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

February 2, 2006

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

SUBJECT: ARKANSAS NUCLEAR ONE - NRC INTEGRATED INSPECTION REPORT 05000313/2005005 AND 05000368/2005005

Dear Mr. Forbes:

On December 31, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Arkansas Nuclear One, Units 1 and 2, facility. The enclosed integrated report documents the inspection findings, which were discussed on January 10, 2006, with you and other members of your staff.

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

The report documents one inspector identified and two self-revealing findings, all of which were determined to be of very low safety significance (Green). Two of these findings were determined to involve violations of NRC requirements; however, because the findings were entered into your corrective action program, the NRC is treating these violations as noncited violations consistent with Section VI.A of the Enforcement Policy. Additionally, a licensee identified violation which was determined to be of very low safety significance is listed in this report. If you contest these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at Arkansas Nuclear One, Units 1 and 2, facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

David N. Graves, Chief
Project Branch E
Division of Reactor Projects

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

Enclosure: NRC Inspection Report 05000313/2005005 and 05000368/2005005
w/attachment: Supplemental Information

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SUNSI Review Completed: ___DNG___ ADAMS: / Yes ☐ No Initials: _DNG__
/ Publicly Available ☐ Non-Publicly Available ☐ Sensitive / Non-Sensitive

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RIV:RI:DRP/E	RI:DRP/E	SRI:DRP/E	C:DRS/PSB
JLDixon	ELCrowe	RWDeese	MPShannon
E-DNGraves	T-DNGraves	T-DNGraves	/RA/
1/19/06	2/2/06	2/2/06	2/2/06

C:DRS/OB	C:DRS/EMB	C:DRS/PEB	C:DRP/E
TGody	JAClark	LJSmith	DNGraves
/RA/	/RA/	DLProulx	/RA/
2/2/06	1/31/06	2/2/06	2/2/06

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-313, 50-368
Licenses:	DPR-51, NPF-6
Report:	05000313/2005005 and 05000368/2005005
Licensee:	Entergy Operations, Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	Junction of Hwy. 64W and Hwy. 333 South Russellville, Arkansas
Dates:	September 24 through December 31, 2005
Inspectors:	E. Crowe, Resident Inspector R. Deese, Senior Resident Inspector J. Dixon, Resident Inspector G. Guerra, Jr., CHP, Health Physicist J. Kirkland, Project Engineer R. Lantz, Senior Emergency Preparedness Inspector W. McNeill, P.E., Reactor Inspector, Engineering Branch 1 C. Stancil, Project Engineer N. Taylor, Resident Inspector, Cooper Nuclear Station
Accompanying Personnel:	J. Keeton, Consultant
Approved By:	David N. Graves, Chief, Project Branch E Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000313/2005005, 05000368/2005005; 9/24/05 - 12/31/05; Arkansas Nuclear One, Units 1 and 2; Operability Evaluations, Identification and Resolution of Problems, and Access Control to Radiologically Significant Areas

This report covered a 3-month period of inspection by resident inspectors and regional specialist inspectors. Three Green findings, two of which were noncited violations 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." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management's review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria III, for the failure to include provisions to assure that appropriate quality standards were specified, and that deviations from such standards were controlled. As a result, non-safety grade components were installed in the high pressure injection, low pressure injection, and reactor building spray systems. A walk down of the high pressure injection pumps revealed that one temperature element appeared to be bent significantly more than the others. Further investigation revealed that the temperature elements were nonsafety grade (affected high and low pressure injection systems). Additionally, one temperature element was missing its protective sheath which was not in accordance with its design. The installed optomatic oilers and piping connections were also determined to be nonsafety grade (affected high and low pressure injection and reactor building spray systems). Since these components are part of the lube oil system boundary, they should have been classified as safety grade components.

  The inspectors determined that the failure to utilize safety-related components in safety-related systems, and the temperature element missing the protective sheath (not in accordance with design), was a performance deficiency. This finding was more than minor because it affected the design control attribute under the Mitigating Systems Cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. Using the significance determination process the issue was determined to have very low safety significance because the finding did not result in a loss of function per Part 9900 Technical Guidance, "Operability Determination Process for Operability and Functional Assessment," did not represent an actual loss of safety function, and is not potentially risk significant due to external events (Section 1R15).

Cornerstone: Barrier Integrity

- Green. The inspectors reviewed a self-revealing finding which occurred when a Unit 2 steam generator developed a tube leak (February 2005). A metallic piece of foreign material fretted a hole in one steam generator tube and wore away some thickness of two others. The licensee identified several more pieces of foreign material after conducting more thorough searches in both of the Unit 2 steam generators. The licensee performed a thorough review of the event to determine the short and long term corrective actions.

  This issue is more than minor because it affected the reactor coolant system barrier performance attribute under the Barrier Integrity Cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The finding was determined to be of very low safety significance after management review, because the affected tubes could have withstood three times the differential pressure across them during normal full power, steady state operation (Section 4OA2).

Cornerstone: Occupational Radiation Safety

- Green. The inspector reviewed a self-revealing noncited violation of Technical Specification 6.7.1.a because the licensee failed to control a high radiation area by not barricading and conspicuously posting the area. Specifically, on March 15, 2005, the licensee removed a temporary barrier (scaffold boards) creating an entrance to a high radiation area without the proper radiological controls in place for a high radiation area. It was not until two radiation workers entered the area that a radiation protection technician identified the unposted entry and took appropriate actions to control the area. The finding was entered into the licensee's corrective action program as Condition Report ANO-2-2005-0574.

  The failure to control a high radiation area as per Technical Specification requirements is a performance deficiency. The finding is greater than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of program and process and affected the cornerstone objective, to ensure the adequate protection of the worker health and safety from exposure to radiation, in that not controlling high radiation areas could increase worker exposure. The finding was evaluated using the Occupational Radiation Safety Significance Determination Process and is of very low safety significance because it does not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. In addition, this finding has crosscutting aspects associated with human performance because poor coordination and communication between the scaffold crew and radiation protection personnel directly contributed to the finding (Section 2OS1).

B. Licensee-Identified Violations

A violation of very low safety significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at 100 percent rated thermal power and remained there until October 4, 2005, when the unit shut down for an outage to refuel and to replace the reactor vessel head and steam generators (SGs). The unit was restarted on December 21, 2005, and achieved 95 percent rated thermal power on December 25, 2005. The unit was holding at 95 percent rated thermal power to resolve emergency feedwater (EFW) initiation and control indication problems when on December 26, 2005, the unit tripped due to a turbine trip on low turbine bearing pressure. The unit was restarted on December 29, 2005, and achieved 95 percent rated thermal power on December 31, 2005, and remained there for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent rated thermal power and remained there throughout the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

Readiness for Impending Adverse Weather Conditions. The inspectors completed a review of the licensee's readiness for impending adverse weather involving icy conditions. The inspectors: (1) reviewed plant procedures, the Updated Safety Analysis Report, and Technical Specifications to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) walked down portions of the systems listed below to ensure that adverse weather protection features were sufficient to support operability including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to ensure the licensee would maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program (CAP) to determine if the licensee identified and corrected problems related to adverse weather conditions.

- December 7, 2005, Unit 1 service water, fire protection, and condensate storage tank systems; Unit 2 service water, fire protection, and condensate storage systems; and the common alternate ac diesel generator systems.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

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

- November 23, 2005, Unit 2 Emergency Diesel Generator 2 during planned surveillance testing of Emergency Diesel Generator 1.
- December 6, 2005, Units 1 and 2 safety-related electrical buses during Startup Transformer 1 maintenance
- December 14, 2005, Unit 1 decay heat system following complete core reload

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

Fire Protection Tours: The inspectors walked down the six plant areas listed below to assess the material condition of active and passive fire protection features, their operational lineup, and their operational effectiveness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features; and (7) reviewed the CAP to determine if the licensee identified and corrected fire protection problems.

- October 5, 2005, Unit 1 Fire Zone 105-T lower south electrical penetration room
- November 25, 2005, Unit 2 Fire Zone 2154-E control element drive mechanism equipment room

- November 25, 2005, Unit 2 Fire Zone 2093-P south emergency diesel generator room

- December 6, 2005, Unit 2 Fire Zone 2097-X east dc equipment room

- December 23, 2005, Unit 1 Fire Zone 129-F Unit 1 control room

- December 23, 2005, Unit 2 Fire Zone 2199-G Unit 2 control room

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

This inspection procedure requires a minimum sample size of four samples consisting of Sections 02.01, 02.02, 02.03, and 02.04. All sections were completed except for Section 02.02 because the reactor vessel head was replaced and those inspections were not required. Temporary Instruction (TI) 2515/150, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009)," was completed during Refueling Outage 1R19 as a result of the licensee replacing the reactor vessel head.

.1 Inspection Activities Other than SG Tube Inspection, Pressurizer Water Reactor Vessel Upper Head Penetration Inspections, and Boric Acid Corrosion Control (Section 02.01)

a. Inspection Scope

The inspection procedure requires review of two or three types of nondestructive examination activities and one to three welds performed on the reactor coolant pressure boundary.

The inspectors observed five nondestructive examination activities including volumetric and surface examinations as follows:

System	Component/Weld Identification	Examination Method
Main Steam	4 welded lugs 1-045W	Magnetic Particle
Emergency Feedwater	Field Welds FW-7, 8, 10, and 11	Radiographic
Main Feedwater	Field Weld FW-31	Radiographic
Main Steam	Field Weld FW-6C1 and 43	Radiographic

System	Component/Weld Identification	Examination Method
Reactor Coolant	Reactor Vessel Closure Head Penetrations	Ultrasonic

During the observation of each examination, the inspectors verified that activities were performed in accordance with ASME Boiler and Pressure Vessel Code requirements and applicable procedures. The inspectors verified that the licensee compared the indications revealed by the examinations against the previous outage examination reports as applicable. No defects or reportable flaws were detected during the inservice examinations. The inspectors verified that the licensee used calibrated and qualified instruments and personnel.

Of the eight ASME Class 1 and 2 large bore welding activities performed by licensee personnel, the inspectors reviewed a sample of one Work Package Instruction 3065C, on the making of a large bore main steam circumferential pipe on the hot leg of SG A Welds 40 and 41. The inspectors verified that the welding activities met ASME Code requirements.

b. Findings

No findings of significance were identified.

.2 Pressurizer Water Reactor Vessel Upper Head Penetration Inspection Activities (Section 02.02)

a. Inspection Scope

The inspectors verified that the licensee replaced the reactor vessel head. The inspectors verified that licensee personnel performed a baseline ultrasonic inspection of the upper head during this outage as required by the ASME code.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control Inspection Activities (Section 02.03)

a. Inspection Scope

The procedure requires observation or review of boric acid corrosion control activities. Specifically, the procedure requires review of one to three engineering evaluations performed for boric acid residue found on reactor coolant system piping and components. This procedure also required review of one to three corrective actions taken because of evidence of boric acid leaks.

The inspectors reviewed records of a visual examination of the reactor coolant system pressure boundary integrity walkdown. The inspectors reviewed the 65 areas with boric acid residue identified by licensee personnel as of the time of this review (the licensee had not completed all the inspections) to assure identification and correction of leakage. The inspectors reviewed the corrective action documents written to evaluate the areas identified during this outage. The inspectors verified that licensee personnel adequately evaluated minor leaks to assure correction of leakage problems. The inspectors reviewed the corrective actions taken at that time and reviewed the corrective actions taken and completed during the previous outage.

b. Findings

No findings of significance were identified.

.4 SG Tube Inspection Activities (Section 02.04)

a. Inspection Scope

The inspectors reviewed the leakage history for the SGs to verify that the licensee had no excessive leakage during operations before the shutdown. The inspectors verified that licensee personnel and contractors used properly qualified eddy current probes and equipment for the expected types of tube degradation to assure proper identification and evaluation of indications for the new baseline data. The inspectors observed the collection, analysis, and resolution of nine calibration groups of the new baseline eddy current data, performed by contractor personnel to evaluate tubes and possible loose parts in the SGs to assure proper implementation of the procedures and program requirements. The inspectors verified that the licensee analysts reviewed the areas of potential degradation, based on site-specific and industry experience, to assure proper use of this information. The inspectors reviewed the repair criteria used to assure compliance with technical requirements. The inspectors also verified the licensee's eddy current examination scope and expansion criteria met the Technical Specifications, industry guidelines, and commitments to the NRC.

Regarding plugging and in-situ pressure testing, because the SGs were new replacement SGs, the licensee had no need for plugging and in-situ pressure testing. The inspectors verified that the predictions of tube plugging were reasonable.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspection procedure requires review of a sample of problems associated with inservice inspections and SG inspections documented by licensee personnel in the CAP for appropriateness of the corrective actions.

The inspectors reviewed a sample of the condition reports (CRs) written since the last outage which dealt with the boric acid control program, inservice inspection, and SG eddy current inspection activities and found the corrective actions were appropriate. The inspectors performed this review to assure that the licensee had an appropriate threshold for entering issues into the CAP and had procedures that direct root cause evaluations when necessary.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

Resident Inspector Quarterly Review. On November 8, 2005, the inspectors observed testing and training in the Unit 2 simulator of the senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. Unit 2 Dynamic Exam Scenario SES-2-007, Revision 2, was used to evaluate operations Crew D for Requal Cycle 0602 Simulator Evaluation A2SESLOR0602. The scenario involved a stator water runback, a failed letdown flow controller, degraded voltage and loss of offsite power, failure of service water to one emergency diesel generator, and failure of the safety parameter display system to update. The inspectors observed the evaluator and crew critiques following the evaluation scenarios.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the maintenance activities of the two systems listed below for items such as: (1) appropriate work practices; (2) identifying and addressing common cause failures; (3) scoping in accordance with 10 CFR 50.65(b) of the maintenance rule; (4) characterizing reliability issues for performance; (5) trending key parameters for condition monitoring; (6) charging unavailability for performance; (7) classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2);
and (8) appropriateness of performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified as (a)(1). In addition, the inspectors specifically reviewed events where ineffective equipment maintenance has resulted in invalid automatic actions of engineering safeguards systems affecting the operating units.

- November 27-29, 2005, Unit 1 decay heat system
- November 30 through December 2, 2005, Unit 1 reactor building spray system

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- August 18, 2005, Unit 1 blocking Door 61 train bay to auxiliary building for approximately 4 hours

- September 24 through December 31, 2005, Units 1 and 2 activities inside the protected area for the Unit 1 replacement outage

- December 19 through December 23, 2005, Units 1 and 2 work activities

- December 21, 2005, Unit 2 maintenance and surveillance activities during preparation for startup on Unit 1

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Nonroutine Plant Evolutions and Events (71111.14, 71153)

a. Inspection Scope

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

- December 26, 2005, Unit 1 turbine and reactor trip from 95 percent rated thermal power resulting from a low turbine bearing pressure

- December 31, 2005, Unit 1 EFW initiation and control actuation on SG low level setpoint

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- August 26, 2005, Unit 1 high pressure injection, low pressure injection, and building spray pump lube oil oilers; and high pressure injection motor and gearbox, and low pressure injection motor lube oil temperature elements

- October 18, 2005, Units 1 and 2 Startup Transformers 1, 2, and 3 voltage regulators spiking low

- November 22, 2005, Unit 2 emergency diesel generator service water expansions joints found blistered

- December 13, 2005, Unit 2 intake structure service water sluice gates failure to properly stroke

The inspectors completed four samples.

b. Findings

Introduction: The inspectors identified an example of a Green noncited violation of 10 CFR Part 50, Appendix B, Criteria III, “Design Control,” for the failure to ensure safety grade components were installed in the high pressure injection system.

Description: On August 26, 2005, the inspectors were performing a walk down of the high pressure injection pumps and noticed that a temperature element on the high pressure injection Pump C gearbox appeared to be bent significantly more than the others. Further investigation revealed that the temperature elements installed in the high pressure injection pump motors and gearboxes and the low pressure injection pump motors were nonsafety grade. Additionally, the high pressure injection Pump C outboard motor bearing temperature element protective sheath was missing. Since these temperature elements make up part of the lube oil system boundary, they should be classified as safety-grade components. The identified condition affected 22 out of 22 temperature elements between the two systems, and had existed since initial construction.

As part of the walk down, the inspectors also noted that the inboard and outboard optomatic oilers on the low pressure injection Pump B motor were different. As a result, the inspectors walked down the remaining low pressure injection pumps as well as the high pressure injection and building spray pumps. The inspectors discovered that only one oiler was different. Upon questioning the licensee, it was determined that, except for the one oiler that was different, all the others were nonsafety grade. Since the oilers make up part of the lube oil system boundary for the pumps, they should be classified as safety grade components. Additionally, the installed piping that connected the oilers and the motors was also determined to be nonsafety grade when it should have been safety-grade material. This condition affected 13 out of 14 oilers and the connecting piping issue affected 14 out of 14 connections. The licensee's data base, included in their PassPort system, incorrectly identified these components as nonsafety-related. This designation had existed since initial construction and transferred into the PassPort data base.

Analysis: The inspectors determined that the installation of nonsafety-related components into safety-related systems, and the temperature element missing the protective sheath, was a performance deficiency. This finding was more than minor because it affected the design control attribute under the Mitigating Systems Cornerstone objective of ensuring the reliability of systems that respond to initiating events to prevent undesirable consequences. Using the Phase 1 worksheets in Manual Chapter 0609, “Significance Determination Process,” the issue was determined to have very low safety significance (Green) because the finding did not result in a loss of function per Part 9900 Technical Guidance, “Operability Determination Process for Operability and Functional Assessment”; did not represent an actual loss of safety function; and is not potentially risk significant due to external events.

Enforcement: 10 CFR Part 50, Appendix B, Criteria III, “Design Control,” requires, in part, that measures shall be established for the selection and review of materials, parts, equipment, and processes that are essential to safety-related functions. These measures shall include provisions to assure that appropriate quality standards are specified, and that deviations from such standards are controlled. Contrary to this, the licensee failed to ensure adequate controls were in place to prevent nonsafety-related components from being installed in safety-related systems. Because of the very low safety significance and because the licensee included these conditions in their CAP as CRs ANO-1-2005-1251 and ANO-1-2005-1252, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000313/2005005-01, “Failure to Prevent Nonsafety-Related Components from Being Installed in Safety-Related Systems.”

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

Cumulative Review of the Effects of Operator Workarounds: On November 22, 2005, the inspectors reviewed the cumulative effects of operator workarounds to determine: (1) the reliability, availability, and potential for misoperation of a system; (2) if multiple mitigating systems could be affected; (3) the ability of operators to respond in a correct and timely manner to plant transients and accidents; and (4) if the licensee has identified and implemented appropriate corrective actions associated with operator workarounds.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17B)

a. Inspection Scope

The inspection procedure requires a minimum sample size of five plant modifications. The inspectors reviewed eight permanent plant modification packages (engineering requests) and their associated documentation, such as 10 CFR 50.59 safety evaluations, applicability determinations and screenings, to verify that the modifications were performed in accordance with regulatory requirements and plant procedures. The inspectors reviewed procedures governing plant modifications to evaluate the effectiveness of the programs for implementing modifications to risk-significant systems, structures, and components, such that these changes did not adversely affect the design and licensing basis of the facility. The inspectors have listed in the attachment to this report the procedures and permanent plant modifications reviewed. The inspectors interviewed the cognizant design and system engineers for the identified modifications to gain their understanding of the modification packages.

The inspectors evaluated the effectiveness of the licensee’s corrective action process to identify and correct problems concerning the performance of permanent plant modifications. In this effort, the inspectors reviewed a sample of corrective action reports identified in the attachment to this report. The review included the subsequent corrective actions pertaining to licensee-identified problems and errors in the performance of permanent plant modifications to assure proper resolution of the issues.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the eight postmaintenance test activities of risk significant systems or components listed below. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly realigned, and deficiencies during testing were documented. The inspectors also reviewed the CAP to determine if the licensee identified and corrected problems related to postmaintenance testing.

- August 11, 2005, Unit 2 component cooling water heat Exchanger A leakage
- October 2, 2005, Units 1 and 2 diesel-driven Fire Pump P-6B
- November 3, 2005, Unit 1 failure of Valve SV-2613 during shutdown of turbine driven EFW pump
- November 4, 2005, Unit 2 control element drive mechanism control system and control element assembly testing following restoration of dropped control element assembly
- November 16, 2005, Unit 2 service water Bay A emergency cooling pond Sluice Gate 2CV-1471-1
- December 13, 2005, Unit 2 service water Bay C emergency cooling pond Sluice Gate 2CV-1475-2
- December 14, 2005, Unit 1 service water supply to Unit 2 control room Emergency Chiller 2SW-69A weld repair
- December 15, 2005, Unit 1 decay heat room Cooler VUC-1D replacement

The inspectors completed eight samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

Unit 1 Refueling and SG and Reactor Vessel Head Replacement Outage.
The inspectors reviewed the outage safety plan and contingency plans for the Unit 1 refueling, reactor vessel head, and SG replacement outage, conducted October 4 through December 22, 2005, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. Other activities that were accomplished during this outage, per Procedures 50001 and 71007 are documented in NRC Inspection Report 05000313/2005010. The inspectors observed portions of the shutdown and cooldown, and of the heatup and startup processes, and monitored licensee controls over the outage activities listed below:

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the outage safety plan for key safety functions and compliance with the applicable Technical Specification when taking equipment out of service

- Implementation of clearance activities and 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 and an accounting for instrument error

- Controls over the status and configuration of electrical systems to ensure that Technical Specification and outage safety plant requirements were met and controls over switchyard activities

- Monitoring of decay heat removal processes

- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system

- Monitoring of licensee activities during periods of reduced inventory and midloop conditions to ensure risk was appropriately managed and two independent means of monitoring level and temperature were always available

- 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 Technical Specifications

- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage

- Containment sump controls and maintenance to ensure no damage or deformation

- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the 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

Unit 1 Forced Outage: On December 26, 2005, the unit tripped due to a momentary loss of turbine lube oil bearing pressure. The licensee exited this forced outage on December 29, 2005. The inspectors reviewed the outage plan and contingency plans to confirm that the licensee 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 outage, the inspectors reviewed computer trends and control activities for portions of the shutdown and cooldown, monitored licensee configuration management, reviewed controls over the status and configuration of mitigating systems, monitored controls over activities that could affect reactivity, and reviewed trends and control room activity associated with startup and ascension to power operation. Finally, the inspectors reviewed licensee personnel's identification and resolution of problems related to the outage activities.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- November 10, 2005, Unit 2 quarterly containment isolation valve stroke test
- November 14, 2005, Unit 2 midcycle main steam safety valve test
- December 14, 2005, Unit 1 containment sump isolation Valve CV-1406
- December 22, 2005, Unit 1 high pressure injection Pump P-36A in-service test

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspector reviewed licensee submissions and verified with the licensee that no emergency plan or emergency action level changes were made during calendar year 2005. Procedure 71114.04 was not performed for the licensee during calendar year 2005 due to lack of opportunity.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control To Radiologically Significant Areas (71121.01)

a. Inspection Scope

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

- PI events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone

- Controls (surveys, posting, and barricades) of four radiation, high radiation, or airborne radioactivity areas

- Radiation work permit, procedure, engineering controls, and air sampler locations

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

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

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

- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection

- Corrective action documents related to access controls

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions

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

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

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

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

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

Either because the conditions did not exist or an event had not occurred, no opportunities were available to review the following items:

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem control element drive mechanism

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

The inspector completed 21 of the required 21 samples

Additionally, using Procedure 71121.01, the inspector reviewed activities associated with the Unit 1 SG and reactor vessel head replacement to fulfill the inspection requirements of Procedure 50001, "Steam Generator Replacement Inspection," and Inspection Procedure 71007, "Reactor Vessel Head Replacement Inspection."

b. Findings

Introduction: The inspector reviewed a Green self-revealing noncited violation of Technical Specification 6.7.1.a because the licensee failed to control an HRA by not barricading and conspicuously posting the area. The violation had very low safety significance.

Description: On March 15, 2005, the licensee removed a temporary barrier (scaffold boards) creating an entrance to an HRA with surveyed dose rates of up to 400 millirem per hour at 30 centimeters. The entrance that was created did not have the proper radiological controls in place for an HRA as per the licensee's Technical Specification 6.7.1.a requirements. Poor coordination and communication between the scaffold crew and radiation protection personnel directly contributed to the finding. It was not until two radiation workers entered the area that a radiation protection technician identified the uncontrolled entry point and established the necessary controls. The workers were on a proper radiation work permit for entering the area and did not receive any unanticipated exposure.

Analysis: The failure to control an HRA as per Technical Specification requirements is a performance deficiency. The finding is greater than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of program and process and affects the cornerstone objective to ensure the adequate protection of the worker's health and safety from exposure to radiation. The finding involved the potential for a worker's unplanned or unintended dose resulting from actions contrary to Technical Specification requirements. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. In addition, this finding has crosscutting aspects associated with human performance because poor coordination and communication between the scaffold crew and radiation protection personnel directly contributed to the finding.

Enforcement: Technical Specification 6.7.1.a requires that each entryway into an HRA with dose rates not exceeding 1.0 rem/hour at 30 centimeters from the radiation source shall be barricaded and conspicuously posted as an HRA. Contrary to the above, the licensee allowed the removal of scaffold boards creating an entryway into an HRA that was not barricaded and conspicuously posted. The finding was entered into the licensee's CAP as CR ANO 2-2005-0574. Because the failure to barricade and conspicuously post this HRA is of very low safety significance and has been entered into the licensee's CAP, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000368/2005005-02, "Failure to Barricade and Conspicuously Post an HRA."

4. OTHER ACTIVITIES

4OA1 PI Verification (71151)

a. Inspection Scope

The inspector sampled licensee submittals for the PIs listed below for the period from May 2004 through October 2005. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, “Regulatory Assessment Indicator Guideline,” Revision 3, were used to verify the basis in reporting for each data element.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness PI

Licensee records reviewed included corrective action documentation that identified occurrences in locked HRAs (as defined in the licensee’s Technical Specifications), very HRAs (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the PI data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very HRAs were properly controlled.

Public Radiation Safety Cornerstone

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

Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded PI thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the PI data.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

.1 Review of Items Entered into the CAP

As required by Procedure 71152, “Identification and Resolution of Problems,” and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed screening of all items entered into the licensee’ CAP. This was accomplished by reviewing the description of each new CR and attending daily management review committee meetings.

.2 Semiannual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” the inspectors performed a review of the licensee’s CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors’ review was focused on repetitive equipment and corrective maintenance but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.1. The review also included issues documented outside the normal site monthly meeting reports and maintenance rule assessments. The inspectors' review nominally considered the 6-month period of June through December 2005, although some examples expanded beyond those dates when the scope of the trend warranted. The inspectors compared and contrasted their results with results from similar licensee efforts. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy.

b. Findings

From the 2005 midcycle assessment meeting, the inspectors were committed to review licensee efforts to identify and track instances of ineffective or untimely corrective action. The inspectors conducted this review along with reviewing the licensee's list of open corrective actions for deficient conditions. The inspectors also examined all items classified under Generic Letter 91-18, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," as well as all CRs greater than one year old, specifically looking for any safety significant deficiencies which were not being adequately addressed or not being addressed in a timely manner. The inspectors found no instances of ineffective or untimely corrective action.

During 2005, licensee personnel documented eight instances of poor communications between maintenance personnel working on various plant equipment and operations personnel who were not cognizant of the work being performed. None of these instances actually challenged plant safety, but the number of documented instances was indicative of a need for improved communications between maintenance and operations personnel. Licensee management was made aware of this performance issue and have implemented corrective actions to improve communications.

Also during 2005, the following conditions pertaining the control room emergency ventilation system (CREVS) were documented by the licensee: four instances of improper implementation of Technical Specifications, four instances of inadvertent system initiations, and two instances of poor design control. None of these instances actually challenged plant safety but the number of documented instances led the inspectors to question the licensee's attentiveness to the CREVS. Licensee management was made aware of these issues and have implemented corrective actions to improve system awareness.

.3 Annual Sample Review

a. Inspection Scope

The inspectors chose two issues for more in-depth review to verify that licensee personnel had taken corrective actions commensurate with the significance of the issues. The issues and their bases for their selection is described below:

- On March 8, 2005, the licensee shut down Unit 2 due to indications of SG tube leakage from an SG which had been installed less than 5 years earlier. The inspectors chose to review this occurrence due to its long standing and recurring nature in light of the planned replacement of the Unit 1 SGs.

- In calendar year 2005, recurring instances of lack of attentiveness to the CREVS were noted which dealt with a variety of issues including unplanned breaches of the system and failures to enter required Technical Specifications in a timely manner. Also, several unplanned initiations of the system occurred in 2005. The inspectors chose to review the commonality of these issues and how the licensee dealt with them.

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

- Complete and accurate identification of the problem in a timely manner commensurate with its significance and ease of discovery

- Evaluation and disposition of operability and reportability issues

- Consideration of extent of condition, generic implications, common cause, and previous occurrences

- Classification and prioritization of the resolution of the problem commensurate with its safety significance

- Identification of root and contributing causes of the problem for significant conditions adverse to quality

- Identification of corrective actions which are appropriately focused to correct the problem

- Completion of corrective actions in a timely manner commensurate with the safety significance of the issue

The inspectors completed two samples.

b. Findings

Introduction: The inspectors reviewed a Green self-revealing finding for the failure to prevent the introduction of foreign material in the Unit 2 SGs which eventually caused tube leakage in one of the SGs.

Description: On February 26, 2005, Unit 2 operators noticed that indication on the SG A Nitrogen-16 radiation monitor had increased from less than 1 gallon per day (gpd) to 4.5 gpd. Additionally, Argon-41 isotopic sampling indicated an increase from less than 0.37 gpd to 1.8 gpd and tritium leak rate determinations had increased from around 0.5 gpd to 2.89 gpd. All of these attendant indications confirmed leakage from SG A. The licensee monitored the leakage as it slowly increased until March 8, 2005, when the leak rate was noticed to increase, spiking above 30 gpd at 10:30 a.m. that morning. Based on this information, licensee management decided to shut the unit down and enter Refueling Outage 2R17 that night, 1 week earlier than the scheduled start date of March 15, 2005.

Upon shutting and cooling down the plant, the licensee conducted a nitrogen overpressure test on SG A and determined the source of the leakage was Tube 70-169 on the hot leg side. The licensee drained the secondary side of the SG and removed the lower hand holes to facilitate visual inspection. This inspection found that a piece of metal was located between Tubes 70-169 and two adjacent tubes. The adjacent tubes had indications that the piece of metal had worn away part of their wall thickness also, but had not worn completely through. The licensee expanded the scope of their visual inspection for foreign material and found five more pieces of debris in SG A and four pieces in SG B. Some of these pieces were determined to originate from SG fabrication.

The licensee analyzed the piece of metal which caused the SG tube leak and determined it to be a piece of cold worked steel that was approximately 1.25 inches long, 0.5 inches wide, and 0.125 inches thick. The licensee's root cause could not definitively pinpoint the source of the piece, but SG fabrication activities conducted between 1997 and 2000 were identified as the most likely source.

Entergy had contracted Westinghouse to design and build their new SGs. Westinghouse, in turn, contracted Equipos Nucleares (ENSA) to build the SG's at the ENSA facility in Santander, Spain. The SGs were installed in 2000 and replaced the original Unit 2 SGs which had been in service since plant construction. The licensee had previously identified numerous instances of finding foreign material in the SGs during fabrication construction at ENSA. ENSA Procedure OEB2-ES-801, "Foreign Object Control," addressed foreign material exclusion. This procedure had been reviewed by the licensee prior to use during steam generator fabrication. The licensee initiated numerous surveillance reports documenting the intrusion of foreign materials and poor foreign material exclusion practices during this time, along with actions to correct these deficiencies. Despite these actions, foreign material from fabrication was discovered in the SGs in March 2005. The inspectors concluded that even though the foreign material entry most likely occurred under ENSA and Westinghouse foreign material exclusion programs, the licensee's oversight of contractors during fabrication was inadequate and was in part responsible for the introduction of the foreign materials.

After shutting down and determining the source of the leak, the licensee conducted testing of the leaking tube and confirmed that the tube would withstand three times the differential pressure across it during normal full power, steady state operation, in addition to withstanding the maximum differential pressure expected during a design basis main steam line break event. Before restarting Unit 2, the licensee plugged the three affected tubes and removed any foreign material which could potentially impact tube performance during future cycles.

Analysis: The inspectors considered that the failure to properly implement foreign material exclusion procedures to prevent the introduction of foreign material into the Unit 2 SGs was a performance deficiency. The inspectors determined that this issue is greater than minor because it affected the reactor coolant system barrier performance attribute under the Barrier Integrity Cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The inspectors then referred to Manual Chapter 0609, "Significance Determination Process," which guided the inspectors to Appendix J, "Steam Generator Tube Integrity Findings Significance Determination Process," of Manual Chapter 0609 since the finding was assumed to degrade SG tube integrity. The inspectors determined that Appendix J was not directly applicable since the finding was not related to deficiencies in the inservice inspection program. However, through management review using the guidance contained in Appendix J for equivalently affected SG tube issues resulting from inservice inspection deficiencies, the finding was determined to be of very low safety significance (Green) since (1) the affected tubes were not susceptible to burst, (2) the affected SG did not exceed accident leakage performance criteria, (3) the affected tubes could withstand three times normal SG operating pressure, and (4) the tubes could withstand the maximum expected differential pressure expected during a main steam line break.

Enforcement: Because the inspectors and the licensee were unable to determine when the foreign material was introduced into the SG, the inspectors could not conclude that a violation of regulatory requirements occurred. The licensee included this condition in their CAP as CR ANO-2-2005-0344. This issue is being treated as a finding: FIN 05000368/2005005-03, "Foreign Material Causes Leak in a Unit 2 SG."

.4 Access Control To Radiologically Significant Areas Review

Section 2OS1 evaluated the effectiveness of the licensee's problem identification and resolution processes regarding access controls to radiologically significant areas and radiation worker practices. The inspector reviewed corrective action documents for root cause/apparent cause analysis against the licensee's problem identification and resolution process. No issues of significance were identified during this review.

4OA5 Other Activities

.1 Temporary Instruction (TI) 2515/150, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009)," Revision 3

Unit 1: During refueling Outage 1R19 which lasted from October 2005 through December 2005, the licensee replaced the reactor pressure vessel (RPV) head for Unit 1. TI 2515/150 states the plants that replace their RPV heads during an outage are not expected to perform TI 2515/150 unless the licensee intends to inspect the RPV head prior to removal from service. The licensee did not inspect the Unit 1 RPV head prior to removing it from service, therefore, TI 2515/150 was not performed during Refueling Outage 1R19. TI 2515/150 also states that the TI expires when a plant replaces its RPV head, therefore, TI 2515/150 is closed for Unit 1.

Unit 2: TI 2515/150 was performed during Refueling Outages 2R16 in Fall 2003 and 2R17 in Spring 2005. In accordance with TI 2515/150, this TI has been completed twice prior to expiration of the TI and is, therefore, closed for Unit 2.

.2 TI 2515/160, "Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01)"

a. Inspection Scope

The inspectors performed applicable sections of TI 2515/160 on Unit 1 to determine whether the inspections by the licensee are consistent with the licensee's response to NRC Bulletin 2004-01, "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized Water Reactors," and any subsequent, related correspondence between the licensee and the NRC staff.

(1) For each of the examination methods used during the outage, was the examination:

(a) Performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

The responsible engineers used to complete Procedure 2311.009, “ANO Unit 1 and Unit 2 Alloy 600 Inspection,” Revision 9, and its attachments all received boric acid training. Boric acid training consists of the following: (1) importance of accurate reporting of the location, (2) examples of industry leaks, (3) importance of not disturbing deposits, (4) identification of the source and targets, (5) NRC findings against the boric acid control program, (6) industry document reviews that address boric acid corrosion, (7) distinction between wet and dry leak, (8) distinction between color of the leak, (9) review of operating experience and industry photographs of boric acid leaks, and (10) required documentation.

(b) Performed in accordance with demonstrated procedures?

Yes, Procedure 2311.009, “ANO Unit 1 and Unit 2 Alloy 600 Inspections,” Revision 9, and its attachments, is the one previously utilized that identified heater sleeve leakage on Unit 2.

(c) Able to identify, disposition, and resolve deficiencies?

The inspectors determined that the licensee’s threshold for initiating CRs was low, thereby, capturing most deficiencies identified. The inspectors also concluded that corrective actions were being appropriately addressed.

(d) Capable of identifying the leakage in pressurizer penetration nozzle or steam space piping components as discussed in NRC Bulletin 2004-01?

Yes, the procedural controls in place and the requirements of the inspecting personnel were adequate to ensure that the licensee was capable of identifying small leaks.

(2) What was the physical condition of the penetration nozzle and steam space piping components in the pressurizer system (e.g., debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

The pressurizer heaters are mounted horizontally in the lower portion of the side shell and are made of stainless steel. For the initial walkdown, the insulation was still in place and no signs of boric acid, wetting, or deposits were present on the outside of the insulation. For the bare metal visual examination, the as-found condition when the insulation was removed was clean. The inspectors visually observed the licensee perform the initial walkdown with the insulation in place, as well as accompany the inspection team for the as-found inspection.

(3) How was the visual inspection conducted (e.g., with video camera or direct visual by the examination personnel)?

The inspection was conducted by direct visual inspection by a responsible engineer. A responsible engineer has completed the boric acid training. Additionally, the licensee had a second person perform a direct visual inspection to ensure that nothing was missed. The inspectors performed an independent direct visual inspection to review the categorization of the licensee's inspection results to verify the accuracy of the as-found condition.

(4) How complete was the coverage (e.g., 360E around the circumference of all the nozzles)?

The penetrations that were inspected were directly inspected 360E around the circumference.

(5) Could small boron deposits, as described in the NRC Bulletin 2004-01, be identified and characterized?

Yes, the licensee did in fact identify small boron deposits on other systems, just not on the pressurizer.

(6) What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair?

None.

(7) What, if any, impediments to effective examinations, for each of the applied methods, were identified (e.g., centering rings, insulation, thermal sleeves, instrumentation nozzle distortion)?

The licensee did not have any impediments that precluded an effective examination.

(8) If volumetric or surface examination techniques were used for the augmented inspections examinations, what process did the licensee use to evaluate and dispose of any indications that may have been detected as a result of the examinations?

The licensee did not perform additional examinations due to not finding any evidence of boric acid.

(9) Did the licensee perform appropriate followup examinations for the indications of boric acid leaks from the pressure-retaining components in the pressurizer system?

The licensee did not perform additional examinations due to not finding any evidence of boric acid.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On November 4, 2005, the inspector presented the access controls inspection results to Mr. David Moore, Radiation Protection Manager, and other members of your staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On December 2, 2005, the engineering inspectors presented the inspection results by telephone to Mr. Jeff Forbes, Vice President of Operations, and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On January 10, 2006, the inspector conducted a telephonic meeting with Mr. R. Holeyfield, Emergency Preparedness Manager, to verify the licensee had not made changes to its emergency plan or emergency action levels during calendar year 2005.

The resident inspectors presented the inspection results of the resident inspections to Mr. J. Forbes, Vice President, Operations, and other members of the licensee's management staff on January 10, 2006. The licensee acknowledged the findings presented. The inspectors noted that while proprietary information was reviewed, none would be included in this report.

4OA7 Licensee-Identified Violations

The following is an example of a violation of very low safety significance (Green) which was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of NUREG-1600, “NRC Enforcement Policy,” for being dispositioned as an NCV.

10 CFR 50.65(a)(4) requires, in part, that the licensee shall assess and manage the increase in risk that may result from proposed maintenance activities. Contrary to this, on August 17, 2005, the licensee did not adequately assess risk from maintenance activities that resulted in a high energy line break/fire door being open for approximately 4 hours. Door 61 connects the Unit 1 auxiliary building to the train bay. While the licensee did have a continuous firewatch stationed, they failed to consider the possible undesirable heat up of the room itself or adjacent areas. This finding is of very low safety significance because the licensee did have a continuous firewatch posted and no indications that a significant heat up occurred. This condition is captured in the licensee’s CAP as CR ANO-1-2005-1212.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee Personnel</u>

R. Barnes, Manager, Planning and Scheduling
S. Bennett, Project Manager, Licensing
B. Berryman, Manager, Unit 1 Operations
E. Blackard, Supervisor, Mechanical Design Engineering
J. Browning, Manager, Unit 2 Operations
J. Eichenberger, Manager, Corrective Actions and Assessments
J. Forbes, Vice President, Operations
W. Greeson, Supervisor, Engineering Programs and Components
J. Harrell, Radiation Protection Specialist
A. Hawkins, Licensing Specialist
J. Hoffpauir, Manager, Maintenance
R. Holeyfield, Manager, Emergency Planning
D. James, Manager, Licensing
W. James, Manager, Alloy 600 Group
J. Johnson, Technical Specialist, Fire Protection
R. Jones, Technical Specialist, Engineering Programs and Components
J. Kowalewski, Director, Engineering
R. Kowalewski, Manager, Technical Support
D. Lomax, Manager, Dry Fuels
T. Marlow, Director, Nuclear Safety Assurance
J. Miller, Manager, Systems Engineering
T. Mitchell, General Manager, Plant Operations
D. Moore, Manager, Radiation Protection
K. Nichols, Manager, Design Engineering
R. Puckett, Supervisor, Fire Protection
S. Plye, Licencing Specialist, Licencing
L. Qualls, Radiation Protection Specialist
S. Pyle, Licensing Specialist
C. Reasoner, Manager, Engineering Programs and Components
R. Scheide, Licensing Specialist
C. Tyrone, Manager, Quality Assurance
B. Williams, Director, Reactor Vessel Head/SG Replacement Project
G. Woerner, Supervisor, ROTSG/RVCH Project

<u>NRC Personnel</u>

J. Fair, Senior Mechanical Engineer
K. Karwoski, Senior Level Advisor

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000313/2005005-01	NCV	Failure to Prevent Nonsafety-Related Components from being Installed in Safety-Related Systems (Section 1R15)
05000368/2005005-02	NCV	Failure to Barricade and Conspicuously Post an HRA (Section 2OS1)
05000368/2005005-03	FIN	Foreign Material Causes Leak in a Unit 2 SG (Section 4OA2)

Closed

None

Discussed

None

LIST OF DOCUMENTS REVIEWED

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

Section 1R01: Adverse Weather Protection

Operating Procedures

NUMBER	TITLE	REVISION
1104.039	Plant Heating and Cold Weather Operations	17
2106.032	Unit 2 Freeze Protection Guide	10

Section 1R04: Equipment Alignment

Drawing

M-232, Sheet 1, Revision 100

Operating Procedures

NUMBER	TITLE	REVISION
1104.004	Decay Heat Removal Operating Procedure	72
1104.036	Emergency Diesel Generator Operation	43
1107.001	Electrical System Operations	60
1107.002	ES Electrical System Operation	20
2104.036	Emergency Diesel Generator Operations	49
2104.037	Alternate AC Diesel Generator Operations	8
2107.001	Electrical System Operations	50
2107.002	ESF Electrical System Operation	16
2107.009	DC Electrical System Operation	21

Engineering Calculation

Miscellaneous Documents

NUMBER	TITLE	REVISION
	Arkansas Nuclear One Fire Hazards Analysis Report	9

Plant Drawings

NUMBER	TITLE	REVISION
FP-103	Intermediate Floor Plan Elevation 368' - 0" and 372' - 0"	24
FP-2101	Fuel Handling Floor Plan Elevation 404' - 0" and 422' - 0"	14
FP-2103	Intermediate Floor Plan Elevation 368' - 0" and 372' - 0"	26

1R08: Inservice Inspection Activities (71111.08)

Drawings

NUMBER	TITLE	REVISION
1-MS-4	Large Pipe Isometric Steam Generator E-24B Secondary Line from Containment to CV-2692	14
MS-112	Hanger Detail Steam Generator Secondary	2

Procedures

NUMBER	TITLE	REVISION
1032.037	Inspection and Evaluation of Boric Acid Leaks	001-04-0
55-OI0035-03	Reactor Coolant System Welding for B & W Unit Steam Generator Replacement Projects	3

GTM/1.1-4	ASME Section IX Welding Procedure Specification	0
CEP-NDE-0731	Magnetic Particle Examination (MT) (ASME Section XI)	
31-045W	SGT GTMSSW/1,1-4. ASME Section IX Welding Procedure Specification	0

Nondestructive Examination Reports

SGT Radiographic Examination Reports for emergency feedwater Welds FW-7, 8, 10, and 11
SGT Radiographic Examination Reports for main steam Welds FW-6C1 and 43
Entergy Radiographic Examination Report for main feedwater FW-31

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ANO-1-2002-1221	ANO-1-2005-1779	ANO-1-2005-2474
ANO-1-2005-1438	ANO-1-2005-2161	ANO-1-2005-2564
ANO-1-2005-1458	ANO-1-2005-2190	ANO-2-2004-0620
ANO-1-2005-1563	ANO-1-2005-2258	ANO-2-2005-0496
ANO-1-2005-1704	ANO-1-2005-2283	ANO-C-2005-0271

Miscellaneous

Boric Acid Database

3065C Work Package Instruction

3065C-01 Weld Data Card

3065C-03 Weld Data Card

Steam Generator Integrity Program Steam Generator Eddy Current Training Manual

EN-EP-S-001-A, ANO-1 Steam Generator Eddy Current Examination Data Analysis Guidelines, Revision 0

ER-ANO-2005-522-000, Steam Generator Pre-outage Degradation Assessment And Repair Criteria for 1R19, Revision 0

Arkansas Nuclear One Unit 1 Enhanced Once Through Steam Generator Baseline Equivalency Report Steam Generator A

BARK-B-105, ANTS (analysis technique specification sheet) Conventional Bobbin Coil Probe Revision 0

Section 1R11: Licensed Operator Requalification

Simulator Evaluation

NUMBER	TITLE	REVISION
A2SESLOR0602	Requal Cycle 0602 Simulator Evaluation	2

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ANO-1-2004-0029	ANO-1-2004-1087	ANO-1-2004-2403
ANO-1-2004-0100	ANO-1-2004-1119	ANO-1-2005-0825
ANO-1-2004-0226	ANO-1-2004-1161	ANO-1-2005-1251
ANO-1-2004-0369	ANO-1-2004-1471	ANO-1-2005-1252
ANO-1-2004-0619	ANO-1-2004-1702	ANO-1-2005-2160
ANO-1-2004-0937	ANO-1-2004-1714	ANO-1-2005-2224
ANO-1-2004-0943	ANO-1-2004-1738	ANO-1-2005-2437
ANO-1-2004-0952	ANO-1-2004-1742	ANO-1-2004-2501
ANO-1-2004-1040	ANO-1-2004-1799	ANO-1-2005-2553
ANO-1-2004-1067	ANO-1-2004-1832	ANO-1-2005-2554

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ANO-2004-0395-000

Miscellaneous Documents

NUMBER	TITLE	REVISION
ULD-1-SYS-05	Arkansas Nuclear One Upper Level Document Reactor Building Spray System	3
CEP-IST-1	IST Bases Document	3
	Unit 1 System Performance Indicators - Decay Heat Removal	
	Unit 1 System Performance Indicators - Reactor Building Spray	

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

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ANO-C-2004-1402　　ANO-1-2005-1212

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ANO-1996-3555-081

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CALC-95-R-0024-01	ANO-2 T-2 Report Week of 12/19/2005	System Status Report 12/21/2005

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ANO-1-2005-1251	ANO-1-2005-2742	ANO-2-2005-2570
ANO-1-2005-1252	ANO-2-2005-2271	ANO-2-2005-2572
ANO-1-2005-1325	ANO-2-2005-2353	ANO-2-2005-2576
ANO-1-2005-1564	ANO-2-2005-2354	ANO-C-2005-2017
ANO-1-2005-2224	ANO-2-2005-2360	ANO-C-2005-2105
ANO-1-2005-2630	ANO-2-2005-2381	

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ANO-2005-0605-000	ANO-2005-0605-002
ANO-2005-0605-001	ANO-2005-0693-000

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Operations Impact Concerns Database

1R17: Permanent Plant Modifications (71111.17B)

Calculations

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08-5015531	Design Specification	2
51-5038983	Section XI ASME Code Reconciliation for ANO-1 Enhanced Once Through Steam Generators	1
33-5020072	ASME Design Report	3

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ANO-1-20054-0365	ANO-1-2005-01365

Drawings

NUMBER	TITLE	REVISION
5017946	ANO-1 Enhanced Once Through Steam Generator Shell Outline	15
5018500E	ANO-1 Enhanced Once Through Steam Generator Lower Shroud Details	8
5018506E	ANO-1 Enhanced Once Through Steam Generator Main Feedwater Nozzle Components and Details	5
5018507E	ANO-1 Enhanced Once Through Steam Generator Emergency Feedwater Nozzle Components and Details	6
5022066E	ANO-1 Enhanced Once Through Steam Generator	
5018501E	ANO-1 Enhanced Once Through Steam Generator Upper Shroud Details	8

Engineering Requests

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ANO-2002-1381-000	Steam Generator Replacement	0
ANO-2002-1381-003	Reactor Coolant System Impact Review due to Replacement of the ANO-1 Steam Generators	0
ANO-2002-1381-004	Electrical/Instrument and Control Systems Impact Review due to Replacement of the ANO-1 Steam Generators	0
ANO-2002-1381-005	Nuclear Steam Supply Steam and Components Impact Review due to Replacement of the ANO-1 Steam Generators	0
ANO-2002-1381-006	Plant Component Programs Impact Review due to Replacement of the ANO-1 Steam Generators	0
ANO-2002-1381-007	Civil/Structural Components and Documents Impact Review due to Replacement of the ANO-1 Steam Generators	0
ANO-2002-1381-008	Piping Systems and Components Impact Review due to Replacement of the ANO-1 Steam Generators	0
ANO-2002-1381-009	Systems and Components Unaffected by the Replacement of the ANO-1 Steam Generators	0

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(ENS)-LI-101	10 CFR 50.59 Review Program	8
DC-105	Configuration Management	2
DC-112	Engineering Request and Project Initiation Process	4
DC-115	Engineering Request Response Development	7
EN-DC-114	Project Management	1
EN-LI-102	Corrective Action Program	3

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NUMBER	TITLE	REVISION
	Framatome ANP Quality Assurance Data Package (23-5063672-01) for Enhanced Once Through Steam Generators for Entergy Operations ANO-1 (P.O. 10008952) September 2005	

NUMBER	TITLE	REVISION
18-1173987-05	Reactor Coolant Systems Arkansas Nuclear One Unit 1	5
	Entergy Operation, Inc. Stand alone Agreement between Entergy Operation, Inc. and Framatome ANP, Inc. for Design, Fabrication, and Delivery of Replacement Steam Generators for Unit 1 at Arkansas Nuclear One	
18-1173987-05	Functional Specification-Reactor Coolant System Arkansas Nuclear One Unit 1	5
ANO-—559	ANO-1 Replacement Once Through Steam Generators	0

Section 1R19: Postmaintenance Testing

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ANO-1-2005-0825	ANO-2-2005-2481	ANO-C-2005-1527
ANO-1-2005-1343	ANO-2-2005-2570	ANO-C-2005-1725
ANO-1-2005-2714	ANO-2-2005-2572	ANO-C-2005-2105
ANO-2-2005-2191	ANO-2-2005-2576	ANO-C-2005-2268
ANO-2-2005-2192	ANO-C-2004-2274	
ANO-2-2005-2281	ANO-C-2005-1335	

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ANO-2001-0107-009	ANO-2005-0678-001
ANO-2001-0511-007	ANO-2005-0780-000
ANO-2005-0678-000	

Miscellaneous

TD A480.0020, “Installation, Operation and Maintenance Manual for ARMCO Sluice Gates Heavy Duty Series”

Operating Procedures

NUMBER	TITLE	REVISION
1025.033	Control of Post-Maintenance Testing	8
1104.032	Fire Protection Systems	57
1403.002	Unit 1 Target Rock Solenoid Valve Maintenance	7
2104.029	Service Water System Operations - Supplement 1A	55
2305.005	Valve Stroke and Position Verification - Supplement 3	22
5120.010	Unit 1 & Unit 2 MOV Testing Utilizing the VOTES Test System	5

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00061061 01	00073256 01	51010045 01
00070272 01	00077788 01	

Section 1R22: Surveillance Testing

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ANO-1-2004-1119	ANO-1-2005-2553

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NUMBER	TITLE	REVISION
CEP-IST-1	IST Bases Document	3

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NUMBER	TITLE	REVISION
2305.005	Valve Stroke and Position Verification	22
2306.006	Unit 2 Main Steam Safety Valve Test	16

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00038191 01	00038193 01	50968906 01
00038192 01	00038194 01	

Section 2OS2: Access Controls to Radiologically Significant Areas

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ANO-C-2004-0969	ANO-1-2005-1147	ANO-2-2005-0919
ANO-C-2004-1172	ANO-1-2005-1945	ANO-2-2005-1035
ANO-C-2004-1756	ANO-1-2005-1684	ANO-2-2005-1078
ANO-C-2004-1814	ANO-1-2005-2070	ANO-2-2005-1134
ANO-C-2004-1843	ANO-1-2005-2103	ANO-2-2005-1157
ANO-1-2004-1076	ANO-1-2005-2107	ANO-2-2005-1243
ANO-1-2004-1344	ANO-1-2005-2116	ANO-2-2005-1942
ANO-1-2004-1514	ANO-2-2005-0551	ANO-2-2005-2044
ANO-C-2005-0566	ANO-2-2005-0574	

Audits and Self-Assessments

Audit of Radiation Protection QA-14-2005-ANO-1
RP Third Quarter, 2005 Quarterly Roll-up Assessment
RP Second Quarter, 2005 Quarterly Roll-up Assessment
RP First Quarter, 2005 Quarterly Roll-up Assessment
RP Fourth Quarter, 2004 Quarterly Roll-up Assessment
RP Third Quarter, 2004 Quarterly Roll-up Assessment
RP Second Quarter, 2004 Quarterly Roll-up Assessment
RP First Quarter, 2004 Quarterly Roll-up Assessment

Radiation Work Permits

2005-1414, Revision 1	2005-1658, Revision 0	2005-2452, Revision 1
2005-1650, Revision 0	2005-2442, Revision 3	

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1012.016	Administration of the ANO Radiation Protection	2
1012.017	Radiological Posting and Entry Exit Requirements	10
RP-105	Radiation Work Permits	7
RP-103	Access Control	2

Other Documents

Routine Radiological Survey Documentation Checklist
Daily Shift Turnover Log
1R19 Shift ALARA Report
Reactor Vessel Closure Head Removal and Transport Plan

4OA1: PI Verification

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ANO-1-2004-1155	ANO-1-2004-1738	ANO-2-2004-1588

Other Document

Annual Radioactive Effluent Release Report for 2004

4OA5: Other Activities

TI 2515/160, "Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01)"

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NUMBER	TITLE	REVISION
2311.009	ANO Unit 1 and Unit 2 Alloy 600 Inspections	9
2311.009E	Unit 1 Pressurizer A-600 Butt-Weld Inspections	9
2311.009N	Unit 1 Pressurizer A-600 Small Bore Nozzle/Weld Inspections	9

0CAN070404, Response to NRC Bulletin 2004-01 Regarding Inspection of Alloy 82/182/600 Materials Used in Pressurizer Penetrations and Steam Space Piping Connections, July 27, 2004

0CAN090501, Response to NRC Request for Additional Information for Bulletin 2004-01 Regarding Inspection of Pressurizer Penetrations, September 21, 2005

LIST OF ACRONYMS

ALARA	as low as reasonably achievable
ANO	Arkansas Nuclear One
ASME	American Society of Mechanical Engineers
CAP	corrective action program
CFR	*Code of Federal Regulation*
CR	condition report
CREVS	control room emergency ventilation system
EFW	emergency feedwater
ENSA	Equipos Nucleares
gpd	gallons per day
HRA	high radiation area
PI	performance indicator
RPV	reactor pressure vessel
SG	steam generator
SSC	structure, system, and component
TI	temporary instruction