzle)	1, 05/01/07
WSI Traveler No. 102830-304	Work Traveler For Pressurizer Nozzle WOL Repair (Surge Nozzle)	1, 04/30/07

Relief Requests

"ARKANSAS NUCLEAR ONE, UNIT 1 - REQUEST FOR ALTERNATIVE NO. ANO1-ISI-007 TO EXTEND THE THIRD 10-YEAR INSERVICE INSPECTION INTERVAL FOR REACTOR VESSEL INTERIOR ATTACHMENTS AND CORE SUPPORT STRUCTURE VISUAL EXAMINATIONS (TAC NO. MD1395)," 01/31/2007.

"ARKANSAS NUCLEAR ONE, UNIT 1 - REQUEST FOR ALTERNATIVE ANO1-PT-001, RELIEF FROM SYSTEM HYDROSTATIC TEST REQUIREMENTS FOR THE EXTENDED REACTOR COOLANT PRESSURE BOUNDARY PIPING (TAC NO. MD1394)," 01/31/2007.

"ARKANSAS NUCLEAR ONE, UNIT 1 - REQUEST FOR ALTERNATIVE NO. ANO1-ISI-006 TO EXTEND THE THIRD INSERVICE INSPECTION INTERVAL FOR REACTOR VESSEL EXAMINATION CATEGORY B-F WELD EXAMINATIONS (TAC NO. MD1397)," 02/01/2007.

"ARKANSAS NUCLEAR ONE, UNIT 1 - REQUEST FOR ALTERNATIVE ANO1-R&R-010 TO USE PROPOSED ALTERNATIVE TO THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE REQUIREMENTS FOR PRESSURIZER NOZZLE WELD OVERLAY REPAIRS (TAC NO. MD4019)," 04/16/2007.

Miscellaneous

NUMBER	TITLE	REVISION/ DATE
	Boric Acid Corrosion Control Program 90 Day Report Cycle 19 and Unit 1 Refueling Outage, 1R19	
ANO-EC-413	Steam Generator Pre-Outage Degradation Assessment and Repair Criteria for 1R20	1
ANO-EC-855	1R20 Steam Generator Eddy Current Examination Technique Equivalency Report	
BARK-01-07	Analysis Technique Specification Sheet Arkansas 0.510" X-Probe - 24 IPS	2
BARK-02-07	Analysis Technique Specification Sheet Arkansas 0.510" Bobbin Coil	1
BARK-03-07	Analysis Technique Specification Sheet Arkansas 0.480" Bobbin Coil	1
BARK-04-07	Analysis Technique Specification Sheet Arkansas 3-Coil RPC	1
BARK-A-07	Analysis Technique Specification Sheet X-probe Bobbin Coil Channels	1, 04/27/07
BARK-B-07	Analysis Technique Specification Sheet Conventional Bobbin Coil Probe	1, 04/27/07
BARK-C-07	Analysis Technique Specification Sheet 3 Coil +PT RPC	1, 04/27/07
BARK-D-07	Analysis Technique Specification Sheet X-probe Array Channels	2, 04/29/07

BARK-E-07	Analysis Technique Specification Sheet Bobbin and Rotating Coil DQM	1, 04/27/07
BARK-F-07	Analysis Technique Specification Sheet Bobbin and Rotating Coil - RES	1, 04/27/07
CNRO-2007-00001	Request for Alternative ANO1-R&R-010 Proposed Alternative to ASME Code Requirements for Weld Overlay Repairs	01/12/07
CNRO-2007-00014	Request for Alternative ANO1-R&R-010 Proposed Alternative to ASME Code Requirements for Weld Overlay Repairs	03/22/07
EC-854	Arkansas Nuclear One Unit 1 Steam Generator Integrity Program Steam Generator ECT Data Analysis Training Manual	1R20
EC-847	ANO-1 Steam Generator Eddy Current Examination Data Analysis Guidelines	0, 04/24/07
ER-ANO-2006-0207-000	Operational Assessment of ANO-1 Steam Generator Tubing for Cycle 20	03/31/06
WSI Document NO. 102987-MR-001	Surge Line Welding Issue Southern California Edison (SCE) - SONGS Unit 3, Non Proprietary Version	0

Section 1R12: Maintenance Effectiveness

Procedures:

NUMBER	TITLE	REVISION
EN-DC-207	ANO Maintenance Rule Program	2

Crs:

ANO-1-2007-1532 ANO-1-2007-1422 ANO-1-2007-1532
ANO-1-2007-1422

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures:

NUMBER	TITLE	REVISION
COPD-024	Risk Assessment Guidelines	18

Miscellaneous Documents

Plant Risk Assessment, Dated 4/19/2007
Plant Risk Assessment, Dated 5/04/2007
Plant Risk Assessment, Dated 6/12/2007
Plant Risk Assessment, Dated 6/13/2007

Section 1R15: Operability Evaluations

Procedures:

NUMBER	TITLE	REVISION
EN-OP-104	Operability Determinations	2
OP-2304.135	Unit 2 EDG 2K4B Instrumentation Calibration	17

Crs:

ANO-2-2007-0650
ANO-2-2007-0593
ANO-1-2007-0631
ANO-2-2007-0671
ANO-1-2007-0785
ANO-1-2007-0354
ANO-C-2007-0588
ANO-C-2007-0592
ANO-1-2007-0536

Work Order:

51055885
108229
108244

Section 1R19 Postmaintenance Testing

Procedures:

NUMBER	TITLE	REVISION
OP-1304.063	Unit 1 P-7A Speed Control Calibration	14
OP-1106.006	Emergency Feedwater Pump Operation	67
OP-1304.041	Unit 1 Reactor Protection System Channel A Calibration	34
OP-1304.057	Unit 1 Source Range Channels Calibration	015-00-0
OP-1104.036	Emergency Diesel Generator Operation	46

Work Order:

51030381
51031034
00086818
0393311
00111069
109967
51055885
108229
108244

MISCELLANEOUS DOCUMENTS

EC-1274, Temporary Modification - Disable NI-5 RPS Ch. A Output to EFIC CH. A
EC-1419, Temporary Modification - Force Fail NI-5 Signal to Zero

CRs:

ANO-C-2007-0588
ANO-C-2007-0592
ANO-1-2007-0536
ANO-1-2007-0759

Section 1R20: Refueling and Outage Activities

Procedures:

NUMBER	TITLE	REVISION
EN-RP-101	Access Control For Radiologically Controlled Areas	1
OP-1103.011	Draining And N_2 Blanketing The RCS	32
OP-1015.036	Containment Building Closeout	19
OP-1015.021	ANO-2 EOP/AOP User Guide	004-07-0
OP-1000.006	Procedure Control	58
OP-1203.003	Control Rod Drive Malfunction Action	020-04-0
OP-1102.002	Plant Startup	74

Drawings:

NUMBER	TITLE	REVISION
OP-1103.011	Draining and N2 Blanketing The RCS	32

CRs:

ANO-1-2007-0668
ANO-1-2007-0637
ANO-1-2007-0701
ANO-1-2007-0646
ANO-1-2005-1434
ANO-C-2005-0755
ANO-1-2007-0549
ANO-C-2007-0240
ANO-1-2007-0744
ANO-C-2007-0694
ANO-C12007-0740
ANO-1-2007-0968
ANO-1-2007-1190
ANO-2-2006-1757
ANO-C-2007-0779
ANO-1-2007-1305
ANO-1-2007-0758
ANO-1-2007-1279
ANO-1-2007-1189
ANO-C-2007-0751
ANO-1-2007-0972
ANO-C-2007-0714
ANO-1-2007-0907
ANO-1-2007-0816
ANO-1-2007-0855
ANO-1-2007-0908
ANO-1-2007-0856
ANO-1-2007-0681
ANO-1-2007-0765
ANO-1-2007-0684
ANO-1-2007-1351
ANO-1-2007-1334
ANO-1-2007-1319
ANO-1-2007-1316

MISCELLANEOUS DOCUMENTS

Containment Survey ANO-0704-0374
Containment Survey ANO-0704-0478
Dosimetry Investigation Report, Dated 4/23/2007
License Amendment No. 174
License Amendment No. 215
System Training Manual 1-63, Reactor Protection System, Revision 6
System Training Manual 1-66, Emergency Feedwater Initiation and Control, Revision 9
ER-ANO-2001-1208-011, Chesterton 772 was used instead of N-5000 on sump project, Revision 0

Section 1R22 Surveillance Testing

Procedures:

NUMBER	TITLE	REVISION
OP-2106.006	Emergency Feedwater System Operations	60
OP-2312.044	CPC-A Triannual Channel Functional Test	7
OP-1305.018	Local Leak Rate Testing - Type C	16
OP-1103.013	RCS Leak Detection	28
OP-2104.040	LPSI System Operations	40
OP-1104.005	Reactor Building Spray System Operation	47

Crs:

ANO-1-2007-1465	ANO-1-2007-0354	ANO-C-1997-0288
ANO-2-2007-0612	ANO-1-2007-1645	

Work Order:

51031064	51083132	51020690

1R23: Temporary Plant Modifications

Procedures:

NUMBER	TITLE	REVISION
EN-DC-136	Temporary Modifications	1

Crs:

ANO-1-2005-1434	ANO-1-2007-1306

Work Order:

00111069

MISCELLANEOUS DOCUMENTS

ER-ANO-2005-0858-000, Engineering Evaluation for Temporary Alteration of P-32C Motor, Revision 0

EC-1274, Temporary Mod - Disable NI-5 RPS CH. A Output to EFIC CH. A, Revision 0

Section 2OS1: Access to Radiologically Significant Areas

Corrective Action Documents

CR-ANO-1-2007-00171,	CR-ANO-1-2007-00637,	CR-ANO-1-2007-00668,
CR-ANO-1-2007-00778,	CR-ANO-1-2007-00825,	CR-ANO-1-2007-00895,
CR-ANO-1-2007-01044,	CR-ANO-C-2007-00014,	CR-ANO-C-2007-00070,
CR-ANO-C-2007-00285		

Oversight Observations

O2C-ANO-2007-0006,	O2C-ANO-2007-0012,	O2C-ANO-2007-0052,
O2C-ANO-2007-0081,	O2C-ANO-2007-0098	

Procedures

1012.017	Radiological Posting and Entry/Exit Requirements, Change Number 11
1012.018	Administration of Radiological Surveys, Change Number 10
EN-RP-100	Radworker Expectations, Revision 0
EN-RP-101	Access Control for Radiologically Controlled Areas, Revision 1
EN-RP-106	Radiological Survey Documentation, Revision 0
EN-RP-108	Radiation Protection Posting, Revision 3
EN-RP-131	Air Sampling, Revision 2
EN-RP-141	Job Coverage, Revision 1
EN-RP-304	Operation of Counting Equipment, Revision 0

Radiation Work Permits

2007-1415, 2007-1430, 2007-1442, 2007-1461

Miscellaneous

1R20 Alpha Monitoring Plan, Revised April 10, 2007
Air Samples: AS-2007-00145, AS-2007-00151, AS-2007-00520
Radiological Surveys from the Auxiliary and Reactor Buildings

Section 2OS2: ALARA Planning and Work Controls

Corrective Action Documents

CR-ANO-1-2007-00117, CR-ANO-2-2007-00029, CR-ANO-2-2007-00307

Oversight Observations

O2C-ANO-2007-0005, O2C-ANO-2007-0008, O2C-ANO-2007-0009

Procedures

1012.032	ALARA Work Control and Planning, Change Number 0
EN-RP-100	Radworker Expectations, Revision 0
EN-RP-104	Personnel Contamination Events, Revision 1
EN-RP-105	Radiation Work Permits, Revision 1

Radiation Work Permits

2007-1415, 2007-1430

Miscellaneous

Shiftly ALARA Reports Outage Days 8, 9, and 10

Section 4OA1: Performance Indicator Verification

Procedures

EN-LI-114	Performance Indicator Process, Revision 0

Miscellaneous

Annual Radioactive Effluent Release Report for 2006

NRC Performance Indicator Technique/Data Sheet, Attachment 9.2 for January, February, March 2006

Section 4OA2: Identification and Resolution of Problems

Procedures:

NUMBER	TITLE	REVISION
OP-2104.028	Component Cooling Water System Operations	26
OP-2203.012L	Annunciator 2K12 Corrective Action	34
EN-OP-115	Conduct of Operations	3
EN-WM-102	Work Implementation and Closeout	0
EN-LI-102	Corrective Action Process	8

EN-WM-105	Planning	2
91-E-0090-03	ANO-2 Battery DC and Corridor 2104 Emer. Ventilation	4
91-E-0090-04	4160V Switchgear Room Ventilation	2
91-E-0090-05	North Electrical Room 2091 Ventilation	1
91-E-0090-12	Effects of Loss of 4160V Switchgear Exhaust Ventilation	0
91-E-0090-01	Heat Load Determination for Rooms 2091, 2097, 2100, 2100, 2101, 2104 for Post Accident Cooling	3

CRS:

ANO-2-2006-2523	ANO-C-2004-2140	ANO-2-2007-0435
ANO-2-2007-0313	ANO-1-2002-0475	ANO-C-2007-0512
ANO-2-2002-1870	ANO-C-2002-0417	ANO-C-2007-0431
ANO-C-2007-0289	ANO-C-2002-0989	ANO-C-2007-0522
ANO-2-2007-0387	ANO-C-2003-0407	ANO-C-2007-0559
ANO-2-2007-0525	ANO-C-2004-2192	ANO-C-2007-0567
ANO-2-2007-0393	ANO-C-2005-0643	ANO-C-2007-0532
ANO-2-2007-0442	ANO-C-2005-0684	ANO-C-2002-0426
ANO-2-2007-0489	ANO-C-2004-0700	ANO-C-2005-1294
ANO-C-2001-0624	ANO-C-2005-0863	ANO-C-2005-1475
ANO-2-2007-0410	ANO-C-2004-2129	ANO-C-2004-2149
ANO-C-2004-2133	ANO-1-2005-3056	
ANO-C-2003-0453	ANO-C-2007-0008	

Work Order:

50274410
00062357-01

Miscellaneous

System Training Manual 1-71, Safety Parameter Display System, Revision 5

Section 4OA3: Event Follow-up

Crs:

ANO-C-2007-0964	ANO-2-2007-0816	ANO-2-2003-1158
ANO-2-2007-0630	ANO-2-2007-0839	ANO-2-2007-0718

Work Order:

00089004

Procedures

PROCEDURE	TITLE	REVISION
EN-DC-127	Control of Hot Work and Ignition Sources	2

Section 4OA5: Other Activities (TI 2515/0166)

Safety Evaluation

NUMBER	TITLE	REVISION
FFN-07-002	ANO-1 Upgrade Reactor Building Sump Strainer	0

Procedures

PROCEDURE	TITLE	REVISION
1000.168	ANO Safety-Related Coatings Program	0
1015.036	Containment Building Closeout	19

Engineering Requests

NUMBER	TITLE	REVISION
ER-ANO-2001-1205-000	ANO-1 Sump Screen Blockage Issue as Defined by the NRC in GSI-191	0
ER-ANO-2001-1205-002	Prepare NCP for Unit 1 Cal-Sil & Fibrous Insulation Replacement	0

Miscellaneous

NUMBER	TITLE	REVISION/DATE
0CAN080501	Response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" Arkansas Nuclear One - Units 1 and 2	08/31/05
0CAN120504	Additional Response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" Arkansas Nuclear One - Units 1 and 2	12/15/05
0CAN080602	Generic Letter 2004-02 Extension Request and Commitment Revision Arkansas Nuclear One - Units 1 and 2	08/23/06
0CAN100601	Generic Letter 2004-02 Extension Request Withdrawal Arkansas Nuclear One - Units 1 and 2	10/18/06

LIST OF ACRONYMS

ALARA	as low as is reasonable achievable
ANO	Arkansas Nuclear One
ASME Code	American Society of Mechanical Engineers Boiler and Pressure Vessel Code
CAP	corrective action program
CCW	component cooling water
CFR	*Code of Federal Regulations*
CR	condition report
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
HEPA	high-efficiency particulate air
IN	Information Notice
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PI	performance indicator
PWR	pressurized water reactor
RTP	rated thermal power
SSC	structures, systems, component
TI	temporary instruction
UFSAR	*Updated Final Safety Analysis Report*