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

February 14, 2012

Christopher J. Schwarz, Site Vice President
Arkansas Nuclear One
Entergy Operations, Inc.
1448 SR 333
Russellville, AR 72802-0967

SUBJECT: ARKANSAS NUCLEAR ONE - NRC INTEGRATED INSPECTION REPORT NUMBER 05000313/2011005 AND 05000368/2011005

Dear Mr. Schwarz:

On December 31, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Arkansas Nuclear One facility Units, 1 and 2. The enclosed inspection report documents the inspection results which were discussed on January 20, 2012, with Mr. M. Chisum, General Manager, Plant Operations, and other members of your staff.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Two NRC identified and four self-revealing findings of very low safety significance (Green) were identified during this inspection.

Five of these findings were determined to involve violations of NRC requirements. Further, two licensee-identified violations which were determined to be of very low safety significance are listed in this report. The NRC is treating these violations as noncited violations (NCVs) consistent with Section 2.3.2 of the 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 Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Arkansas Nuclear One.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC Resident Inspector at Arkansas Nuclear One.

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 available electronically for public inspection in the

NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management 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/

Donald B. Allen, Branch Chief
Project Branch E
Division of Reactor Projects

Docket Nos: 05000313, 05000368
License Nos: DPR-51, NPF-6

Enclosure: Inspection Report 05000313/2011005 and 05000368/2011005
w/ Attachment: Supplemental Information

cc w/ encl: Electronic Distribution

<u>DISTRIBUTION</u>:
Regional Administrator (Elmo.Collins@nrc.gov)
Deputy Regional Administrator (Art.Howell@nrc.gov)
DRP Director (Kriss.Kennedy@nrc.gov)
DRP Deputy Director (Troy.Pruett@nrc.gov)
DRS Director (Anton.Vegel@nrc.gov)
DRS Deputy Director (Tom.Blount@nrc.gov)
Senior Resident Inspector (Alfred.Sanchez@nrc.gov)
Resident Inspector (Jeff.Rotton@nrc.gov)
Resident Inspector (William.Schaup@nrc.gov)
Branch Chief, DRP/E (Don.Allen@nrc.gov)
Senior Project Engineer, DRP/E (Ray.Azua@nrc.gov)
Project Engineer (Jim.Melfi@nrc.gov)
Project Engineer (Dan.Bradley@nrc.gov)
ANO Administrative Assistant (Gloria.Hatfield@nrc.gov)
Public Affairs Officer (Victor.Dricks@nrc.gov)
Public Affairs Officer (Lara.Uselding@nrc.gov)
Project Manager (Kaly.Kalyanam@nrc.gov)
Branch Chief, DRS/TSB (Ryan.Alexander@nrc.gov)
RITS Coordinator (Marisa.Herrera@nrc.gov)
Regional Counsel (Karla.Fuller@nrc.gov)
Congressional Affairs Officer (Jenny.Weil@nrc.gov)
ACES (Heather.Gepford@nrc.gov)
OEMail Resource
ROPreports
OEDO RIV Coordinator (Lydia.Chang@nrc.gov)
NSIR/DPR/EP (Eric.Schrader@nrc.gov)
Regional State Liaison Officer (Bill.Maier@nrc.gov)

R:_REACTORS_ANO\2011\ANO2011005-RP-AS.docx ML120450828

SUNSI Rev Compl.	☑ Yes ☐ No	ADAMS		☑ Yes ☐ No	Reviewer Initials	DBA
Publicly Avail	☑ Yes ☐ No	Sensitive		☐ Yes ☑ No	Sens. Type Initials	DBA
SRI:DRP/E	RI:DRP/E	RI:DRP/E		SPE:DRP/E	C:DRS/EB1	C:DRS/EB2
FSanchez	JRotton	WSchaup		RAzua	TFarnholtz	GMiller
/RA via Email/	***/RA via Email/***	***/RA via Email/***		***/RA/***	***/RA/***	***/RA/***
2/10/12	2/10/12	2/10/12		2/10/12	2/13/12	2/13/12
C:DRS/OB	C:DRS/PSB1			C:DRS/PSB2	C:DRS/TSB	C:DRP/E
MHaire	MHay			GWerner	DPowers	DAllen
/RA/	***/RA/***			***/RA/***	***/RA/***	***/RA/***
2/13/12	2/13/12			2/13/12	2/13/12	2/14/12

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

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000313; 05000368
License:	DPR-51; NPF-6
Report:	05000313/2011005; 05000368/2011005
Licensee:	Entergy Operations Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	Junction of Hwy. 64 West and Hwy. 333 South Russellville, Arkansas
Dates:	October 1 through December 31, 2011
Inspectors:	A. Sanchez, Senior Resident Inspector J. Rotton, Resident Inspector W. Schaup, Resident Inspector G. Guerra, CHP, Emergency Preparedness Inspector R. Kopriva, Senior Reactor Inspector M. Williams, Reactor Inspector
Approved By:	Don Allen, Chief, Project Branch E Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000313/2011005; 05000368/2011005; 10/1/2011-12/31/2011; Arkansas Nuclear One Integrated Resident and Regional Report; Operability Evaluations and Functionality Assessments; Refueling and Other Outage Activities; Problem Identification and Resolution.

The report covered a 3-month period of inspection by resident inspectors and announced baseline inspections by region-based inspectors. Five Green noncited violations of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” The cross-cutting aspect is determined using Inspection Manual Chapter 0310, “Components Within the Cross Cutting Areas.” Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of Unit 1 Technical Specification 3.8.4, “DC Sources-Operating,” Technical Specification 3.8.7, “Inverters- Operating,” and Technical Specification 3.8.9, “Distribution Systems-Operating,” due to the licensee’s failure to complete the associated required action prior to the specified completion time while the associated emergency switchgear room chillers were out of service for planned maintenance. The licensee immediately implemented corrective actions to direct Operations to enter the applicable technical specifications and notify ANO management. The issue was identified to the licensee and entered into their corrective action program as Condition Report CR-ANO-1-2012-0043.

  The inspectors determined that not completing the required actions for the applicable technical specifications prior to the specified completion time while the associated emergency switchgear room chillers were out of service for planned maintenance is a performance deficiency. The performance deficiency is determined to be more than minor because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone, and adversely affects the associated cornerstone objective to ensure availability, reliability, and the capability of systems that respond to initiating events to prevent undesirable consequences and is therefore a finding. Specifically, on December 7, 2011, the failure to complete the required actions prior to the specified completion times for Technical Specification 3.8.4, “DC Sources – Operating,” Technical Specification 3.8.7, “Inverters – Operating, and Technical Specification 3.8.9, “ Distribution Systems – Operating,” after removing the VCH-4A from service for maintenance was a violation of technical specifications. Additionally, on December 19, 2011, the failure to complete the required actions prior to the specified completion time for Technical Specification 3.8.7, “Inverters – Operating,” after removing the

VCH-4B from service for maintenance, was a violation of technical specifications. Using Inspection Manual Chapter 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” the finding was determined to require a Phase 2 analysis because removing each VCH-4 chiller from service in December 2011 did result in an actual loss of safety function of a single train for greater than its technical specification allowed completion time. A phase 2 analysis from a previous noncited violation that bounds this issue determined the finding to be of very low safety significance (Green). Specifically, although the function was lost by the designated support equipment (emergency switchgear chillers), the licensee had an evaluation that credited compensatory measures and specific environmental conditions that assured the overall functionality of the applicable switchgear train was not lost. The finding was determined to have a cross-cutting aspect in the area of human performance, associated with the decision making component, in that the licensee did not use conservative assumptions in decision making and adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement that it is unsafe in order to disapprove the action [H.1(b)] (Section 1R15).

- Green. The inspectors documented a self-revealing, noncited violation of 10 CFR 50 Appendix B, Criterion XVI, “Corrective Action,” for the licensee’s failure to promptly identify and correct a condition adverse to quality associated with degradation of the protective wrap (brand name – Denso) installed on the Unit 1 service water pump columns. The Denso protective wrap around the P-4C service water pump suction column became unraveled and was drawn into the pump suction while running and caused high differential pressure across the pump discharge strainer. The licensee took immediate corrective action to secure the pump and then removed the Denso protective wrap from all pump columns in the Unit 1 service water intake structure bays. Unit 2 does not have Denso protective wrap installed on their service water pumps. The licensee has entered this issue into their corrective action program as Condition Report CR-ANO-1-2011-2843.

  The failure to promptly identify and correct the observed degradation of the protective wrap installed on the Unit 1 service water pump columns is determined to be a performance deficiency. The performance deficiency is determined to be more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affects the cornerstone objective to ensure availability, reliability, and the capability of systems that respond to initiating events to prevent undesirable consequences and is therefore a finding. The inspectors performed the significance determination for the failure of service water pump 4C using NRC Inspection Manual Chapter 0609, Attachment 0609.04, “Phase 1 – Initial Screening and Characterization of Findings.” The problem had occurred during an outage, but it could have occurred at power during a system realignment. The at-power model was more conservative, so it was used to evaluate the finding. Service water was a two train system with a swing pump (an installed spare). The allowed outage time for one train was 72 hours. Operators could easily align the swing

pump to provide the train B service water loads within 72 hours. Therefore, this finding screened to Green because: 1) it was not a design or qualification deficiency; 2) it did not result in loss of safety function of one train of equipment for more than its technical specification allowed outage time; 3) It did not result in a loss of one train of non-technical specification equipment; and 4) it did not screen as potentially risk significant due to an external event. The finding was determined to have a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component in that the licensee failed to thoroughly evaluate the degraded protective wrap such that the resolutions addressed causes and extent of conditions, to include operability of the service water pump [P.1(c)] (Section 1R20.2).

- Green. The inspectors documented a self revealing, noncited violation of Unit 1 Technical Specification 5.4.1.a for the failure to implement station procedure OP-1015.049 “Configuration Control Program”, Revision 1. Specifically, on multiple occasions, station personnel failed to maintain configuration control through the use of valve line-ups and station procedures to ensure reactor plant components were in required positions. In each specific example the licensee took action to place the applicable system in a safe configuration. The licensee is implementing long term programmatic corrective actions. The licensee has placed that issue into their corrective action program as Condition Report CR-ANO-C-2011-2942.

  The failure of station personnel to maintain configuration control through the use of valve line-ups and governing station procedures is a performance deficiency. The performance deficiency is more than minor because it is associated with the configuration control attribute of the Mitigating Systems cornerstone and adversely affects the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences and is therefore a finding. Using Manual Chapter 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” the examples included an actual loss of safety function of a non-technical specification train of equipment designated as risk-significant per 10CFR50.65, for greater than 24 hours. A Phase 3 significance determination analysis was performed by a Region IV senior reactor analyst. The dominant core damage sequences for Unit 1 were station blackouts with battery depletion and transients with loss of feedwater and feed and bleed capability. The dominant core damage sequences for Unit 2 were station blackout with loss of emergency feedwater and once-through-cooling, loss of 4160 volt vital bus 2A4 with loss of feedwater and once-through-cooling, and station blackout with an 8-hour battery depletion. Based on both units having the capability to operate a steam driven emergency feedwater pump during the dominate core damage sequences the finding was determined to have very low safety significance (Green). The finding was determined to have a cross-cutting aspect in the area of human performance, associated with the work practices component in that the licensee failed to define and effectively communicate expectations regarding procedural guidance and personnel follow procedures when performing component positioning [H.4(b)] (Section 4OA2.4).

Cornerstone: Barrier Integrity

- Green. The inspectors documented a self-revealing, noncited violation of Unit 1 Technical Specification 5.4.1.a for the failure to implement station procedure OP-1104.006 "Spent Fuel Cooling System", Revision 51. Specifically, SF-10, flow control to purification loop valve, was found 3 turns open when it was required to be closed. This resulted in the spent fuel pool level lowering by 0.6 feet, which was below procedural limits, when the fuel transfer canal was placed in purification and SF-45, transfer tube isolation valve, was closed to support diving operations in the Unit 1 spent fuel pool tilt pit. After receiving the spent fuel pool low level alarm, operations personnel secured purification, and opened SF-45 which allowed water level to return to normal. Additional actions taken by the licensee included identifying that SF-10 requires a torque amplifying device to operate. The issue was entered into the licensee's corrective action program as Condition Report CR-ANO-1-2011-2498.

  The failure of operations personnel to follow the requirements of procedure OP-1104.006 and close SF-10 prior to initiating fuel transfer canal on purification, which resulted in an unexpected loss of approximately 4500 gallons of water from the spent fuel pool, is a performance deficiency. The performance deficiency is more than minor because it is associated with the configuration control attribute of the Barrier Integrity cornerstone and adversely affects the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events and is therefore a finding. Using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to have very low safety significance (Green) because the finding did not result in the loss of spent fuel cooling, did not result from fuel handling errors that caused damage to the fuel clad integrity or a dropped assembly and did not result in a loss of spent fuel inventory of greater than 10 percent of the spent fuel pool volume. The finding was determined to have a cross-cutting aspect in the area of human performance, associated with the work control component in that the licensee failed to ensure that work activities were appropriately coordinated to support long term equipment reliability by limiting operator work-arounds when a torque amplifying device was required to shut SF-10 [H.3(b)] (Section 1R20.1).

- Green. The inspectors identified a noncited violation of 10 CFR 50, Appendix B, Criterion XVI for failure to identify and correct a condition adverse to quality. Specifically, on November 1, 2011, the licensee failed to identify and correct a condition associated with seating an irradiated fuel bundle into a reactor building storage location during core re-loading activities. The licensee failed to thoroughly evaluate a discrepancy associated with an unexpected vertical measurement when inserting an irradiated fuel bundle prior to unlatching the fuel

bundle. This resulted in the bundle dropping 1 1/8 inches when the licensee attempted to retrieve it. After the bundle dropped, the licensee immediately performed a visual inspection and, with vendor analysis support, removed the bundle from service. The licensee entered this issue into the corrective action program as Condition Report CR-ANO-1-2012-0110.

The failure to identify and correct the discrepancy in the vertical position of an irradiated fuel bundle during fuel handling operations is a performance deficiency. The performance deficiency is determined to be more than minor because it is associated with the human performance attribute of the Barrier Integrity cornerstone and adversely affects the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the performance deficiency resulted in a dropped fuel bundle that was subsequently removed from service due to possible fuel pellet damage. The event also took place while the reactor building was open to the atmosphere. Using Manual Chapter 0609, Appendix G, Attachment 1, Checklist 4, "PWR Refueling Operation: RCS Level >23'," the finding was determined to have very low safety significance (Green) because the finding did not adversely affect: 1) core heat removal, 2) inventory control, 3) electrical power, 4) containment control, or 5) reactivity control. The finding was determined to have a cross-cutting aspect in the area of human performance, associated with decision making component in that the licensee failed to use conservative assumptions and adopt a requirement to demonstrate that the proposed action is safe in order to proceed when deciding to accept the discrepancy in the vertical measurement when storing a fuel bundle in the reactor building storage rack [H.1(b)] (Section 1R20.3).

- Green. The inspectors documented a self-revealing finding for the failure to take adequate corrective actions for known deficiencies associated with the Unit 1 fuel transfer system. Specifically, the licensee failed to investigate and correct issues that had been identified by site and vendor personnel from 1996 through 2010. This led to repeated fuel transfer system failures and significant core offload and reload delays during the 1R23 refueling outage, which placed the plant in an unplanned configuration for an extended period of time. After the failure of the fuel transfer equipment, multiple corrective actions were performed which included the installation of a temporary modification which allowed fuel movement to continue to support core reloading. The issue was entered into the licensee's corrective action program as Condition Report CR-ANO-1-2011-2558.

  The failure of the licensee to take effective corrective action for known deficiencies related to the Unit 1 fuel transfer system is determined to be a performance deficiency. The performance deficiency is determined to be more than minor because, if left uncorrected, the performance deficiency could become a more safety significant issue. Specifically, the continued failure of the licensee to correct known deficiencies in the fuel transfer system could lead to damage to a fuel bundle. Using Manual Chapter 0609, Appendix G, Attachment 1, Checklist 4, "PWR Refueling Operation: RCS Level >23'," the

finding was determined to have very low safety significance (Green) because the finding did not adversely affect: 1) core heat removal, 2) inventory control, 3) electrical power, 4) containment control, or 5) reactivity control. The finding was determined to have a cross-cutting aspect in the area of human performance, associated with decision making component in that the licensee failed to use conservative assumptions and adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. Specifically, the decision making efforts affecting the fuel transfer system did not reflect a safety minded culture as past experience and vendor recommendations were disregarded [H.1(b)] (Section 1R20.4).

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 corrective action tracking numbers (condition report numbers) are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the period at 93 percent reactor power in coastdown to refueling outage 1R23. On October 16, 2011, Unit 1 entered Mode 3 to begin refueling outage 1R23. On November 22, 2011, Unit 1 closed the main generator breaker to end refueling outage 1R23. On November 26, 2011, Unit 1 reached 100 percent reactor power and remained there for the remainder of the period.

Unit 2 began the period at 100 percent reactor power. On December 20, 2011, Unit 2 reduced power to 47 percent reactor power due to securing the 2P-8A heater drain pump and to address a main condenser tube leak that was causing high sodium levels above 50 ppb in both steam generators. On December 21, 2011, Unit 2 raised power to 80 percent reactor power after returning the 2P-8A heater drain pump to operation. On December 22, 2011, following repair of the condenser tubes, Unit 2 reached 100 percent reactor power and remained there for the remainder of the period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 <u>Readiness for Seasonal Extreme Weather Conditions</u>

a. <u>Inspection Scope</u>

The inspectors performed a review of the adverse weather procedures for seasonal extreme low temperature preparations. The inspectors verified that weather-related equipment deficiencies identified during the previous year were corrected prior to the onset of seasonal extremes, and evaluated the implementation of the adverse weather preparation procedures and compensatory measures for the affected conditions before the onset of, and during, the adverse weather conditions.

During the inspection, the inspectors focused on plant-specific design features and the procedures used by plant personnel to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Safety Analysis Report (SAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. Specific documents reviewed during this inspection are listed in the attachment. The inspectors also reviewed corrective action program items to verify that plant personnel were identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures. The inspectors' reviews focused specifically on the following plant systems:

- Unit 1 and Unit 2 emergency diesel generator fuel storage vaults

- Unit 1 and Unit 2 service water intake structures
- Unit 2 refuel water tank and Unit 1 borated water storage tank

These activities constitute completion of one (1) readiness for seasonal adverse weather sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- October 18, 2011, Unit 1 spent fuel pool cooling with temporary power modification

- October 19, 2011, Unit 2 service water bay A and bay C while bay B and the emergency cooling pond were unavailable

- November 2, 2011, alternate AC diesel generator and Unit 1 emergency diesel generator 2 while emergency diesel generator 1 was out of service for maintenance

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, SAR, technical specification requirements, administrative technical specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also inspected accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three (3) partial system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

On December 22, 2011, the inspectors performed a complete system alignment inspection of the Unit 2 fire water system to verify the functional capability of the system. The inspectors selected this system because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors inspected the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. The inspectors reviewed a sample of past and outstanding work orders to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program database to ensure that system equipment-alignment problems were being identified and appropriately resolved. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) complete system walkdown sample as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors conducted fire protection walkdowns that were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- October 17, 2011, Unit 1, Fire Zone FZ-1063 through FZ-1067 north and south, reactor building
- October 18, 2011, Unit 1, Fire Zone FZ-1030, service water intake structure during hot work

- December 30, 2011, Unit 2, Fire Zone 2200-MM, electrical switchgear, feedwater heaters and turbine area, elevation 386

- December 30, 2011, Unit 2 , Fire Zone 2076-HH, electrical equipment (motor generator set) room

The inspectors reviewed areas to assess if licensee personnel had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features, in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to affect equipment that could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's corrective action program. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four (4) quarterly fire-protection inspection samples as defined in Inspection Procedure 71111.05-05.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the SAR, the flooding analysis, and plant procedures to assess susceptibilities involving internal flooding; reviewed the corrective action program to determine if licensee personnel identified and corrected flooding problems; inspected underground bunkers/manholes to verify the adequacy of sump pumps, level alarm circuits, cable splices subject to submergence, and drainage for bunkers/manholes; and verified that operator actions for coping with flooding can reasonably achieve the desired outcomes. The inspectors also inspected the areas listed below to verify the adequacy of equipment seals located below the flood line, floor and wall penetration seals, watertight door seals, common drain lines and sumps, sump pumps, level alarms, and control circuits, and temporary or removable flood barriers. Specific documents reviewed during this inspection are listed in the attachment.

- December 19, 2011, Unit 1 and 2 emergency diesel generator fuel oil storage vaults during fire water deluge actuation

- December 22, 2011, manhole MH-9 and manhole MH-10,which contain two trains of Unit 1 emergency diesel generator fuel oil transfer pump electrical power, and manhole MH-4, which contains two trains of Unit 1 service water electrical power cables

- December 30, 2011, Unit 1 west decay heat vault

These activities constitute completion of two (2) flood protection measures inspection samples and one (1) bunker/manhole sample as defined in Inspection Procedure 71111.06-05.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the Unit 1 train B decay heat system heat exchanger. The inspectors verified that performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; the licensee utilized the periodic maintenance method outlined in EPRI Report NP 7552, "Heat Exchanger Performance Monitoring Guidelines"; the licensee properly utilized biofouling controls; the licensee's heat exchanger inspections adequately assessed the state of cleanliness of their tubes; and the heat exchanger was correctly categorized under 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) heat sink inspection sample as defined in Inspection Procedure 71111.07-05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Inspection Activities Other Than Steam Generator Tube Inspection, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, and Boric Acid Corrosion Control (71111.08-02.01)

a. Inspection Scope

The inspectors observed 16 nondestructive examination activities and reviewed five nondestructive examination activities that included seven types of examinations. The licensee did not identify any relevant indications accepted for continued service during the nondestructive examinations.

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Let-down Heat Exchanger, Drawing # 6600-M1J-3-7, Component ID # 37-005, Liquid Penetrant Exam # 1-ISI-PT-11-003	Solvent Soluble Contrasting Dye Penetrant Examination (PT)
Reactor Coolant Core Flood	Component ID: 1FCB-1 Piping, Description: FW-11C1, Drawing # CF-200, Liquid Penentrant Exam # 1-BOP-PT-11-012	Solvent Soluble Contrasting Dye Penetrant Examination (PT)
Reactor Coolant Core Flood	Component ID: 1FCB-1 Piping, Description: FW-12C1, Drawing # CF-200, Liquid Penentrant Exam # 1-BOP-PT-11-012	Solvent Soluble Contrasting Dye Penetrant Examination (PT)
Containment Building Spray System	Containment building spray valve and elbow, Drawing # 5-BS-1, Component ID # BS-4B, Radiograph Exam # 1-BOP-RT-11-016	Radiograph Examination (RT)
Reactor Coolant System	Steam Generator A, E-24A Lower Head to Lower Ring Head Weld. Drawing # M1D-295, Component # 03-102, Ultrasonic Exam # 1-ISI-UT-11-012	Ultrasonic Examination (UT)
Reactor Coolant System	Steam Generator A, E-24A Lower Head Ring to Lower Tubesheet Weld. Drawing # M1D-295, Component # 03-103, Ultrasonic	Ultrasonic Examination (UT)

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
	Exam # 1-ISI-UT-11-014	
High Pressure Injection System	Pipe to Elbow Circumference Seam. Drawing # 17-MU-27 Sheet 1. Component ID # 23-063, Ultrasonic Exam # 1-ISI-UT-11-008	Ultrasonic Examination (UT)
High Pressure Injection System	Pipe to Pipe Circumference Seam. Drawing # 17-MU-27 Sheet 1. Component ID # 23-107, Ultrasonic Exam # 1-ISI-UT-11-009	Ultrasonic Examination (UT)
Steam Generator	Letdown pipe, Elbow to Pipe Seam. Drawing # 17-MU-1 Sheet 2, Component ID # 24-009, Ultrasonic Exam # 1-ISI-UT-11-010	Ultrasonic Examination (UT)
Reactor Coolant System	Pressurizer Relief Nozzle Between Z-W Axis. Drawing # M1G-69, Component ID # 05-15IR, Visual Exam # 1-ISI-VT-11-034	Enhanced Visual Examination (VT-1)
Steam Generator	Steam Generator B Upper Head Manhole studs, washers, and nuts, Drawing # M1D-295 and M1D-251, Component ID # 03-120, Visual Exam # 1-ISI-VT-11-069	Visual Examination (VT-1)
Steam Generator	Steam Generator B Lower Head Manhole studs, washers, and nuts, Drawing # M1D-295 and M1D-251, Component ID # 03-119, Visual Exam # 1-ISI-VT-11-068	Visual Examination (VT-1)
Reactor Coolant System	Steam Generator Upper Primary Inspection Port (Hand Hold) Access E-24A, WO 244173-01, Drawing # M1D-295 (EC 2819), Component ID 3 6.4, Visual Exam # 1-ISI-VT-11-023	Visual Examination (VT-2)
Reactor Coolant	Steam Generator Upper Primary	Visual Examination (VT-2)

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
System	Inspection Port (Hand Hold) Access E-24B, WO 244951-01, Drawing # M1D-295 (EC 2819), Component ID # 6.7, Visual Exam # 1-ISI-VT-11-024	
Reactor Coolant System	Reactor lower head bottom mounted in-core instrumentation Alloy 600 bare metal inspection, Drawing # M-77 and M1B-231, Visual Exam # 1-ISI-VT-11-053	Enhanced Visual Examination (VT-2)
Service Water System	Spring Can Hanger HCD-111-H3. Drawing # 13-SW-110, Component # 54-059, Visual Exam # 1-ISI-VT-11-059	Visual Examination (VT-3)

The inspectors reviewed records for the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Main Steam System	E-24A Steam outlet nozzle to shell weld (@26 degrees), Drawing # M1D-295m Component # 03-117, Magnetic Particle Exam # 1-ISI-MT-11-001	Dry, Color Contrast, Magnetic Particle Examination (MT)
Containment Building Spray System	Containment building spray valve and elbow, Drawing # ISO 5-BS-1 and 5-BS-101, Component ID # BS-4B, Radiograph Exam # 1-BOP-RT-11-012	Radiograph Examination (RT)
Containment Building Spray System	Containment building spray valve and elbow, Drawing # ISO 5-BS-1 and 5-BS-101, Component ID # BS-4B, Radiograph Exam # 1-BOP-RT-11-013	Radiograph Examination (RT)
Containment Building Spray System	Containment building spray valve and elbow, Drawing # ISO 5-BS-1 and 5-BS-101, Component ID # BS-4B, Radiograph Exam # 1-BOP-RT-11-014	Radiograph examination (RT)
Reactor Coolant System	Reactor lower head bottom mounted in-core instrumentation	Enhanced Visual Examination (VT-2)

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
	Alloy 600 bare metal inspection, Drawing # M-77 and M1B-231, Visual Exam # 1-ISI-VT-11-053	

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with the ASME Code requirements and applicable procedures. The inspectors also verified the qualifications of all nondestructive examination technicians performing the inspections were current.

The inspectors observed two welds and reviewed the documentation on two welds on the reactor coolant system pressure boundary.

The inspectors directly observed a portion of the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD TYPE
Reactor Coolant Drain Tank	Component ID: 1FCB-1 Piping, Description: FW-12C1, Drawing # CF-200	Tungsten Inert Gas - GTAW
Reactor Building Spray	BS-4B - 8 inch,150 pound, tilting disc check valve, Drawing # M-236.	Tungsten Inert Gas - GTAW

The inspectors reviewed records for the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD TYPE
Reactor Coolant Drain Tank	Component ID: 1FCB-1 Piping, Description: FW-11C1, Drawing # CF-200	Tungsten Inert Gas - GTAW
Reactor Building Spray	BS-4B - 8 inch,150 pound, tilting disc check valve. Weld attaching 45° elbow to downstream pipe, Drawing # M-236.	Tungsten Inert Gas - GTAW

The inspectors verified, by review, that the welding procedure specifications and the welders had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also verified, through observation and record review, that essential variables for the welding process were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.01.

b. Findings

No findings were identified.

.2 Vessel Upper Head Penetration Inspection Activities (71111.08-02.02)

a. Inspection Scope

The inspectors reviewed the results of the licensee's bare metal visual inspection of the Reactor Vessel Upper Head Penetrations and verified that there was no evidence of boric acid challenging the structural integrity of the reactor head components and attachments. The inspectors also verified that the required inspection coverage was achieved and limitations were properly recorded.

These actions constitute completion of the requirements for Section 02.02.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control Inspection Activities (71111.08-02.03)

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 inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown as specified in Procedure EN-DC-319. The inspectors also reviewed the visual records of the components and equipment. The inspectors verified that the visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components. The inspectors also verified that the engineering evaluations for those components where boric acid was identified gave assurance that the ASME Code wall thickness limits were properly maintained. The inspectors confirmed that the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME Code. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements for Section 02.03.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities (71111.08-02.04)

a. Inspection Scope

Arkansas Nuclear One – Unit One (ANO1) replacement steam generators (1E-24 A&B) are Framatome Enhanced Once Through Steam Generators (EOTSG's). They were constructed in accordance with the 1989 ASME Boiler and Pressure Vessel Code, Section III. They are vertically mounted once through heat exchangers with a counter-flow design. They were installed during the Unit 1 Refueling Outage (1R19) in October 2005. The first inservice inspection was 1R20 in March 2007. During the 1R20 outage, it was identified that locking of the upper tube support plates to the upper shroud in Steam Generator A had occurred. This resulted in bowing of the tie rods in the first span (Condition Report CR-1-2007-959). A second inspection was performed in 1R21 which included both primary and secondary side inspections. The amount of tie rod bowing had increased as well as the number of tube support plate wear indications. However the growth rate of the wear supported skipping one outage. In the next outage, 1R22, only the tubes around the tie rods were inspected to assess the extent of the tie rod bowing only.

The inspection criteria for 1R23 (October 2011) included the following:

- 100 percent full length bobbin testing of both generators from tube end to tube.
- X-probe of tubes full length around all 52 tie rods in both steam generators.
- Plus Point/X probe testing of all proximity signals identified from Lower Tube Sheet to 01S, and all bobbin indications.
- Visual examination of the tube plugs – (10 tubes in Steam Generator A and 6 tubes in Steam Generator B).
- Diagnostic testing of all bobbin I-codes with the Plus Point/X-probe.
- Comparison of deposits based on X-probe data (Condition Report CR-ANO-1-2010-922). This was accomplished by testing the previously tested tubes in both Steam Generators (~ 69 tubes in Steam Generator A and 10 tubes in Steam Generator B).
- There were no secondary side visual inspections.
- X-probe of all tubes with tube-to-tube indications (proximity) due to tie rod bowing.

Results

There are two damage mechanisms currently associated with the Arkansas Nuclear One, Unit 1 steam generators. These include mechanical wear at the tube support plates and tie rod bowing which results in tube to tube contact during cold conditions. These will be addressed separately below:

Tube Support Plate Wear (Indications)

Steam Generator ID	Percent Through Wall 1-19 Percent	Percent Through Wall 20-39 Percent	Percent Through Wall 40-100 Percent
SG A	1456	36	0
SG B	1344	68	2

Maximum Depth was 46 Percent Through Wall (previous indication)
95/50 Growth = (~ 3 Percent Through Wall per Effective Full Power Year)

Plugging was performed at > 35 Percent to justify an interval equal to three cycles. All condition monitoring parameters were met and no in-situ testing was required.

Tie Rod Bowing

Historically, tie rod bowing was isolated to Steam Generator A only. The bowing is a result of the edges of the tube support plates being frictionally locked to the inner shroud during cool downs. During operation, the support plates go back to their free movement status and the rods straighten out. This is evident by no in-service tube to tube wear.

During the 1R23 inspection, bowing was identified in both steam generators. The extent of the bowing will be discussed below:

- Steam Generator A

  Refuel outage 1R21 was the fourth inspection where bowing had been identified. An operability evaluation was developed that addresses the projected curve of bowing based on the number of thermal cycles the unit experiences. Currently the unit has experienced six total thermal cycles. The operability was developed based on both laboratory testing of the tie rods and the support plates and various other analytical models. The maximum extent of the bowing is projected to be approximately 2.0 inches of lateral bow in the first span tie rods. The first span is defined by the area between the top of the lower tube sheet to the first support plate. This span has the longest vertical distance and the smallest diameter tie rods. Therefore the maximum extent of bowing is exhibited in the first span. Based on the projected curve, at six thermal cycles, the bowing could be as much as 1.6 inches of vertical bow. The actual results were consistent with the last inspection results of slightly below 1.3 inches. Steam Generator A has been consistent with the previous results and within the projected estimates in the operability.

  Two tubes in Steam Generator A, with tie rod bowing in the first span, display what appears to be geometric deformations just above the lower tube sheet, at the mid-

span (point of maximum bow) and just below the first tube sheet plate. The two tubes are Row 43 Tube 8 and Row 88 Tube 9. These deformations are seen by the various eddy current (ET) techniques (bobbin, array and +Point) as fill factor or lift-off variations with no evidence of tube wall loss.

The geometric indications in these two tubes are basically the same. There was one indication just above the lower tube sheet (LTS) which responds like a bulge; multiple indications at the mid-span which respond like a “wrinkled” area and one indication at the lower edge of the first tube sheet plate (01S) which responds like a dent.

Both tubes were removed from service.

- Steam Generator B

This was the first time that bowing had been identified in Steam Generator B. The extent of the bowing was approximately 0.5 inch which is well below that of Steam Generator A. There was a delta, in that the direction of the bowing in the first span in Steam Generator A was typically toward the center of the generator. The bowing in Steam Generator B is multi-directional. It is on the “X” side of the generator as compared to Steam Generator A which is on the “Z” side of the generator. This is being addressed through the condition report system under CR-ANO-1-2011-1925.

Repair:

The following tubes were repaired during the outage. There were seven in Steam Generator A and nine in Steam Generator B.

These actions constitute completion of the requirements of Section 02.04.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems (71111.08-02.05)

a. Inspection Scope

The inspectors reviewed 67 condition reports which dealt with inservice inspection activities and found the corrective actions for inservice inspection issues were appropriate. The specific condition reports reviewed are listed in the documents reviewed section. From this review, the inspectors concluded that the licensee has an appropriate threshold for entering inservice inspection issues into the Corrective Action Program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry inservice inspection operating experience. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of the requirements of Section 02.05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On November 29, 2011, the inspectors observed a Unit 2 crew of licensed operators in the plant's simulator to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance
- Crew's clarity and formality of communications
- Crew's ability to take timely actions in the conservative direction
- Crew's prioritization, interpretation, and verification of annunciator alarms
- Crew's correct use and implementation of abnormal and emergency procedures
- Control board manipulations
- Oversight and direction from supervisors
- Crew's ability to identify and implement appropriate technical specification actions and emergency plan actions and notifications

The inspectors compared the crew's performance in these areas to pre-established operator action expectations and successful critical task completion requirements. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) quarterly licensed-operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk significant systems:

- November 29, 2011, Alternate AC generator
- December 15, 2011, Unit 1 L-1 Turbine building crane
- December 22, 2011, Unit 2 emergency diesel generators
- December 30, 2011, Unit 1 reactor building spray

The inspectors reviewed events such as where ineffective equipment maintenance has resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65(b)
- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or -(a)(2)
- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1)

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four (4) quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed licensee personnel's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- October 25, 2011, Unit 1, 1R23 outage risk assessment
- November 11, 2011 Unit 1 and Unit 2, tornado warning with Unit 1 in mode 6 and Unit 2 at 100 percent power

The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two (2) maintenance risk assessments and emergent work control inspection samples as defined in Inspection Procedure 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Evaluations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- June 15, 2011, Unit 1 manhole MH-9 broken fire barrier inside manhole
- October 15, 2011, Unit 1, unplanned failure of “D” reactor protection system power supply
- December 7 and 19, 2011, Unit 1, removal of emergency switchgear room chillers, VCH- 4 A and B, from service for planned maintenance
- December 12, 2011, Unit 1, degraded in-core detector at IDC-32 level 1 that resulted in a quadrant power tilt exceeding the limit for operation above 60 percent reactor power

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that technical specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and SAR to the licensee personnel’s evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four (4) operability evaluations inspection sample(s) as defined in Inspection Procedure 71111.15-05.

b. Findings

Introduction. The inspectors identified a Green noncited violation of Unit 1 Technical Specification 3.8.4, “DC Sources-Operating,” Technical Specification 3.8.7, “Inverters-Operating,” and Technical Specification 3.8.9, “Distribution Systems-Operating,” due to the licensee’s failure to complete the associated required action prior to the specified completion time while the associated emergency switchgear room chillers were out of service for planned maintenance.

Description. On December 7, 2011, the licensee entered the following: (1) Technical Specification 3.7.7 Condition A for one loop of service water being inoperable with an associated completion time of 72 hours; (2) Technical Specification 3.8.1 Condition B for one emergency diesel generator inoperable with a 7 day completion time; and (3) Technical Specification 3.0.6 to support VCH-4A, Train B emergency switchgear room chiller, being out of service for planned maintenance. The licensee entered those Technical Specifications at 5:35 a.m. on December 7, 2011 and exited the respective technical specifications at 8:51 a.m. on December 8, 2011 after successful completion of

surveillance test procedure OP-1104.027, "Battery and Emergency Switchgear Cooling System," Revision 40 for the VCH-4A chiller. On December 19, 2011 at 3:19 a.m., the licensee entered the same technical specifications for the other loop of service water listed above to support VCH-4B, Train A emergency switchgear room chiller, being out of service for planned maintenance. The licensee exited those technical specifications at 6:47 p.m. on December 19, 2011 after successful completion of surveillance test procedure OP-1104.027 for the VCH-4B chiller.

The VCH-4 emergency switchgear chillers are non-technical specification equipment that support safety related equipment with associated technical specification requirements. Specifically, Technical Specification 3.8.4, "DC Sources - Operating," requires, in part, for one DC electrical power subsystem inoperable in Modes 1, 2, 3, or 4 for greater than 8 hours, action must be taken to place Unit 1 in Mode 3 within 12 hours. Technical Specification 3.8.7, "Inverters – Operating," requires, in part, that for two or more inoperable inverters in one of the two trains, while in Modes 1, 2, 3, or 4, action must be taken to place Unit 1 in Mode 3 within 12 hours. Technical Specification 3.8.9, "Distribution Systems – Operating," requires, in part, that for one AC, DC, or 120 VAC electrical power distribution subsystems inoperable in Modes 1, 2, 3, or 4 for greater than 8 hours, action must be taken to place Unit 1 in Mode 3 within 12 hours. Conversely, Technical Specification 3.7.7 for one loop of service water inoperable has a completion time of 72 hours. The issue was identified to the licensee and entered into the licensee's corrective action program as Condition Report CR-ANO-1-2012-0043.

Analysis. The inspectors determined that not completing the associated required actions for the appropriate technical specifications prior to the specified completion time while the associated emergency switchgear room chillers were out of service for planned maintenance is a performance deficiency. The performance deficiency is determined to be more than minor because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone, and affects the associated cornerstone objective to ensure availability, reliability, and the capability of systems that respond to initiating events to prevent undesirable consequences and is therefore a finding. Specifically, failure to complete the required actions prior to the specified completion times for Technical Specification 3.8.4, "DC Sources – Operating," Technical Specification 3.8.7, "Inverters – Operating, and Technical Specification 3.8.9, " Distribution Systems – Operating," due to removing the respective VCH-4 from service for maintenance, was a violation of technical specifications. Using Inspection Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding was determined to require a Phase 2 analysis because removing each VCH-4 chiller from service in December 2011 did result in an actual loss of safety function of a single train for greater than its technical specification allowed completion time. A phase 2 analysis from a previous noncited violation that bounds this issue determined the finding to be of very low safety significance (Green). Specifically, although the function was lost by the designated support equipment (emergency switchgear chillers), the licensee had an evaluation that credited compensatory measures and specific environmental conditions that assured the overall functionality of the applicable switchgear train was not lost. The inspectors reviewed the engineering change EC-25691, "Prepare EC markup to

CALC-92-E-0103-01 to determine maximum outside ambient temperatures and compensatory measures to allow one chiller train to cool DC/BATT/SWGR areas during maintenance," and determined the overall functionality of the applicable switchgear train was not lost, however, the compensatory measures sufficed for the function, but did not satisfy the technical specification switchgear operability requirements. The finding was determined to have a cross-cutting aspect in the area of human performance, associated with the decision making component, in that the licensee did not use conservative assumptions in decision making and adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement that it is unsafe in order to disapprove the action [H.1(b)].

Enforcement. Technical Specifications 3.8.4, "DC Sources - Operating," requires, in part, both DC electrical power subsystems shall be operable in Modes 1, 2, 3, or 4. Technical Specification 3.8.7, "Inverters – Operating," requires, in part, that two red train inverters and two green train inverters shall be operable in Modes 1, 2, 3, or 4. Technical Specification 3.8.9, "Distribution Systems – Operating," requires, in part, that two AC, DC, and 120 VAC electrical power distribution subsystems shall be operable in Modes 1, 2, 3, or 4. Technical Specification 3.8.4 and 3.8.9 require that if one DC electrical power subsystem, or one AC electrical distribution, or one DC electrical distribution, or one 120 VAC electrical power distribution subsystem is inoperable for greater than 8 hours, action must be taken to place Unit 1 in Mode 3 within 12 hours and Mode 5 within 36 hours. Technical Specification 3.8.7 requires that if two or more inverters are inoperable, Unit 1 must be placed in Mode 3 within 12 hours and Mode 5 within 36 hours. Contrary to the technical specification's required action statements, on December 7, 2011, the B train DC electrical power subsystem, the B train inverters, and the B train AC, DC, and 120 VAC electrical power distribution subsystems were inoperable due to a lack of emergency switchgear cooling for greater than the allowed completion time and the licensee failed to take the appropriate required actions. In addition, on December 19, 2011, the A train inverters were also inoperable due to a lack of emergency switchgear cooling and the unit was not placed in Mode 3 within the required 12 hours. Because this violation was of very low safety significance and has been entered into the corrective action program as Condition Report CR-ANO-1-2012-0043, this violation is being treated as a noncited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000313/2011005-01, "Exceeded Technical Specification Allowed Completion Time for Electrical Power Systems".

1R18 Plant Modifications (71111.18)

.1 Temporary Modifications

a. Inspection Scope

To verify that the safety functions of important safety systems were not degraded, the inspectors reviewed the following temporary modifications:

- November 3, 2011, Unit 1, temporary electrical power for spent fuel pool cooling P-40A

- November 15, 2011, Unit 2, reactor coolant system temperature T_{hot} input to core protection calculator C

The inspectors reviewed the temporary modifications and the associated safety-evaluation screening against the system design bases documentation, including the SAR and the technical specifications, and verified that the modification did not adversely affect the system operability/availability. The inspectors also verified that the installation and restoration were consistent with the modification documents and that configuration control was adequate. Additionally, the inspectors verified that the temporary modification was identified on control room drawings, appropriate tags were placed on the affected equipment, and licensee personnel evaluated the combined effects on mitigating systems and the integrity of radiological barriers.

These activities constitute completion of two (2) samples for temporary plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings were identified.

.2 Permanent Modifications

a. Inspection Scope

The inspectors reviewed key parameters associated with energy needs, materials, replacement components, timing, heat removal, control signals, equipment protection from hazards, operations, flow paths, pressure boundary, ventilation boundary, structural, process medium properties, licensing basis, and failure modes for the permanent modification identified as replacement of valve SW-9, and installation of valve SW-23 to provide boundary isolation from emergency cooling pond to service water.

The inspectors verified that modification preparation, staging, and implementation did not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; post-modification testing will maintain the plant in a safe configuration during testing by verifying that unintended system interactions will not occur; systems, structures and components' performance characteristics still meet the design basis; the modification design assumptions were appropriate; the modification test acceptance criteria will be met; and licensee personnel identified and implemented appropriate corrective actions associated with permanent plant modifications. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) sample for permanent plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following postmaintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- October 14, 2011, Unit 2 control element assembly trip circuit breakers

- November 1, 2011, Unit 1, control valve CV-1405, train A reactor building sump outlet valve following refurbishment

The inspectors selected these activities based upon the structure, system, or component's ability to affect risk. The inspectors evaluated these activities for the following:

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed

- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the technical specifications, the SAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with postmaintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two (2) postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the Unit 1 1R23 refueling outage, conducted October 16, 2011, through November 22, 2011, to confirm that licensee personnel had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense in depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

- Configuration management, including maintenance of defense in depth, is commensurate with the outage safety plan for key safety functions and compliance with the applicable technical specifications when taking equipment out of service.
- Clearance activities, including confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.
- Status and configuration of electrical systems to ensure that technical specifications and outage safety-plan requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes, systems, and components.
- Verification that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls, including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by the technical specifications.
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.

- Licensee identification and resolution of problems related to refueling outage activities.

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one (1) refueling outage and other outage inspection sample as defined in Inspection Procedure 71111.20-05.

b. Findings

(1) Failure to Implement Procedure Results in Lowering Spent Fuel Pool Level by 0.6 Feet

Introduction: The inspectors documented a Green, self-revealing, noncited violation of Unit 1 Technical Specification 5.4.1.a for the failure to implement station procedure OP-1104.006 “Spent Fuel Pool Cooling System”, Revision 51. Specifically, SF-10, flow control to purification loop valve, was found 3 turns open when it was required to be closed. This resulted in the spent fuel pool level lowering by 0.6 feet, which is below procedural limits, when the fuel transfer canal was placed in purification and SF-45, transfer tube isolation valve, was closed to support diving operations in the Unit 1 spent fuel pool tilt pit.

Description: On November 2, 2011, F-4A spent fuel filter was replaced and station procedure OP-1104.006 step 27.2 was performed to fill and vent the filter to place it back into service. During step 27.2.3.A, valve SF-10 was positioned to approximately 25 percent open to support the fill and vent of the filter. Later that day, operations performed step 24.0 of station procedure OP-1104.006 to place the fuel transfer canal and reactor cavity on purification. Prior to performing this operation, step 27.2.8.A.1 of OP-1104.006 directed the closing of valve SF-10 to prevent spent fuel pool cooling pump discharge water for cooling the pool from entering the spent fuel pool purification loop. Flow control valve SF-10 was not fully closed prior to placing the fuel transfer canal on purification.

On November 3, 2011, Unit 1 received a spent fuel pool low level alarm which is received when the pool level reaches -0.5 ft. At that time the fuel transfer canal was on purification in accordance with station procedure OP-1104.006 and SF-45 transfer tube isolation valve was closed to isolate the spent fuel pool tilt pit to support diving operations. Prior to closing SF-45, spent fuel pool level, as indicated by level indicator LI- 2004, was -0.3 ft. Operations secured fuel transfer purification and indicated level was -0.9 ft which was below the procedural limit of -0.5 ft. Operations then opened SF-45 which allowed water to sluice back to the spent fuel pool from the reactor cavity and the spent fuel pool level returned to -0.3 ft.

During investigation of the spent fuel pool low level alarm, operations determined that valve SF-10 was open approximately three turns. This allowed the spent fuel pool cooling pumps to pump water from the spent fuel pool to the suction piping for the decay heat removal pumps. The decay heat removal pumps were operating and pumped the water to the reactor coolant system and into the reactor cavity. When SF-45 was closed the water could not sluice back into the spent fuel pool from the reactor cavity.

Approximately 4,500 gallons of water in the spent fuel pool was transferred to the reactor cavity

The licensee identified that a purification valve line up was performed on November 2, 2011 prior to placing the fuel transfer canal on purification during which two operators checked SF-10 in the closed direction using normal force and verified closure by checking stem position that only showed threads. On November 3, 2011 when operators checked the position of SF-10 and found it to be open approximately three turns, they had to use excessive force including a torque amplifying device to close the valve.

The licensee performed a human performance error review in accordance with station procedure EN-HU-103 “Human Performance Error Reviews”, Rev. 6. The review determined that the condition of the valve not being closed was the result of degraded plant equipment and not the result of a human performance error.

Additional actions taken by the licensee included: (1) documenting that SF-10 requires a torque amplifying device to operate in CR-ANO-1-2011-2495, (2) hanging a caution card on SF-10 stating that a torque amplifying device is required to operate the valve, and (3) initiating a work request to address the valve condition.

<u>Analysis</u>: The failure of operations personnel to implement the requirements of procedure OP-1104.006, “Spent Fuel Pool Cooling System,” Revision 51, and close valve SF-10 is a performance deficiency. The performance deficiency is more than minor because it was associated with the configuration control attribute of the Barrier Integrity cornerstone and adversely affects the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events and is therefore a finding. Using Manual Chapter 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” the finding was determined to have very low safety significance (Green) because the finding did not result in the loss of spent fuel pool cooling, did not result from fuel handling errors that caused damage to the fuel clad integrity or a dropped assembly and did not result in a loss of spent fuel pool inventory of greater than 10 percent of the spent fuel pool volume. The finding was determined to have a cross-cutting aspect in the area of human performance, associated with the work control component in that the licensee failed to ensure that work activities to support long term equipment reliability limited operator work-arounds when a torque amplifying device was required to shut valve SF-10 [H.3(b)].

<u>Enforcement</u>: Technical Specification 5.4.1.a states, in part, that written procedures shall be implemented in accordance with Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 3.h, of Appendix A, “Procedures for Startup, Operation and Shutdown of Safety-Related PWR Systems,” requires procedures for operating the fuel storage pool purification and cooling system. Station procedure OP-1104.006, “Spent Fuel Cooling System”, Revision 51, step 27.2.8.A.1, stated to close valve SF-10 prior to returning the fuel transfer canal on purification after the completion of filling and venting spent fuel pool purification filter F-4A. Contrary to the above, valve SF-10 was not closed prior to placing the fuel transfer canal on purification causing spent fuel pool level

to decrease below procedural limits. Because this finding is of very low safety significance and has been entered into the corrective action program as Condition Report CR-ANO-1-2011-2498, this violation is being treated as a noncited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000313/2011005-02, "Failure to Implement Procedure Results in Lowering Spent Fuel Pool Level by 0.6 Feet."

(2) <u>Failure to Identify and Correct Unit 1 Service Water Pump Column Protective Wrap Installation Deficiencies.</u>

<u>Introduction</u>. The inspectors documented a Green, self-revealing, noncited violation of 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to promptly identify and correct a condition adverse to quality associated with degradation of the protective wrap (brand name – Denso) installed on the Unit 1 service water pump columns. The Denso protective wrap around the P-4C service water pump suction column became unraveled and was drawn into the service water pump suction while running, causing the pump to be secured due to pump discharge strainer high differential pressure.

<u>Description</u>. On November 15, 2011, during realignment of service water suction from the emergency cooling pond to the lake intake structure, the control room received a P-4C service water pump discharge strainer high differential pressure alarm. The alarm was received immediately after cross connecting the service water bays B and C via sluice gate 4. The discharge strainer differential pressure rose to at least 25 psid (maximum reading on the differential pressure instrument) and operations personnel manually placed the standby P-4B service water pump in service and secured the P-4C pump. At the time of the event, service water loop I was operable and being supplied from the P-4A pump and met the technical specification for service water supply for Unit 1 in Mode 6. Upon investigation, the licensee determined that the Denso protective wrap applied to the P-4C service water pump column, per Engineering Request, ER-963315E110 in 2005, had become unraveled and was pulled into the pump suction, resulting in debris that clogged the pump discharge strainer. The licensee entered the issue into the corrective action program as Condition Report CR-ANO-1-2011-2843. The licensee took immediate corrective action and removed the Denso protective wrap from all pump columns in the Unit 1 service water intake structure bays. Unit 2 does not have Denso protective wrap installed on their service water pumps.

The licensee performed an apparent cause evaluation that focused on the design change that installed the Denso protective wrap and determined that the design change should have integrated the following items into the design change: (1) the tape product was not specifically designed, qualified, or dedicated for nuclear application; (2) detailed engineering instructions for installation of the product should have been provided; (3) preventive maintenance requirements to identify degradation over time should have been developed; and (4) an estimate for the lifetime of the product in this application should have been determined. The apparent cause evaluation also identified that repeated occurrences of degradation since the original installation should have prompted numerous organizations to question the on-going integrity of the protective wrap applied to the pump columns in the Unit 1 intake structure bays.

The inspectors reviewed the apparent cause evaluation and identified three condition reports since 2005 that identified degradation of the Denso protective wrap applied to the P4-A and P-4C service water pumps. The latest Condition Report, CR-ANO-1-2011-0493 written on April 14, 2011, described two sections of Denso wrap that had peeled off and were hanging from the P-4C service water pump column. One piece was about 18 inches in length and the other section was split into two pieces of several inches each. The condition report only considered this issue as a long term corrosion concern and determined that it had no immediate impact on the pump operability. No evaluation was performed regarding the impact of additional unraveling of the Denso wrap to the service water pump's operability. The only corrective action performed was to immediately trim the loose pieces of Denso wrap to prevent further unraveling.

Analysis. The failure to promptly identify and correct a condition adverse to quality associated with degradation of the protective wrap (brand name – Denso) installed on the Unit 1 service water pump columns is a performance deficiency. The performance deficiency is determined to be more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affects the cornerstone objective to ensure availability, reliability, and the capability of systems that respond to initiating events to prevent undesirable consequences and is therefore a finding. The inspectors performed the significance determination for the failure of service water pump 4C using NRC Inspection Manual Chapter 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The problem had occurred during an outage, but it could have occurred at power during a system realignment. The at-power model was more conservative, so it was used to evaluate the finding. Service water was a two train system with a swing pump (an installed spare). The allowed outage time for one train was 72 hours. Operators could easily align the swing pump to provide the train B service water loads within 72 hours. Therefore, this finding screened to Green because: 1) it was not a design or qualification deficiency; 2) it did not result in loss of safety function of one train of equipment for more than its technical specification allowed outage time; 3) It did not result in a loss of one train of non-technical specification equipment; and 4) it did not screen as potentially risk significant due to an external event. The finding was determined to have a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component, in that, the licensee failed to thoroughly evaluate problems such that the resolutions address causes and extent of conditions. Specifically, the failure to thoroughly evaluate identified issues with the protective wrap prevented corrective action to be taken to prevent the deficiencies with the service water pump [P.1(c)].

Enforcement. Title 10 of the Code or Federal Regulations, Part 50, Appendix B, Criterion XVI states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformance's are promptly identified and corrected." Contrary to the above, from 2005 to November 15, 2011, the licensee failed to ensure that a known condition adverse to quality associated with the degradation of the Denso protective wrap, on the Unit 1 service water pumps, was thoroughly evaluated for continued degradation and/or corrected in a timely manner. Because this finding is of very low safety significance and has been entered into the corrective action program as

Condition Report CR-ANO-1-2011-2843, this violation is being treated as a noncited violation consistent with Section 2.3.3.a of the NRC Enforcement Policy: NCV 0500313/2011005-03, “Failure to Identify and Correct Unit 1 Service Water Pump Column Protective Wrap Installation Deficiencies.”

(3) Failure to Identify and Correct a Condition Adverse to Quality Resulted in Dropping a Fuel Bundle Approximately One Inch

Introduction. The inspectors identified a Green, noncited violation of 10 CFR 50, Appendix B, Criterion XVI for failure to identify and correct a condition adverse to quality. Specifically, on November 1, 2011, the licensee failed to identify and correct a condition associated with seating an irradiated fuel bundle into a reactor building storage location during core re-loading activities. The licensee failed to thoroughly evaluate a discrepancy associated with an unexpected vertical measurement when inserting an irradiated fuel bundle into a reactor building storage location. This resulted in the bundle dropping 1 1/8 inches at the storage location.

Description. On November 1, 2011, the licensee was reloading the Unit 1 reactor core and was experiencing some difficulty inserting an irradiated fuel bundle, NJ0C12, into core location A-10. At this time the reactor building was open to the atmosphere. The refueling team decided to move the fuel bundle to the reactor building storage rack C, while attempting to adjust fuel bundles surrounding core location A-10. The ZZ-tape, (the vertical measuring system used for fuel bundle placement) indicated that the irradiated fuel bundle was at 32 feet and 1/4 inch. This measurement was 1 5/8 inch higher than the nominal reading for this location. The refueling team raised and set the fuel bundle down again and obtained the same ZZ-tape measurement. The nominal reading was noted as 31 feet 10 and 5/8 inch in Attachment J, “Main Fuel Bridge (H-1) Fuel Hoist ZZ Tape Readings and Weight Setpoints,” of procedure OP-1502.003, “Refueling Equipment and Operator Checkouts,” Revision 35. The table also stated an allowable tolerance of ±1/2 inch difference between the current ZZ-tape reading and the nominal readings obtained during fuel handling. System and reactor engineering were notified for resolution.

In an attempt to verify that the fuel bundle was fully seated, the refueling team used an underwater camera to inspect and evaluate the top portion of the bundle and the storage rack. A visual comparison was performed with a smooth side dummy bundle two storage locations away. The refueling team did not visually identify any height difference. Reactor engineering, without going to the refueling bridge, approved the as found ZZ-tape measurement as the current vertical measurement.

After fuel bundle adjustments were made in the reactor core, the refueling team went to the storage rack to retrieve fuel bundle NJ0C12 to load it into core location A-10. When the grapple was lowered onto the assembly, the bundle dropped approximately 1 1/8 inches in the storage rack. An immediate visual inspection did not identify any obvious damage. The licensee decided not to use the bundle. An evaluation later performed by AREVA determined that, although there was no visual damage to the bundle, the fuel pellets may have been damaged due to the 11g of force experienced as

a result of the drop. Another bundle was identified for use and a new core design was developed and approved.

The licensee performed a lower tier apparent cause evaluation, which determined that no human performance errors were involved in this event, and their apparent cause for the dropped bundle was an inadequate procedure that failed to give specific guidance to move fuel bundles to the reactor building storage racks and to verify that the fuel bundle is fully seated in the storage rack. The licensee did not address any human performance issues associated with this event.

The inspectors determined that the licensee failed to identify that the fuel bundle was not fully seated in the storage location, and failed to correct that condition prior to un-grappling. The inspectors also determined that the licensee failed to thoroughly evaluate the discrepancy between the vertical fuel bundle measurement and the expected nominal measurement. The refueling team did attempt to verify fuel bundle position with an underwater camera, but incorrectly compared the heights of the smooth sided dummy bundle, which is shorter in height, and the fuel bundle. The licensee failed to look at the bottom of the storage location for confirmation that the bundle was fully seated. The refueling team did not note any ZZ-tape vertical measurement discrepancies with any other locations, nor did they review any measurement data to rule out any issue with the ZZ-tape. The licensee also incorrectly assumed that the reactor building storage racks on Unit 1 (Babcox & Wilcox) were designed the same as the Unit 2 (Combustion Engineering) storage racks. The Unit 1 storage racks have a cruciform on the bottom of the rack to help align and seat the fuel bundle. The licensee did not thoroughly evaluate the fuel bundle measurement, convinced themselves that the ZZ-tape discrepancy was acceptable and decided to accept the discrepancy. The inspectors determined that the discrepancy associated with the ZZ-tape should have placed the issue into the corrective action program, but was not placed into the program until the bundle was dropped.

<u>Analysis</u>. The inspectors determined that the failure to identify and correct the condition associated with the incorrect placement of an irradiated fuel bundle into a reactor building storage location, is a performance deficiency because the licensee failed to place the nuclear fuel in a safe position. The performance deficiency is determined to be more than minor because it is associated with the human performance attribute of the Barrier Integrity cornerstone and adversely affects the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the performance deficiency resulted in a dropped bundle that caused the bundle to be removed from service due to possible fuel pellet damage. The event also took place during core reloading activities, in which the reactor building was open to the atmosphere. Using Manual Chapter 0609, Appendix G, Attachment 1, Checklist 4, “PWR Refueling Operation: RCS Level >23’,” the finding was determined to have very low safety significance (Green) because the finding did not adversely affect: 1) core heat removal, 2) inventory control, 3) electrical power, 4) containment control, or 5) reactivity control. The finding was determined to have a cross-cutting aspect in the area of human performance, associated with the decision making component in that the licensee failed to use conservative assumptions and adopt a requirement to demonstrate that the proposed action is safe in order to

proceed when deciding to accept the discrepancy in the vertical measurement when storing a fuel bundle in the reactor building storage rack [H.1(b)].

Enforcement. Title 10 of the Code of Federal Regulation Part 50, Appendix B, Criterion XVI, "Corrective Action," 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 non-conformances are promptly identified and corrected." Contrary to the above, on November 1, 2011, the licensee failed to identify and correct a condition adverse to quality regarding the placement of a fuel bundle in a storage location when confronted with evidence that the fuel bundle may not have been fully seated in that location. The fuel bundle subsequently dropped 1 1/8 inches. The drop was of sufficient force to render the bundle unusable due to possible fuel pellet damage concerns. Because this finding is of very low safety significance and has been entered into the corrective action program as Condition Report CR-ANO-1-2012-0110, this violation is being treated as a noncited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000313/2011005-04, "Failure to Identify and Correct a Condition Adverse to Quality Resulted in Dropping a Fuel Bundle Approximately One Inch."

(4) Failure to Take Adequate Corrective Actions for Known Fuel Transfer System Deficiencies

Introduction. The inspectors documented a Green, self-revealing finding for the failure to take adequate corrective actions for known deficiencies associated with the Unit 1 fuel transfer system. Specifically, the licensee failed to investigate and correct issues that have been identified by site and vendor personnel from 1996 through 2010. This led to repeated fuel transfer system failures and significant core offload and reload delays during the 1R23 refueling outage, which placed the plant in an unplanned configuration for an extended period of time.

Description. During the most recent Unit 1 refueling outage 1R23, fall 2011, numerous problems associated with the fuel transfer system caused an interruption of fuel transfer activities while offloading and reloading the reactor. Beginning on October 23, 2011, while unloading the core, the refueling team began to experience fuel transfer carriage overloads while moving fuel from the reactor building to the spent fuel pool, on every other fuel transfer. Eventually the overloads became more frequent, occurring on every fuel transfer until the overload condition could not be cleared and caused fuel transfer activities to be stopped. The licensee subsequently identified worn carriage wheels and cable tension issues as contributing to the overload conditions. The issues were temporarily remedied, but cable tension issues remained. On October 27 reactor core offload was completed, but the fuel transfer system continued to experience overload conditions. No corrective actions were taken during the core defueled window to address the overload issue.

On November 1, 2011, reactor core reload began. Cable tension was being monitored and was increasing and continued with every fuel bundle transfer. At the time, the licensee did not know why the cable tension was increasing, but later determined that some increase in tension should have been expected and that actions could have been

taken to mitigate the issue. On November 2, the fuel transfer carriage unexpectedly stopped approximately three feet inside the reactor building. The licensee had 69 of 177 fuel bundles loaded into the core at this time. The licensee formed a failure modes analysis team to further investigate the issue. It was determined that the fuel transfer carriage wheels on the North side of the carriage were riding up on top of the railing system in the reactor building. The licensee first attempted to realign the rails on the spent fuel pool side to better align the carriage as it transitioned into the reactor building. This action was not effective and the misalignment persisted. A temporary modification was developed and installed that added a wheel extension to the reactor building side of the fuel transfer carriage to prevent the carriage from riding up on top of the rails. On November 13 core reload was completed.

The current fuel transfer system was not original equipment and was installed in 1986. Beginning in 1996, issues associated with the fuel transfer system have been noted. In 2004, 2005, 2007, 2008, and again in 2010 issues with overloads, worn wheels, sheaves, mechanical binding and even a broken retraction cable had been documented in vendor (AREVA) outage reports and in the licensee's corrective action program. Inspections and pre-outage fuel system checkouts were performed prior to 1R23 outage and did identify some overload conditions, but they were attributed to not having calibrated the load cell. An inspection of the fuel transfer system was performed under water and without moving the fuel transfer carriage. The refueling team further directed unloaded and dry check runs of the fuel transfer system. Nothing was identified from this inspection.

The inspectors reviewed the licensee's root cause evaluation. The root cause evaluated why the fuel transfer system experienced overloads and equipment deficiencies that resulted in the loss of 200 hours of critical path time. Three root causes were identified: 1) original design and configuration issues, 2) organizational issues such as communication, direction of field activities, application of field resources, and decision making that was inadequate during the 1R23 refueling outage, and 3) that previous vendor reports and operating experience items were not acted upon in a timely manner to correct historical problems. The inspectors believe that the main cause for the fuel transfer issues experienced in 1R23 was the failure to correct known deficiencies that have been plaguing the licensee for years. The root cause further evaluated safety culture aspects associated with this issue and concluded that several safety culture aspects were applicable. Among these were decision making, corrective action program for failing to correct the deficiencies, and the failure to act upon operating experience. The inspectors determined that the safety culture aspect of non-conservative decision making was the most dominate contributor to not correcting known deficiencies. Specifically, the decision making efforts affecting the fuel transfer system did not reflect a safety minded culture as past experience and vendor recommendations were disregarded.

<u>Analysis</u>. The failure of the licensee to take effective corrective action for known deficiencies related to the Unit 1 fuel transfer system is determined to be a performance deficiency because it was not in accordance with their corrective action program, was within their ability to foresee and correct, and should have been corrected. The performance deficiency is determined to be more than minor because if left uncorrected

the performance deficiency could become a more safety significant issue. Specifically, the licensee's failure to correct known deficiencies of the fuel transfer system demonstrated a lack of knowledge of the system design and function which could fail in unexpected and in unpredictable ways which could lead to more safety significant issues. Using Manual Chapter 0609, Appendix G, Attachment 1, Checklist 4, "PWR Refueling Operation: RCS Level >23'," the finding was determined to have very low safety significance (Green) because the finding did not adversely affect: 1) core heat removal, 2) inventory control, 3) electrical power, 4) containment control, or 5) reactivity control. The finding was determined to have a cross-cutting aspect in the area of human performance, associated with the decision making component in that the licensee failed to use conservative assumptions and adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. Specifically, the decision making efforts affecting the fuel transfer system did not reflect a safety minded culture as past experience and vendor recommendations were disregarded [H.1(b)].

Enforcement. Although a performance deficiency was identified, there were no violations of NRC requirements identified during the review of this issue because the Unit 1 fuel transfer system is not safety-related. The licensee entered this issue into the corrective action program as Condition Report CR-ANO-1-2011-2558. This finding is being documented as: FIN 05000313/2011005-05, "Failure to Take Adequate Corrective Actions for Known Fuel Transfer System Deficiencies."

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the SAR, procedure requirements, and technical specifications to ensure that the surveillance activities listed below demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning
- Evaluation of testing impact on the plant
- Acceptance criteria
- Test equipment
- Procedures
- Jumper/lifted lead controls
- Test data

- Testing frequency and method demonstrated technical specification operability
- Test equipment removal
- Restoration of plant systems
- Fulfillment of ASME Code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for returning tested systems, structures, and components not meeting the test acceptance criteria were correct
- Reference setting data
- Annunciators and alarms setpoints

The inspectors also verified that licensee personnel identified and implemented any needed corrective actions associated with the surveillance testing.

- October 5, 2011, Unit 1 VCH-4A, loop 2 emergency switchgear room chiller temperature switch surveillance test
- November 9, 2011, Unit 1, make up and purification system check valve and control valve full flow inservice surveillance test
- November 10, 2011, Unit 1, fill and vent of makeup and purification, and the high pressure injection system (TI 2515/177 effort)
- November 11, 2011, Unit 1, train A engineered safeguards actuation system integrated test
- November 18, 2011, Unit 1, pressurizer sampling system containment isolation valve SV-1818 local leak rate test
- December 19, 2011, Unit 2 containment isolation valve 2CV-4823-2 local leak rate test

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six (6) surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector performed an in-office review of Emergency Plan Implementing Procedure OP-1903.010, "Emergency Action Level Classification," Change 44 submitted by letter dated July 26, 2011. This revision changed a reference in Attachment 9, "EAL Equipment Compensating Measures," of this procedure from referencing a table in the Technical Requirements Manual listing seismic instrumentation to referencing Procedures 1203.025 and 2203.008, "Natural Emergencies," for Units 1 and 2 respectively, were the compensating measures are specified for seismic instrumentation.

This revision was compared to its previous revision, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, and to the standards in 10 CFR 50.47(b) to determine if the revision adequately implemented the requirements of 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection.

These activities constitute completion of one (1) sample as defined in Inspection Procedure 71114.04-05.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed a Unit 1 simulator training evolution for licensed operators on November 22, 2011, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the postevolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the corrective action program. As part of the inspection, the inspectors reviewed the scenario package and other documents listed in the attachment.

These activities constitute completion of one (1) sample as defined in Inspection Procedure 71114.06-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the performance indicator data submitted by the licensee for the third Quarter 2011 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 0608, "Performance Indicator Program."

This review was performed as part of the inspectors' normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - Emergency ac Power System (MS06)

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index - emergency ac power system performance indicator for Units 1 and 2 for the period from the fourth quarter 2010 through the third quarter 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, mitigating systems performance index derivation reports, issue reports, event reports, and NRC integrated inspection reports for the period of October 2010 through September 2011 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of two (2) mitigating systems performance index - emergency ac power system samples as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index - High Pressure Injection Systems (MS07)

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index - high pressure injection systems performance indicator for Units 1 and 2 for the period from the fourth quarter 2010 through the third quarter 2011. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, issue reports, mitigating systems performance index derivation reports, event reports, and NRC integrated inspection reports for the period of October 2010 through September 2011 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of two (2) mitigating systems performance index - high pressure injection system samples as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. The inspectors reviewed attributes that included the complete and accurate

identification of the problem; the timely correction, commensurate with the safety significance; the evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews; and the classification, prioritization, focus, and timeliness of corrective actions. Minor issues entered into the licensee's corrective action program because of the inspectors' observations are included in the attached list of documents reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure, they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. The inspectors accomplished this through review of the station's daily corrective action documents.

The inspectors performed these daily reviews as part of their daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused their review on repetitive equipment issues, but also considered the results of daily corrective action item screening discussed in Section 4OA2.2, above, licensee trending efforts, and licensee human performance results. The inspectors nominally considered the 6-month period of June 2011 through December 2011 although some examples expanded beyond those dates where the scope of the trend warranted.

The inspectors also included issues documented outside the normal corrective action program in major equipment problem lists, repetitive and/or rework maintenance lists,

departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's corrective action program trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

These activities constitute completion of one (1) single semi-annual trend inspection sample as defined in Inspection Procedure 71152-05.

b. Findings and Observations

No findings were identified. The inspectors did identify the following items during the review: 1) configuration control issues, 2) water intrusion issues into the auxiliary building, turbine building, and manholes; and 3) outage performance with regards to the refueling team performance and refueling equipment. These items have been entered into the corrective action program.

.4 Selected Issue Follow-up Inspection

a. Inspection Scope

During a review of items entered in the licensee's corrective action program, the inspectors recognized a corrective action report documenting an incident where an operator found a diesel oil storage tank outlet valve closed that was required to be open to support the functionality of the alternate AC diesel generator. The licensee entered the issue into the corrective action program as Condition Report CR-ANO-C-2011-2241. The inspectors reviewed the condition report for impact upon the diesel's functionality and the high risk significance associated with potential loss of functionality of the alternate AC diesel generator.

These activities constitute completion of one (1) in-depth problem identification and resolution sample as defined in Inspection Procedure 71152-05.

b. Findings

Introduction: The inspectors documented a Green, self revealing, noncited violation of Unit 1 Technical Specification 5.4.1.a for the failure to implement station procedure OP-1015.049 "Configuration Control Program", Revision 1. Specifically, on multiple occasions, station personnel failed to maintain configuration control through the use of valve line-ups and station procedures to ensure that plant components were in required positions.

Description: On September 3, 2011, Unit 1 outside auxiliary operator discovered FO-37, diesel oil storage tank outlet valve, closed when it was required to be open to supply fuel oil to the alternate AC diesel generator 600 gallon day tank. This condition would have prevented automatic makeup to the day tank but the alternate AC diesel would have started and run when demanded for approximately 1.5 hours. The licensee determined

that FO-37 was not correctly positioned open on August 24, 2011, while performing Attachment B of station operating procedure OP-1104.023, “Diesel Oil Transfer Procedure” during the performance of maintenance on the Unit 1, train A emergency diesel generator. The licensee entered the issue into their corrective action program as Condition Report CR-ANO-C-2011-2241.

On October 18, 2011, while collapsing the pressurizer bubble per station procedure OP-1103.011, reactor vessel level indication became erratic and indicated a low level condition. Draining was secured to evaluate the condition. After securing the drain it was noted that pressurizer level continued to lower and the quench tank volume continued to rise. After investigation it was determined that RBV-71B, T hot loop B root vent was open when it should have been closed. This caused an unintended reactor coolant system loss of approximately 525 gallons to the quench tank during the pressurizer bubble collapse effort. The licensee determined that RBV-71B was not closed as required per station procedure OP-1103.002, Attachment B, “Valve Lineup after completion of Fill and Vent” at the completion of the previous outage. The licensee entered the issue into their corrective action program as condition report CR-ANO-1-2011-1740 and 1744.

On October 19, 2011, while performing station procedure OP-1104.004, Attachment G, “Decay Heat Coolant Purification Using Alternate Purification,” station chemistry personnel determined that the reactor coolant system was not getting cleaner based on the results of the demineralizer effluent sample and this indicated that there was no flow through the demineralizer. Following an investigation, it was determined that valves CZ-33 and CZ-34B were closed and should have been open and valve CZ-35B was open and should have been closed as required by station procedure OP-1104.004. The mispositioned valves allowed the reactor coolant system flow to bypass the demineralizer. The licensee entered the issue into their corrective action program as Condition Report CR-ANO-1-2011-1812.

On October 23, 2011, the licensee used station procedure OP-1104.002, “Makeup and Purification System Operation”, Supplement 8 to perform a full flow check valve test of the makeup system using the A high pressure injection pump. It was determined that the pump curve data obtained was out of the IST limiting range for the pump. An investigation determined that the equalizing valve for PDT-1210 D, high pressure flow indication, was open one-half turn and caused flow indication to read lower. The valve did not have the hand wheel installed and was operated with channel locks. During the subsequent retest the valve was again found three turns open following its operation to flush the lines. The valve was finally replaced with a new one with a hand wheel. The licensee entered the issue into their corrective action program as Condition Report CR-ANO-1-2011-2312.

The inspectors reviewed Condition Report CR-ANO-C-2011-2942, and its associated apparent cause evaluation relating to 12 potential mispositioned components since June 2011. The evaluation concluded that the causes included: (1) a lack of commitment to program implementation; (2) documents not followed correctly involving

both programmatic and component control document usage; and (3) guidance was not well defined or understood.

Based upon the multiple examples of failures to satisfy station configuration control procedures the inspectors have determined the failures to be indicative of a programmatic failure to position plant components as required per the configuration control program.

Analysis: The inspectors determined that the failure of station personnel to maintain configuration control through the use of valve line-ups and governing station procedures to ensure reactor plant components were in their required positions, is a performance deficiency. The performance deficiency is more than minor because it is associated with the configuration control attribute of the Mitigating Systems cornerstone and adversely affects the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences and is therefore a finding. Using Manual Chapter 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” the finding included an example that was determined to be an actual loss of safety function of a non-technical specification train of equipment designated as risk-significant per 10CFR50.65, for greater than 24 hours. A phase 3 significance determination analysis was performed by a Region IV senior reactor analyst. The dominant core damage sequences for Unit 1 were station blackouts with battery depletion and transients with loss of feedwater and feed and bleed capability. The dominant core damage sequences for Unit 2 were station blackout with loss of emergency feedwater and once-through-cooling, loss of 4160 volt vital bus 2A4 with loss of feedwater and once-through-cooling, and station blackout with an 8-hour battery depletion. Based on both units having the capability to operate a steam driven emergency feedwater pump during the dominate core damage sequences the finding was determined to have very low safety significance (Green). The finding was determined to have a cross-cutting aspect in the area of human performance, associated with the work practices component to support human performance in that the licensee failed to define and effectively communicate expectations regarding procedural guidance and personnel follow procedures when performing component positioning in accordance with the licensee’s program for configuration control [H.4(b)].

Enforcement: Technical Specification 5.4.1.a states, in part, that written procedures shall be established, implemented and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 1 of Appendix A to Regulatory Guide 1.33, states in part, that safety related activities should be covered by written procedures such as “equipment control.” Station procedure OP-1015.049, “Configuration Control Program”, Revision 1, step 6.1 stated the control of plant equipment status is established by performing valve/breaker line-ups and then governed by procedures, work orders, log readings, or protective tagging. Contrary to the above, on multiple occasions, between September 3 and October 23, 2011, the licensee failed to control plant equipment status by inappropriately performing valve/breaker line-ups and for failing to follow governing procedures. Because this finding is of very low safety significance and has been entered into the corrective action program as Condition Report CR-ANO-1-2011-2942, this violation is being treated as a

noncited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000313/2011005-06, “Failure to Adequately Implement the Configuration Control Program.”

.5 In-depth Review of Operator Workarounds

a. Inspection Scope

The inspectors selected this issue for review to verify that licensee personnel were identifying operator workaround problems at an appropriate threshold and entering them in the corrective action program, and has proposed or implemented appropriate corrective actions. The inspectors reviewed and evaluated the licensee's operator workaround log, for both Units 1 and 2, operator logs and associated condition reports. The inspectors considered the following, as applicable, 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.

b. Findings

No findings were identified.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) LER 05000368/2009003 Steam Generator Tube Exceeding Technical Specification Plugging Criteria Remained in Service During Previous Cycles as a Result of the Failure to Use Proper Independent Verification

On September 8, 2009, Unit 2 was shutdown in Mode 6 for 2R20 outage activities. During the ‘B’ steam generator inspection it was discovered that a steam generator tube was incorrectly plugged during the previous outage. During the 2R17 spring outage a steam generator tube with an identified flaw was correctly plugged on the cold leg side of the steam generator but not on the hot leg side. An adjacent steam generator tube on the hot leg side was incorrectly plugged instead. The condition resulted in Unit 2 operating at power, from April 2005 until discovery, with a steam generator tube characterized with an approximate 43 percent through wall defect which was in violation of the Unit 2 Technical Specification of less than 40 percent through wall required to be in service. Licensee investigation determined that the error in plugging was caused by a failure to use proper independent verification that the correct tube was plugged. Both steam generator tubes were plugged to remove them from service. The issue was placed into the corrective action program as Condition Report CR-ANO-2-2009-2357. A licensee identified noncited violation was documented in Inspection Report 05000368/2009004 for this issue. This licensee event report is closed.

.2 <u>(Closed) LER 05000368/2009002</u> Containment Building Penetration Isolation Valves Open During Core Alterations without Application of Administrative Controls Required by Technical Specifications Due to Inadequate Procedural Instructions

On September 7, 2009, with Unit 2 in Mode 6 for refueling, licensed operators discovered that containment penetration isolation valves located on the return line of the containment atmospheric monitoring system were configured such that a direct path existed between the containment atmosphere and the auxiliary building atmosphere and the resulting containment breech was not being administratively controlled as required by Unit 2 technical specifications. The licensee determined that the system was initially placed in the correct configuration during reactor shutdown, but a local leak rate testing evolution required these vales to be repositioned. The valves were not restored to the required configuration following completion of the local leak rate testing. Core alteration commenced shortly after completion of the testing. The licensee determined that the local leak rate procedure failed to give adequate guidance to restore the system for shutdown plant conditions. The licensee took corrective action to modify the procedure to specify position of the valves depending on the plant mode. The issue was placed into the corrective action program as Condition Report CR-ANO-2-2009-2329. A licensee identified noncited violation was documented in Inspection Report 05000368/2009004. This licensee event report is closed.

.3 <u>(Closed) LER 05000368/2009004</u> Emergency Diesel Automatic Actuation While Performing Offsite Power Transfer Testing Due to a High Resistance Contact Supplying Voltage to a Synchronizing Check Relay

On September 20, 2009, Unit 2 was shutdown in Mode 5 for 2R20 outage activities. During the performance of planned surveillance testing of the Offsite Power Transfer Test, the 2K-4A emergency diesel generator automatically started. An Offsite Power Transfer Test was being performed to test automatic transfer from the Startup 3 Offsite Transformer to the Startup 2 Offsite Transformer. During the Offsite Power Transfer Test, a permissive contact in the Startup 2 feeder breaker failed resulting in a slow transfer to the 2A1 bus instead of the expected fast transfer. The slow transfer resulted in a momentary loss of power to the 4160 Volt Safety Electrical Bus 2A3 which is powered from 2A1. The momentary undervoltage condition on 2A3 caused the 2K-4A emergency diesel generator to auto start as designed. The 2K-4A emergency diesel generator did not power 2A3, since 2A3 was successfully powered from 2A1 after the slow transfer completed. During the momentary loss of power, 2A3 automatically shed all loads as designed. This load shed caused the running shutdown cooling pump, 2P-60A , to secure which resulted in a loss of shutdown cooling flow to the reactor coolant system for approximately three and one half minutes. The licensee determined that the cause of the event was a loss of one of the voltage inputs that feed the 2A1 bus synchronizing check relay (125-111), located in the 2A-111 breaker cubicle, due to a high resistance contact. This high resistance condition blocked one of the voltage inputs to the synchronizing check relay, causing the relay to falsely indicate that the startup 2 transformer and the 2A1 bus were not synchronized. The licensee took immediate corrective action to modify the circuit with alternate contacts with the appropriate resistance. The licensee also took corrective action to modify the maintenance procedures for these type breakers to inspect and maintain these contacts. The issue

was placed into the corrective action program as Condition Reports CR-ANO-2-2009-2997. A self-revealing noncited violation was documented in Inspection Report 05000368/2009005 for this issue. The review of this licensee event report is complete and no findings were identified and no violations of NRC requirements occurred. This licensee event report is closed.

4OA5 Other Activities

(Open) NRC TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems (NRC Generic Letter 2008-01)"

As documented in Section 1R22, the inspectors confirmed the acceptability of the licensee's procedures and processes for filling and venting ECCS systems. This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later NRC Inspection Report following further inspection activities to follow-up on previously identified issues documented in inspection report ANO 2011-04.

4OA6 Meetings

Exit Meeting Summary

On October 28, 2011, the inspectors presented the inspection results of the review of inservice inspection activities to Mr. C. Schwarz, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On December 1, 2011, the inspector, during a telephonic meeting, discussed the results of the in-office inspection of changes to the licensee's emergency plan and emergency action levels to Mr. R. Holeyfield, Manager, Emergency Preparedness, and other members of the licensee's staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On January 20, 2012, the inspectors presented the inspection results to Mr. M. Chisum, General Manager, Plant Operations, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section 2.3.2 of the NRC Enforcement Policy for being dispositioned as noncited violations.

- Unit 1 Technical Specification 5.4.1.a, requires, in part, that "Written procedures shall be established, implemented, and maintained covering the following activities…the

applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Regulatory Guide 1.33, Revision 2, Appendix A, Section 2 specifies written procedures for the safety-related activity of refueling and core alterations. Contrary to the above, the licensee failed to implement procedures for core alterations during 1R23 Unit 1 refueling outage. Specifically, on two occasions, the refueling team failed to follow refueling procedures for verifying neutron counts prior to un-grappling a fuel bundle in the core and for moving a fuel bundle in fast speed prior to obtaining adequate clearance from other fuel bundles in the core. Using Manual Chapter 0609, Appendix G, Attachment 1, Checklist 4, "PWR Refueling Operation: RCS Level >23'," the finding was determined to have very low safety significance (Green) because the finding did not adversely affect: 1) core heat removal; 2) inventory control; 3) electrical power; 4) containment control; or 5) reactivity control. These issues were entered into the corrective action program as Condition Reports CR-ANO-1-2011-2085, and 2552.

- Unit 1 Technical Specification 5.4.1.a, requires, in part, that "Written procedures shall be established, implemented, and maintained covering the following activities…the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Regulatory Guide 1.33, Revision 2, Appendix A, Section 2 specifies written procedures for the safety-related activity of refueling and core alterations. Contrary to the above, the licensee failed to provide adequate procedures for refueling and core alterations during 1R23 Unit 1 refueling outage. Specifically, the licensee over rotated a control rod drive lead screw during reactor disassembly and resulted in having to replace the control rod drive mechanism. Using Manual Chapter 0609, Appendix G, Attachment 1, Checklist 4, "PWR Refueling Operation: RCS Level >23'," the finding was determined to have very low safety significance (Green) because the finding did not adversely affect: 1) core heat removal, 2) inventory control, 3) electrical power, 4) containment control, or 5) reactivity control. This issue was entered into the corrective action program as Condition Report CR-ANO-1-2011-1921.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

C. Schwarz, Site Vice President
D. Bice, Licensing Specialist
B. Byford, Manager, Training
T. Chernivec, Manager, Outages
M. Chisum, General Manager, Plant Operations
B. Daiber, Manager, Design Engineering
A. Dodds, Manager, Maintenance
M. Farmer, Maintenance, Refueling Program Manager
R. Fowler, Senior Emergency Preparedness Planner
R. Fuller, Manager, Quality Assurance
W. Greeson, Manager, Engineering Programs and Component
R. Holeyfield, Manager, Emergency Preparedness
R. Holman, Welding Engineer, Entergy Code Programs
D. Hughes, Manager (Acting), Engineering Programs and Component
K. Jones, Manager, Operations
B. Lovin, Manager, Security
D. Marvel, Manager, Radiation Protection
J. McCoy, Director, Engineering
R. McGaha, NDE Technician, Entergy Code Programs
D. Metheany, Steam Generator Programs Owner
N. Mosher, Licensing Specialist
B. Pace, Manager, Planning, Scheduling, and Outage
K. Panther, Manager, ISI Program
D. Perkins, Manager, Maintenance
S. Pyle, Manager, Licensing
T. Sherrill, Manager, Chemistry
P. Williams, Manager, System Engineering

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000313/2011005-01	NCV	Exceeded Technical Specification Allowed Completion Time for Electrical Power Systems (Section 1R15)
05000313/2011005-02	NCV	Failure to Implement Procedure Results in Lowering Spent Fuel Pool Level by 0.6 Feet (Section 1R20(1))
05000313/2011005-03	NCV	Failure to Identify and Correct Unit 1 Service Water Pump Column Protective Wrap Installation Deficiencies (Section 1R20(2))
05000313/2011005-04	NCV	Failure to Identify and Correct a Condition Adverse to Quality

Opened and Closed

		Resulted in Dropping a Fuel Bundle Approximately One Inch (Section 1R20(3))
05000313/2011005-05	FIN	Failure to Take Adequate Corrective Actions for Known Fuel Transfer System Deficiencies (Section 1R20(4))
05000313/2011005-06	NCV	Failure to Adequately Implement the Configuration Control Program (Section 4OA2.4)

Closed

05000368/2009003	LER	Steam Generator Tube Exceeding Technical Specification Plugging Criteria Remained in Service During Previous Cycles as a Result of the Failure to Use Proper Independent Verification
05000368/2009002	LER	Containment Building Penetration Isolation Valves Open During Core Alterations without Application of Administrative Controls Required by Technical Specifications Due to Inadequate Procedural Instructions
05000368/2009004	LER	Emergency Diesel Automatic Actuation While Performing Offsite Power Transfer Testing Due to a High Resistance Contact Supplying Voltage to a Synchronizing Check Relay

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

PROCEDURES

NUMBER	TITLE	REVISION
OP-1104.039	Plant Heating and Cold Weather Operations	22
OP-2106.032	Unit 2 Two Freeze Protection Guide	22

Section 1R04: Equipment Alignment

PROCEDURES

NUMBER	TITLE	REVISION
OP-1104.029	Unit 1 Service and Auxiliary Cooling Water System	55
OP-1104.036	Unit 1 Emergency Diesel Generator Operation	59
OP-2104.037	Alternate AC Diesel Generator	22
OP-1104.032	Unit 1 Fire Protection Systems	68
OP-2104.032	Unit 2 Fire Protection Systems Operations	32

DRAWINGS

NUMBER	TITLE	REVISION
M-210	Service Water	150
M-217	Emergency Diesel Generator and Fuel Oil System	89
M-2241	Alternate AC Generator System	3
M-2219 Sheet 1	Fire Water System Pipe and Instrument Diagram	61
M-2219 Sheet 2	Fire Water System Pipe and Instrument Diagram	69
M-2219 Sheet 4	Deluge Valve Detail	35
M-2219 Sheet 5	Outside Fire Loop	50
M-2219 Sheet 7	Deluge Valve Detail	15

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
STM-1-42	Unit 1 Service and Auxiliary Cooling Water	20
STM-1-31	Unit 1 Emergency Diesel Generators	12
STM-2-33	Unit 2 Alternate AC Diesel Generator	21

Section 1R05: Fire Protection

PROCEDURES

NUMBER	TITLE	REVISION
FHA	ANO Fire Hazard Analysis	13
PFP-U1	ANO Pre-Fire Plan Unit 1	13
PFP-U2	ANO Pre-Fire Plan Unit 2	10

DRAWINGS

NUMBER	TITLE	REVISION
FZ-1063	Unit 1 Fire Zone Detail – Reactor Building	3
FZ-1064	Unit 1 Fire Zone Detail – Reactor Building	3
FZ-1065	Unit 1 Fire Zone Detail – Reactor Building	3
FZ-1066	Unit 1 Fire Zone Detail – Reactor Building	3
FZ-1067	Unit 1 Fire Zone Detail – Reactor Building	3
FZ-2044	Unit 2 Fire Zone Detail – Electrical Switchgear, Feedwater Heaters, and Turbine areas	1
FZ-2025	Unit 2 Fire Zone Detail – Electrical Equipment (motor generator sets) room	2
FZ-1030	Unit 1 Service Water Intake Structure	2

Section 1R06: Flood Protection Measures

PROCEDURES

NUMBER	TITLE	REVISION
ULD-0-TOP-17	ANO Topical Flooding	0

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
CALC-92-R-0024-01	Flooding Evaluation INPO SOER-85-5	0
CALC-92-R-0034-01	Flooding Evaluation INPO SOER-85-5 2nd Iteration	
ULD-1-SYS-01	ANO Unit 1 Emergency Diesel Generator	5
ULD-0-TOP-02	Fire Protection Topical	4

CONDITION REPORTS

CR-ANO-1-2011-1343 CR-ANO-C-2011-0802 CR-ANO-1-2011-0744 CR-ANO-1-2011-0662
CR-ANO-1-2001-0661 CR-ANO-1-2011-0641

Section 1R07: Heat Sink Performance

PROCEDURES

NUMBER	TITLE	REVISION
OP-1309.016	Decay Heat Thermal Test	004-01-0

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
ER-91-R-2013-01	Service Water Performance Testing Methodology	21

CONDITION REPORTS

CR-ANO-1-2011-2134 CR-ANO-1-2011-2014 CR-ANO-1-2011-1750 CR-ANO-1-2011-1712

Section 1R08: Inservice Inspection Activities

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	Entergy Steam Generator Degradation Assessment: Plant and Unit – Arkansas Nuclear One Unit One, Refueling Outage: 1R23	0
	Snapshot Assessment / Benchmark On: Pre-NRC Inspection – In-service Inspection (ISI) 1R23	August 31, 2011
	Quarterly Health Reports 4Q2010, 1Q2011, 2Q2011, 3Q2011	

Section 1R08: Inservice Inspection Activities

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	1R20 Cycle Report	Spring 2007
1032.037	Inspection And Identification Of Boric Acid Leaks For ANO-1 and ANO-2	5
1103.013	RCS Leak Detection	35
1CAN060902	Request for Alternative – Implementation of a Risk-Informed Inservice Inspection Program Based on ASME Code Case N-716, Arkansas Nuclear One, Unit 1, Docket No. 50-313 License No. DPR-51	June 11, 2009
1CNA030901	Arkansas Nuclear One, Unit 1, Grand Gulf Nuclear Station, River Bend Station, and Waterford Steam Electric Station, Unit 3 - Request for Alternative CEP-ISI-012, Use Alternative Requirements In ASME Code Case N-753 (TAC NOS. MD8813, MD8814, MD8815 AND MD8816)	March 6, 2009
1CNA061001	Arkansas Nuclear One, Unit No. 1 -Request For Alternative AN01-ISI-014 Re: Implementation Of a Risk-Informed Inservice Inspection Program Based on ASME Code Case N-716 (TAC No. ME1488)	June 2, 2010
20004-017	ENGINEERING INFORMATION RECORD, Document No.: 51 - 9135783 – 000, Technical Summary of Steam Generator Eddy Current Examinations at Arkansas Nuclear One, 1R22	March 2010
51-9135783-000	Areva NP Inc, Engineering Information Record, Technical Summary of Steam Generator Eddy Current Examinations at Arkansas Nuclear One, 1R22.	March 2010
CNRO-2008-00016	Relief Requests for Third 120 Month Inservice Testing Interval	May 20, 2008
EN-DC-319	Inspection and evaluation of Boric Acid Leaks	7
EN-DC-319	Inspection and evaluation of Boric Acid Leaks	6
LO-ALO-2008-00090	Boric Acid Corrosion Control Program (BACCP) Self Assessment	August 13, 2009
LO-ALO-2010-00056	Assessment Report: Welding Program Assessment	August 2011

NDE PROCEDURES

NUMBER	TITLE	REVISION
CEP-NDE-0255	Radiographic Examination ASME, ANSI,AWS Welds and Components	6
CEP-NDE-0400	Ultrasonic Examination	3
CEP-NDE-0404	Manual Ultrasonic Examination of Ferritic Piping Welds (ASME XI)	5
CEP-NDE-0407	Straight Beam Ultrasonic Examinations of Bolts and Studs (ASME XI)	3
CEP-NDE-0423	Manual Ultrasonic Examination of Austenitic Piping Welds (ASME XI)	5
CEP-NDE-0497	Manual Ultrasonic Examination of Welds in Vessels (Non-App. VIII)	5
CEP-NDE-0641	Liquid Penetrant Examination (PT) for ASME Section XI	7
CEP-NDE-0731	Magnetic Particle Examination (MT) for ASME Section XI	3
CEP-NDE-0901	VT-1 Examination	4
CEP-NDE-0902	VT-2 Examination	7
CEP-NDE-0903	VT-3 Examination	5

CONDITION REPORTS

CR-ANO-1-2011-02807	CR-ANO-1-2011-02789	CR-ANO-1-2011-00554	CR-ANO-1-2010-00956
CR-ANO-1-2010-00968	CR-ANO-1-2010-01986	CR-ANO-1-2011-00685	CR-ANO-1-2010-01983
CR-ANO-1-2010-00977	CR-ANO-1-2010-02009	CR-ANO-1-2011-00753	CR-ANO-1-2011-00512
CR-ANO-1-2010-01118	CR-ANO-1-2010-02021	CR-ANO-1-2011-00872	CR-ANO-1-2010-01966
CR-ANO-1-2010-01124	CR-ANO-1-2010-02055	CR-ANO-1-2011-00909	CR-ANO-1-2011-00318
CR-ANO-1-2010-01295	CR-ANO-1-2010-02071	CR-ANO-1-2011-01126	CR-ANO-1-2010-01948
CR-ANO-1-2010-01361	CR-ANO-1-2010-02073	CR-ANO-1-2011-01379	CR-ANO-1-2011-02736
CR-ANO-1-2010-01462	CR-ANO-1-2010-02089	CR-ANO-1-2011-01380	CR-ANO-1-2011-00250
CR-ANO-1-2010-01475	CR-ANO-1-2010-02087	CR-ANO-1-2011-01395	CR-ANO-1-2010-01933
CR-ANO-1-2010-01493	CR-ANO-1-2010-02173	CR-ANO-1-2011-01489	CR-ANO-1-2011-02258
CR-ANO-1-2010-01564	CR-ANO-1-2010-02197	CR-ANO-1-2011-01728	CR-ANO-1-2011-00157
CR-ANO-1-2010-01587	CR-ANO-1-2011-02213	CR-ANO-1-2011-01824	CR-ANO-1-2010-01930
CR-ANO-1-2010-01613	CR-ANO-1-2010-02218	CR-ANO-1-2011-01895	CR-ANO-1-2011-02224
CR-ANO-1-2010-01644	CR-ANO-1-2010-02516	CR-ANO-1-2011-01926	CR-ANO-1-2011-00034
CR-ANO-1-2010-01716	CR-ANO-1-2010-02605	CR-ANO-1-2011-01979	CR-ANO-1-2010-01922
CR-ANO-1-2010-01754	CR-ANO-1-2010-02734	CR-ANO-1-2011-01998	CR-ANO-1-2011-02213
CR-ANO-1-2010-01802	CR-ANO-1-2010-02736	CR-ANO-1-2011-02071	CR-ANO-1-2010-03760
CR-ANO-1-2010-01810	CR-ANO-1-2010-02900	CR-ANO-1-2011-02084	CR-ANO-1-2010-01907
CR-ANO-1-2010-01823	CR-ANO-1-2010-03617	CR-ANO-1-2011-02128	CR-ANO-1-2011-02173

CONDITION REPORTS

CR-ANO-1-2010-01856 CR-ANO-1-2010-03754

Section 1R12: Maintenance Effectiveness

PROCEDURES

NUMBER	TITLE	REVISION
EN-DC-203	Maintenance Rule Program	1
EN-DC-204	Maintenance Rule Scope and Basis	2
EN-DC-205	Maintenance Rule Monitoring	3
EN-DC-206	Maintenance Rule (a)(1) Process	1
ULD-0-TOP-19	Upper Level Document Station Blackout	0
OP-2104.037	Alternate AC Diesel Generator Operations	21

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
	Maintenance Rule Database Scoping and Performance Criteria – Unit 1 Alternate AC diesel generator	October 12, 2011
	Unit 1 Alternate AC diesel generator Functional Failure Determination Report	October 12, 2011
	Maintenance Rule Database Scoping and Performance Criteria – Unit 1 Turbine Building	November 15, 2011
	Unit 1 Turbine Building Functional Failure Determination Report	November 15, 2011
	Unit 1 Reactor Building Spray – Maintenance Rule Database Scoping and Performance Criteria	November 28, 2011
	Unit 1 Reactor Building Spray – Maintenance Rule Functional Failure Determination Report	November 28, 2011

CONDITION REPORTS

CR-ANO-C-2011-1639 CR-ANO-C-2011-1971 CR-ANO-C-2011-1862 CR-ANO-1-2011-0567
CR-ANO-C-2011-0061 CR-ANO-1-2011-1617 CR-ANO-1-2011-2075 CR-ANO-1-2011-0588
CR-ANO-1-2011-0999

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

PROCEDURE

NUMBER	TITLE	REVISION
OP-1203.025	Natural Emergencies	35

CONDITION REPORTS

CR-ANO-C-2011-2952

Section 1R15: Operability Evaluations

PROCEDURES

NUMBER	TITLE	REVISION
EN-OP-104	Operability Evaluations	5
OP-1105.001	Unit 1 Nuclear Instrumentation and Reactor Protection System Operating Procedure	25

CALCULATIONS

NUMBER	TITLE	REVISION
CALC-ANO1-NE-11-00002	ANO Unit 1 Cycle 24 Core Operating Limits Report	3

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
STM-1-63	Unit 1 Reactor Protection System	9

CONDITION REPORTS

CR-ANO-1-2011-1655	CR-ANO-1-2011-1659	CR-ANO-1-2011-1667	CR-ANO-1-2010-3653
CR-ANO-1-2011-1672	CR-ANO-1-2011-3044	CR-ANO-1-2011-3183	CR-ANO-1-2011-0896

Section 1R18: Plant Modifications

PROCEDURES

NUMBER	TITLE	REVISION
EN-DC-115	Engineering Change Process	12
EN-DC-136	Temporary Modifications	6

ENGINEERING CHANGE DOCUMENTS

EC-31408 EC-30016

WORK ORDERS

00277055 00279037

Section 1R19: Postmaintenance Testing

PROCEDURES

NUMBER	TITLE	REVISION
OP-2304.039	Unit 2 Plant Protection System Channel C Test	47
OP-1305.007	RB Isolation and Miscellaneous Valve Stroke Test	39
EN-MA-101	Fundamentals of Maintenance	9
EN-WM-102	Work Implementation and Closeout	6
EN-WM-105	Planning	9
EN-WM-107	Post Maintenance Testing	3

WORK ORDERS

50271508 52326209

Section 1R20: Refueling and Other Outage Activities

PROCEDURES

NUMBER	TITLE	REVISION / DATE
1-OPG-002	Unit 1 Tank Volume Book	April 5, 2011
OP-1104.006	Unit 1 Spent Fuel Cooling System	51
OP-1506.001	Fuel and Control Component Handling	41
OP-1502.004	Control of Unit 1 Refueling	49

Section 1R20: Refueling and Other Outage Activities

PROCEDURES

NUMBER	TITLE	REVISION / DATE
OP-1502.003	Refueling Equipment and Operator Checkouts	35
OP-1103.011	Draining and N_2 Blanketing the RCS	39
EN-OM-123	Fatigue Management Program	3

CONDITION REPORTS

CR-ANO-1-2011-2495	CR-ANO-1-2011-2498	CR-ANO-1-2011-2843	CR-ANO-C-2011-3017
CR-ANO-1-2011-0493	CR-ANO-1-2005-1405	CR-ANO-1-2010-0370	CR-ANO-1-2011-2558
CR-ANO-1-2011-2211	CR-ANO-1-2011-2814	CR-ANO-1-2011-2815	CR-ANO-1-2010-1028
CR-ANO-1-2011-2412	CR-ANO-1-2011-0769	CR-ANO-1-2011-1846	

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
STM-1-51-1	Refueling Machine & Reactor Bldg Fuel Handling Equipment	4
STM-1-51-2	Spent Fuel Handling & SFP Area Equipment	10
STM-1-51-3	Fuel Transfer System	2

Section 1R22: Surveillance Testing

PROCEDURES

NUMBER	TITLE	REVISION
OP-1305.018	Unit 1 Local Leak Rate Testing – Type C	23
OP-1305.006	Unit 1 Integrated Engineered Safeguards System Test	35
OP-1104.002	Unit 1 Makeup and Purification System Operation	72
OP-1104.027	Unit 1 Battery and Switchgear Emergency Cooling System	40
OP-2305.017	Local Leak Rate Testing	28

CONDITION REPORTS

CR-ANO-1-2011-1882	CR-ANO-1-2011-2660	CR-ANO-1-2011-2783	CR-ANO-2-2011-0820
CR-ANO-1-2011-2757	CR-ANO-1-2011-2021	CR-ANO-1-2011-2526	CR-ANO-2-2011-0800
CR-ANO-1-2011-2312	CR-ANO-1-2011-2316	CR-ANO-1-2011-2524	CR-ANO-1-2011-2700
CR-ANO-1-2011-2516	CR-ANO-1-2011-2130		

WORK ORDERS

52274060

Section 1EP4: Emergency Action Level and Emergency Plan Changes

PROCEDURES

NUMBER	TITLE	REVISION
OP-1903.011P	SAE Emergency Direction and Control Checklist Shift Manager	42
OP-1903.011Y	Emergency Class Initial Notification Message	40

Section 4OA1: Performance Indicator Verification

PROCEDURES

NUMBER	TITLE	REVISION
EN-LI-114	Performance Indicator Process	4

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
	Unit 1 MSPI Derivation Report – Emergency AC Power System – Unavailability Index	October 28, 2011
	Unit 1 MSPI Derivation Report – Emergency AC Power System – Unreliability Index	October 28, 2011
	Unit 1 Emergency Diesel Generator 1 Conditional Probability Data	October 28, 2011
	Unit 1 Emergency Diesel Generator 2 Conditional Probability Data	October 28, 2011
	Unit 1 MSPI Derivation Report – High Pressure Injection System – Unavailability Index	October 28, 2011
	Unit 1 MSPI Derivation Report – High Pressure Injection System – Unreliability Index	October 28, 2011
	Unit 1 Makeup and Purification 1P36A Pump Conditional Probability Data	November 30, 2011
	Unit 1 Makeup and Purification 1P36B Pump Conditional Probability Data	November 30, 2011
	Unit 1 Makeup and Purification 1P36C Pump Conditional Probability Data	November 30, 2011

Section 4OA2: Identification and Resolution of Problems

PROCEDURES

NUMBER	TITLE	REVISION
COPD-001	Operations Expectations and Standards	55
COPD-020	ANO Operations Concerns Program	10
EN-FAP-OP-006	Operator Aggregate Impact Index Performance Indicator	6
OP-2304.258	Unit 2 Escape Airlock Leak Rate Test	17
OP-2305.017	Local Leak Rate Testing	26
OP-2411.029	Emergency Air Lock Inspection, Lubrication and Chalk Test	5
OP-1015.001	Conduct of Operations	89
OP-1015.049	Configuration Control Program	1
OP-1103.002	Draining and Nitrogen Blanketing the Reactor Coolant System	41
OP-1103.011	Filling and Venting the Reactor Coolant System	37
OP-1104.004	Decay Heat Removal Operating Procedure	94
OP-1104.002	Unit 1 Makeup and Purification System Operation	72

DRAWINGS

NUMBER	TITLE	REVISION
DWG 30970	Emergency Access Airlock – General Arrangement	0
DWG 30970	Emergency Access Airlock – General Assembly	0

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
	Nuclear Oversight Fleet Trimester Report	October 2011
	Unit 1 Top Ten Reliability Issues	
	Unit 2 Top Ten Reliability Issues	

CONDITION REPORTS

CR-ANO-2-2011-0888	CR-ANO-2-2011-1197	CR-ANO-2-2011-1687	CR-ANO-2-2011-3264
CR-ANO-2-2011-0768	CR-ANO-C-2011-2241	CR-ANO-C-2011-2942	CR-ANO-2-2011-3170
CR-ANO-1-2011-1740	CR-ANO-1-2011-1744	CR-ANO-1-2011-1851	CR-ANO-2-2011-3294

CONDITION REPORTS

CR-ANO-1-2011-1812	CR-ANO-1-2011-2312	CR-ANO-1-2011-2498	CR-ANO-2-2011-2696
CR-ANO-1-2007-1667	CR-ANO-1-2011-0328	CR-ANO-1-2010-2370	CR-ANO-2-2011-3533
CR-ANO-1-2009-0014	CR-ANO-1-2011-0967	CR-ANO-1-2011-1666	CR-ANO-2-2011-2263
CR-ANO-1-2011-1797	CR-ANO-1-2011-2145	CR-ANO-1-2011-2319	CR-ANO-2-2011-2166
CR-ANO-1-2011-3049	CR-ANO-1-2011-3077	CR-ANO-1-2011-0858	CR-ANO-2-2011-2179
CR-ANO-1-2011-3070	CR-ANO-2-2008-2360	CR-ANO-2-2009-0176	CR-ANO-2-2011-1663
CR-ANO-2-2011-3250	CR-ANO-2-2009-3566	CR-ANO-2-2010-0923	CR-ANO-2-2011-1687
CR-ANO-2-2010-0056	CR-ANO-2-2011-0103	CR-ANO-2-2011-0644	CR-ANO-2-2011-1343
CR-ANO-2-2011-0924	CR-ANO-2-2011-1318	CR-ANO-2-2011-1411	