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

August 5, 2005

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 05000313/2005003 AND 05000368/2005003

Dear Mr. Forbes:

On June 23, 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 June 30, 2005, with you and other members of your staff.

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

This report documents four self-revealing findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements; however, because of the very low safety significance and 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 NRC Enforcement Policy. Additionally, this report documents an apparent violation regarding a damaged reactor coolant pump seal which necessitated an unplanned entry into reduced inventory conditions and had potential safety significance of greater than very low significance. Determination of the significance of this finding is currently under review. The safety concern was resolved with replacement of the pump seal. 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 05000313/2005003 and 05000368/2005003
w/Attachment: Supplemental Information

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SISP 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	SPE:DRP/E	C:DRS/OB
JLDixon	ELCrowe	RWDeese	VGGaddy	ATGody
VGGaddy for	**VGGaddy for**	**VGGaddy for**	**NA**	**RELantz for**
8/1/05	8/1/05	8/1/05	7/ /05	8/1/05

C:DRS/PSB	C:DRS/EB1	C:DRS/EB2	C:DRP/E
MPShannon	JAClark	LJSmith	DNGraves
/RA/	**/RA/**	**GDReplogle**	**/RA/**
8/1/05	8/1/05	8/1/05	8/5/05

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-313, 50-368
Licenses:	DPR-51, NPF-6
Report:	05000313/2005003 and 05000368/2005003
Licensee:	Entergy Operations, Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	Junction of Hwy. 64W and Hwy. 333 South Russellville, Arkansas
Dates:	March 25 through June 23, 2005
Inspectors:	J. Adams, Reactor Inspector E. Crowe, Resident Inspector R. Deese, Senior Resident Inspector J. Dixon, Resident Inspector C. Johnson, Senior Reactor Inspector G. Replogle, Senior Reactor Inspector L. Ricketson, Senior Health Physicist J. Reynoso, Reactor Inspector D. Stearns, Health Physicist
Approved By:	David N. Graves, Chief, Project Branch E Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000313/2005003, 05000368/2005003; 3/25/05 - 6/23/05; Arkansas Nuclear One, Units 1 and 2; Refueling and Outage Activities, Surveillance Testing, Event Followup, ALARA Planning and Controls, and Other Activities.

This report covered a 3-month period of inspection by resident inspectors and regional specialist inspectors. The inspection identified four Green noncited violations and one apparent violation with significance yet to be determined. 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: Initiating Events

- Green. A self-revealing noncited violation of Unit 1 Technical Specification 5.4.1, "Procedures," was reviewed by the inspectors when Unit 1 operators secured flow to the auxiliary cooling water system when performing surveillance testing. This resulted in a loss of cooling water to the condensate pumps and increased the potential of a plant transient. This issue involved human performance crosscutting aspects associated with an operator not following a procedure.

  The inspectors determined this finding was greater than minor because it affected the initiating events cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions attributable to human performance error. The inspectors concluded this finding was of very low safety significance after performing a Phase 2 analysis using Appendix A, "Technical Basis For At Power Significance Determination Process," of Manual Chapter 0609, "Significance Determination Process," and the Phase 2 worksheets from "Risk-Informed Inspection Notebook for Arkansas Nuclear One - Unit 1," since the emergency feedwater and high pressure injection systems which would have been relied upon to mitigate a reactor trip transient remained unaffected (Section 1R22).

  Green. The inspectors documented a self-revealing, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, because the licensee failed to correct a 4160 Vac cable failure mechanism (a significant condition adverse to quality). In addition, the licensee failed to properly address industry operating experience on the same topic. The cables were submerged in water but they were not designed for submergence. Consequently, several 4160 Vac service water pump and fire pump motor cables failed in service between 1993 and 2003. The licensee replaced all the vulnerable cables in 2003. This issue had crosscutting aspects associated with problem identification and resolution in that the licensee failed to adequately evaluate the condition.

The failure to take appropriate corrective measures to address a significant condition adverse to quality was a performance deficiency. This finding was more than minor because it affected the initiating events and mitigating system cornerstone objectives of limiting the likelihood of initiating events and ensuring the availability of systems that mitigate plant accidents. The issue required a Phase 3 significance determination because it had screened out of the Phase 2 significance determination as potentially greater than Green. The Phase 3 significance determination concluded that the issue was of very low risk significance (Section 4OA5).

Cornerstone: Mitigating Systems

- TBD. An apparent violation of Unit 2 Technical Specification 6.4.1, “Procedures,” occurred when reactor coolant pump seal injection flow was established with the reactor coolant pump uncoupled from its motor. This activity led to damage of the seal for Reactor Coolant Pump 2P-32C. This damage required conducting an additional reduced reactor coolant system inventory maintenance period to replace the seal. This issue involved human performance crosscutting aspects associated with an inadequate operations procedure that failed to prevent operators from damaging the seal and incomplete communications by engineers that resulted in an inadequate operability evaluation of the seal.

  The inspectors determined this finding was greater than minor because it affected the mitigating systems cornerstone objective of ensuring the availability and reliability of the reactor coolant system inventory, such that the licensee had to enter a higher risk plant operating state to repair the seal. Using the Phase 1 checklist in Appendix G, “Shutdown Operations,” of Manual Chapter 0609, “Significance Determination Process,” the inspectors determined the finding required a Phase 2 analysis and was sent to regional senior risk analysts for risk quantification. This risk quantification had not been performed at the end of this inspection period (Section 1R20).

- Green. A self-revealing noncited violation of Unit 2 Technical Specification 3.0.4 was reviewed by the inspectors when the licensee made an inappropriate mode change without all required equipment being operable. On September 30, 2004, the licensee proceeded from Mode 4 to Mode 3 with an inoperable train of pressurizer proportional heaters. This issue involved problem identification and resolution crosscutting aspects in that operations, engineering, and management personnel did not identify, prioritize, nor evaluate the condition adverse to quality for many years.

The inspectors determined this finding was greater than minor because it affected the mitigating systems cornerstone objective of ensuring the availability and reliability of the pressurizer proportional heaters, such that, if left uncorrected, both banks of pressurizer proportional heaters could have become inoperable. Using the Phase 1 worksheets in Manual Chapter 0609, "Significance Determination Process," the finding was determined to have very low safety significance because mitigating systems were available and it did not affect the likelihood of an external initiating event (Section 4OA3).

Cornerstone: Occupational Radiation Safety [OS]

- Green. The inspector reviewed a self-revealing noncited violation of 10 CFR 20.1501(a) resulting from the licensee's failure to evaluate radiological hazards. Because of an inadequate job planning procedure, the licensee did not evaluate the effect on dose rates caused by the lack of water in the cask loading pit during fuel movement. Consequently, when a fuel assembly was moved near the empty cask loading pit on March 20, 2005, higher than anticipated dose rates were experienced by workers on the spent fuel pool bridge. The licensee was alerted to the problem by workers' alarming electronic dosimeters which measured a maximum dose rate of 220 millirems per hour. This issue involved human performance crosscutting aspects associated with an inadequate job planning procedure.

  The finding is more than minor because it is associated with the occupational radiation safety cornerstone attribute of exposure control and affected the cornerstone objective in that not adequately evaluating the radiological hazards could lead to inadequate radiological controls. Since this occurrence involved workers' unplanned, unintended dose or potential for such a dose that could have been significantly greater as a result of a single minor, reasonable alteration of circumstances, this finding was evaluated with the occupational radiation safety significance determination process. The inspector determined that the finding was of very low safety significance (Green) because it did not involve: (1) as low as is reasonably achievable planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding was entered into the licensee's corrective action program (Section 2OS2).

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 action are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at 100 percent rated thermal power (RTP). Reactor power was lowered to 84 percent RTP on May 6, 2005, to facilitate testing of the main turbine governor valves and returned to 100 percent RTP on May 7, 2005. The unit remained at 100 percent RTP for the remainder of the inspection period.

Unit 2 began the inspection period in a refueling outage with the reactor shut down. The reactor was brought critical on April 10, 2005, the main generator output breakers were closed on April 11, 2005, and the plant achieved approximately 100 percent RTP on April 14, 2005, and remained there for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

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 corrective action program (CAP) to ensure problems were being identified and corrected.

- March 30, 2005, Unit 2 shutdown cooling (low pressure coolant injection) system
- June 13, 2005, Unit 2 emergency feedwater system
- June 16, 2005, Unit 1 high pressure injection system

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

Routine Inspection

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.

- March 30, 2005, Unit 2 east high pressure safety injection, low pressure safety injection, and containment spray pump area, Fire Zone 2007-LL

- April 15, 2005, Unit 2 upper south electrical penetration room and hot instrument shop, Fire Zone 2137-I

- April 15, 2005, Unit 2 access room, pump room, and tank room, Fire Zone 2073-D

- April 15, 2005, Unit 1 lower south piping penetration room, Fire Zone 46-Y

- April 15, 2005, Unit 1 south switchgear room, Fire Zone 100-N

- June 20, 2005, Unit 1 south battery room, Fire Zone 110-L

The inspectors completed six samples.

Annual Inspection

The inspectors observed a fire brigade drill on June 3, 2005, to evaluate the readiness of licensee personnel to prevent and fight fires, including the following aspects: (1) use of protective clothing, (2) use of breathing apparatuses, (3) placement and use of fire hoses, (4) entry into the fire area, (5) use of firefighting equipment, (6) brigade leader command and control, (7) communications between the fire brigade and control room, (8) searches for fire victims and fire propagation, (9) smoke removal, (10) use of prefire plans, and (11) adherence to the drill scenario. The licensee simulated a fire in the north electrical equipment room, Fire Zone 2091-BB, in Electrical Panel 2B53.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

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

C April 13, 2005, Unit 2 auxiliary building corridor on Elevation 354'.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed test data from performance tests and verified the licensee's execution and online monitoring of biofouling controls for the Unit 2 Shutdown Cooling Heat Exchanger. The inspectors verified that: (1) test acceptance criteria and results considered differences between testing and design conditions; (2) inspection results were appropriately categorized against acceptable pre-established acceptance criteria;

(3) the frequency of testing or inspection was sufficient to detect degradation prior to loss of the heat removal function; (4) the test results considered instrument uncertainties; and (5) the licensee had established biofouling controls.

- June 15, 2005, Unit 2 shutdown cooling heat exchanger

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08P)

Procedure 71111.08 requires a minimum sample size of four, one sample for each section (Sections 02.01, 02.02, 02.03, and 02.04). The inspectors fulfilled the requirements of Inspection Procedure 71111.08, "Inservice Inspection Activities."

2.01 Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections, Pressurized Water Reactor Vessel Upper Head Penetrations Inspections, Boric Acid Control

a. Inspection Scope

The procedure requires the review of two to three types of nondestructive examination activities. The inspectors reviewed the records of approximately 24 eddy-current (surface), 3 radiography, and 16 ultrasonic (volumetric) examinations and witnessed the performance of 4 eddy-current and 3 ultrasonic examinations. This sample of nondestructive examination activities is listed in the attachment.

For each of the nondestructive examination activities reviewed and observed above, the inspectors verified that the examinations were performed in accordance with site procedures and the applicable American Society of Mechanical Engineers (ASME) Code requirements.

During the review of each examination, the inspectors verified that appropriate nondestructive examination procedures were used, that examinations and conditions were as specified in the procedure, and that test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors also reviewed documentation to verify that indications revealed by the examinations were dispositioned in accordance with site procedures and the ASME Code specified acceptance standards.

The inspectors verified the certifications of nondestructive examination personnel observed performing examinations or identified during review of completed examination packages.

The inspection procedure requires review of one or two examinations from the previous outage with recordable indications that were accepted for continued service to ensure that the disposition was done in accordance with the ASME Code. There were no apparent recordable indications from other inspection activities found. There were several recordable indications in Unit 2 Steam Generator A (Tubes 100-25 and 109-92) that required evaluation during the previous outage. These were indications that were found in the Unit 2 steam generator during Refueling Outage 2R15. The licensee evaluated the same indications in this current outage (Refueling Outage 2R17) in accordance with their site procedural requirements. The inspectors reviewed both outage results of the examinations and evaluations of any growth that had occurred from Refueling Outage 2R15 to Refueling Outage 2R17. The inspectors reviewed the licensee's corrective action plans and determined it to be appropriate. The inspectors also reviewed recordable indications on nine Unit 2 pressurizer heater sleeve penetrations. These heater sleeve penetrations were repaired with a mechanical nozzle seal assembly (MNSA) which was approved by the NRC until the next refueling outage.

The procedure requires verification of one to three welds that the welding process and welding examinations were performed in accordance with the ASME Code if welding on the pressure boundary for Class 1 or 2 systems has been completed by the licensee. The inspectors reviewed one in-process welding activity on the reactor coolant system (RCS) drain line to the reactor drain tank. The inspectors observed Field Welds FW1C1 and FW5C1 performed on RCS drain line Valve 2RC-5B. Maintenance personnel performed work in accordance with site procedures and the ASME Code requirements.

b. Findings

No findings of significance were identified.

2.02 Pressurized Water Reactor Vessel Upper Head Penetration Inspection Activities

The procedure requires that Temporary Instruction (TI) 2515/150, “Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009),” be reviewed before completion of this section. Credit for this section will be taken for completion of TI 2515/150.

2.03 Boric Acid Corrosion Control Inspection Activities (Pressurized Water Reactors)

a. Inspection Scope

The inspection procedure requires a review of one to three engineering evaluations performed for boric acid found on RCS piping and components. The inspectors reviewed one interim and one final disposition engineering evaluation performed for boric acid found on pressurizer piping and components and the outage boric acid walkdown results. The inspectors performed an as-found walkdown of the pressurizer penetrations, top and bottom, both before and after insulation removal. In addition, the inspectors also performed a Mode 3 walkdown. The inspectors determined that the licensee was identifying boric acid during the walkdown and documenting the location for a final engineering disposition evaluation.

The procedure requires review of one to three corrective actions performed for evidence of boric acid leaks. The inspectors reviewed four condition reports (CRs) from Refueling Outage 2R17 relating to leakage found on Unit 2 pressurizer heater sleeves. The inspectors determined that the identified boric acid leaks have been evaluated and corrected through the licensee's corrective action process. The CRs that the inspectors reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

2.04 Steam Generator Tube Inspection Activities

b. Inspection Scope

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

The inspectors observed and reviewed the results of the in-situ test performed on Unit 2 Steam Generator A, Tube 70-169. The inspectors concluded that the licensee performed the required test at various pressures in accordance with site procedures and EPRI guidelines.

The inspection procedure specified comparing the estimated size and number of tube flaws detected during the current outage against the previous outage operational assessment predictions to assess the licensee's prediction capability.

The inspectors selected several flaws (Tubes 100-25, 102-25, and 109-92) that were identified in the previous outage and compared the current outage results to determine if flaw growth had occurred. The inspectors verified that there was some minor growth in the flaws reviewed, but well below the 40 percent established criteria. The inspectors also verified that the licensee used criteria specified in steam generator program guidelines initiative NEI 97-06, "Industry Steam Generator Program Guidelines," for degradation assessment.

The inspection procedure requires confirmation that the steam generator tube eddy-current test scope and expansion criteria meet Technical Specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors verified that the eddy-current test scope and expansion criteria did meet Technical Specifications and the EPRI guidelines.

The inspection procedure required confirmation that the licensee inspected all areas of potential degradation, especially areas which were known to represent potential eddy-current test challenges (e.g., top-of-tube sheet, tube support plates, and U-bends). The inspectors confirmed that the licensee inspected all areas of potential degradation, specifically in the U-bend area due to wear caused by the antivibration bars.

The inspection procedure requires, if steam generator leakage greater than 3 gallons per day is identified, assessment of whether the licensee has identified a reasonable cause for this leakage based on inspection results. The inspectors reviewed an assessment conducted by the licensee which included postshutdown actions and revised degradation assessment. Inspection results indicated that the licensee had identified a reasonable cause for the leakage and preventive actions were in place.

The inspection procedure requires, if the licensee had identified loose parts or foreign material on the secondary side of the steam generator, focus on licensee corrective actions. The inspectors reviewed several CRs regarding loose parts identified on the secondary side of the Unit 2 steam generators. The licensee's corrective actions were appropriate, including the identification and removal of most loose parts found. The licensee had taken or planned appropriate repairs for both Unit 2 steam generator tubes affected by loose parts.

b. Findings

No findings of significance were identified.

2.05 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed selected inservice inspection related CRs issued during current Refueling Outage 2R17. The review served to verify that the licensee's corrective action process was being correctly utilized to identify conditions adverse to quality and that those conditions were being adequately evaluated, corrected, and trended. The inspectors confirmed that the licensee's threshold for initiating CRs was low, thereby, capturing most deficiencies identified in the inservice inspection program. The inspectors also verified that corrective actions were being appropriately addressed.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On May 19, 2005, the inspectors observed testing and training of Unit 2 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. Training Scenario ASPGLOR050401, “Functional Recovery,” Revision 0, was used and involved degrading electrical grid conditions leading to a loss of offsite power complicated by a loss of feeding capability to the steam generators.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- June 8, 2005, Units 1 and 2, 4160 Volt ac electrical distribution
- June 13-15, 2005, Unit 2 containment 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 three 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.

- May 9, 2005, Unit 2 planned maintenance during the week
- May 16, 2005, Unit 1 planned maintenance during the week
- May 31, 2005, Units 1 and 2 planned maintenance during the week

The inspectors completed three samples.

Emergent Work Control

For the three emergent work activities listed below, the inspectors: (1) verified that the licensee performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) verified that emergent work-related activities such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications, and equipment restoration did not place the plant in an unacceptable configuration; and (3) reviewed the CAP to determine if the licensee identified and corrected risk assessment and emergent work control problems.

- April 11, 2005, Unit 2 Service Water Pump 2P-4A strainer cleaning during severe thunderstorm and tornado warnings
- April 29, 2005, Units 1 and 2 turbine building to auxiliary building Hatch 493 pulled during severe weather conditions.
- June 17, 2005, Unit 2 high pressure safety injection system maintenance on Valve 2SI-12

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

For the six operability evaluations listed below, the inspectors: (1) reviewed plants 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.

- CR-ANO-2-2004-2173 April 5, 2005, Unit 2 auxiliary feedwater pump
- CR-ANO-2-2005-1048 April 6, 2005, Unit 2 service water hydro-lazing activities and as-left service water flow testing
- CR-ANO-1-2005-0653 May 4, 2005, Unit 1 Emergency Diesel Generator K-4B reactive loading swings during monthly testing
- CR-ANO-2-2005-1505 May 4, 2005, Unit 2 auxiliary feedwater pump discharge piping snubber
- CR-ANO-2-2004-1516 May 13, 2005, Unit 2 pressurizer insurge and outsurge transients as part of the original design basis
- CR-ANO-2-2005-1678 June 1, 2005, Unit 2 pressurizer level transmitter momentarily spiking low

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

Cumulative Review of the Effects of Operator Workarounds. On May 13, 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.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the two 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.

- June 3, 2005, Unit 2 Containment Spray Pump 2P-35B
- June 10, 2005, Unit 2 Containment Sump Valve 2CV-5650-2

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense in depth commensurate with the outage risk control plan and compliance with the Technical Specifications: (1) the risk control plan, (2) tagging/clearance activities, (3) RCS instrumentation, (4) electrical power, (5) decay heat removal, (6) spent fuel pool cooling, (7) inventory control, (8) reactivity control, (9) containment closure, (10) reduced inventory conditions, (11) refueling activities, (12) heatup activities, and (13) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities.

- March 25 through April 11, 2005, completion of Unit 2 planned Refueling Outage 2R17

The inspectors completed one sample.

b. Findings

Introduction. The inspectors identified an apparent violation (AV) of the Unit 2 Technical Specification 6.4.1, “Procedures,” for an inadequate procedure related to the alignment of reactor coolant pump (RCP) seal injection flow when the pump was uncoupled.

Description. During Unit 2 Refueling Outage 2R17, on March 13, 2005, operators commenced filling of the RCS after a period of reduced inventory to install steam generator nozzle dams. Section 8.0, “RCS Fill Operations,” of Procedure 2103.002, “Filling and Venting the RCS,” Revision 39, instructed operators to align seal injection to all RCPs as part of the fill evolution. In their efforts to align seal injection to RCP 2P-32C, operators encountered difficulties attaining adequate flow, so they adjusted seal flow but observed abnormal seal pressures. The licensee had replaced the motor for RCP 2P-32C earlier in the outage and the pump and motor for RCP 2P-32C were still uncoupled. The pump and motor should have been recoupled prior to initiating seal injection flow to the pump. The observed abnormal pressures and difficulty in establishing seal injection flow were captured in the licensee’s corrective action program in CR ANO-2-2005-0545. In the operability evaluation for this CR, engineers declared the seal operable. Their determination was, in part, based on a discussion with the seal vendor. However, in this conversation, the engineers did not make it clear to the vendor that the pump was uncoupled during the periods of observed abnormal pressure. Replacement of the seal would have been desired at this time, since the overall risk of the outage would have been minimized because the plant was defueled at this time.

The damaged seal went undetected until April 4, 2005, when operators commenced filling the RCS in preparation for returning to power operations. Initial RCS level was 84 inches, which was just below the reactor vessel flange level. At approximately 188 inches in the RCS, operators noticed an estimated 26 gallon per minute leak from the RCP seal and secured the RCS fill activity. The RCS was subsequently drained to the 90-inch level.

The licensee entered reduced RCS inventory conditions at 11:01 p.m. on April 4, 2005, and continued draining to seal replacement level. They remained in reduced inventory to replace the seal until 5:12 a.m. on April 6, 2005, (approximately 30 hours). During this reduced inventory activity, RCS temperatures were controlled between 112EF and 129EF and time-to-boil was approximately 1 hour. The inspectors considered this an unplanned entry into reduced RCS inventory. Additionally, the inspectors considered the lower inventory of the RCS to be an affected mitigating system for the prevention of boiling conditions in the reactor vessel. The seal replacement that resulted in the unplanned reduced inventory condition was successfully completed.

Analysis. The inspectors determined that this issue is more than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors used Appendix G, “Shutdown Operations Significance Determination Process,” of Manual Chapter 0609, “Significance Determination Process,” to further determine the significance of this finding. Using Checklist 3, “PWR Cold Shutdown and Refueling Operation - RCS Open and Refueling Cavity Level < 23' Or RCS Closed and No Inventory in Pressurizer, Time to Boiling < 2 hours,” in Attachment 1, “Phase 1 Operational Checklists for both PWRs and BWRs,” of Appendix G of Manual Chapter 0609, the inspectors determined this finding required quantitative assessment and referred it to regional senior risk analysts for quantification. This quantification risk assessment had not been completed at the end of this inspection period and could potentially be greater than very low safety significance. This issue involved human performance crosscutting aspects associated with an inadequate operations procedure that failed to prevent operators from damaging the seal and incomplete communications by engineers that resulted in an inadequate operability evaluation of the seal.

Enforcement. Unit 2 Technical Specification 6.8.1, “Procedures,” requires that the licensee establish and implement written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, which required procedures for abnormal, off normal, or alarm conditions. Filling and venting the RCS is one of the listed procedures. Contrary to the above, before April 4, 2005, Procedure 2103.002, “Filling and Venting the Reactor Coolant System,” did not provide guidance for ensuring seal injection is isolated anytime a pump is uncoupled. Because this procedure was inadequate, operators damaged the RCP seal requiring an additional entry into a reduced RCS inventory condition during the refueling outage. The pump seal was subsequently replaced and seal injection was properly established so no safety issue currently exists. Pending determination of the final safety significance of this issue, this violation is being treated as an AV consistent with Section VI.A of the NRC Enforcement Policy: AV 05000368/2005003-01, “Inadequate Procedure Leads to Reactor Coolant Pump Seal Damage.”

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

For the four surveillances listed below, the inspectors reviewed the Updated Final Safety Analysis Report, procedure requirements, and Technical Specifications to ensure that the four below listed surveillance activities demonstrated that 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 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.

- February 21, 2005, Unit 2 Emergency Control Room Ventilation Fan 2VSF-9 flow measurement methodology
- April 25, 2005, Unit 1 Service Water Pump P-4B quarterly surveillance test (inservice testing)
- April 26, 2005, Unit 2 service water full flow testing
- May 25, 2005, Unit 2 containment spray and sump valve stroke test

The inspectors completed four samples.

b. Findings

Introduction. The inspectors reviewed a Green noncited violation (NCV) of Unit 1 Technical Specification 5.4.1, "Procedures," for failure to follow a Unit 1 service water surveillance procedure which led to securing cooling water to the condensate pump motors.

Description. On April 24, 2005, Unit 1 operators were performing surveillance testing on Service Pump P-4B in accordance with Supplement 2 of Operating Procedure 1104.029, "Service Water and Auxiliary Cooling Water," Revision 56. Operators had reached the last portion of the procedure where the service water and auxiliary cooling water (ACW) systems were being returned to their normal operating lineups. Step 2.18.2 of the procedure instructed the operators to verify service water loop crosstie Valves CV-3640, CV-3642, CV-3644, and CV-3646 were open prior to securing Service Water Pump P-4B. This action ensured that sufficient pressure for ACW flow would be maintained upon securing Pump P-4B. The operator secured the service water pump without checking the position of these valves and, as a result, ACW cooling water flow to components was lost. Two of these loads, Control Room Chiller VCH-2B and Sample Chiller VCH-7, secured automatically. The shift manager noticed low ACW loop pressure immediately and had an operator open one of the service water loop crosstie valves, which restored cooling water flow to the ACW system components. Upon further review, the inspectors noted that ACW flow to the main condensate pumps had been lost which, had the loss been sustained, could have tripped the main feed pumps and subsequently initiated a reactor trip.

Analysis. The failure to follow procedure for the service water surveillance test was considered to be a performance deficiency. This finding is more than minor because it affected the initiating events cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions attributable to human performance error. Using Manual Chapter 0609, "Significance Determination Process," the inspectors determined this finding was a transient initiator which contributed to the likelihood that mitigation equipment (the power conversion system) would not be available. As a result, the inspectors performed a Phase 2 analysis using Appendix A, “Technical Basis For At Power Significance Determination Process,” of Manual Chapter 0609, “Significance Determination Process,” and the Phase 2 worksheets from “Risk-Informed Inspection Notebook for Arkansas Nuclear One - Unit 1.” The inspectors assumed that all condensate pumps would fail rapidly after a sustained loss of ACW flow and that operators under all circumstances would not be able to restore this flow to the condensate pumps rapidly enough to prevent this failure, even though the operators did restore the flow quickly enough in this instance. From this Phase 2 analysis, the inspectors determined that this finding was of very low safety significance (Green) because the emergency feedwater and high pressure injection systems which would have been relied upon to mitigate a reactor trip transient remained unaffected. This issue involved human performance crosscutting aspects associated with control room operators not following a procedure.

Enforcement. Unit 1 Technical Specification 5.4.1, “Procedures,” requires that written procedures shall be implemented covering the procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Service water system functional testing is one of the procedures listed in the regulatory guide. Contrary to the above, on April 25, 2005, during surveillance testing of the Unit 1 service water system, operators did not ensure Valves CV-3640, CV-3642, CV-3644, and CV-3646 were open prior to securing Service Water Pump P-4B in accordance with step 2.18.2 of Supplement 2 to Procedure 1104.029, “Service Water and Auxiliary Cooling Water,” Revision 56. Because of the very low safety significance and because the licensee included this condition in their CAP as CR ANO-1-2005-0629, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000313/2005003-02, “Failure to Follow a Service Water Surveillance Procedure.”

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

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

- May 19, 2005, Unit 2 adjustment of containment sump isolation Valve 2CV-5650-2 open position from fully opened to 60 percent opened

The inspectors completed one sample.

b. Findings

No findings of significance was identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The drill listed below contributed to drill/exercise performance and emergency response organization performance indicators. The inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in classification, notification, and protective action requirements development activities; (2) compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee is properly identifying failures; and (3) determined whether licensee performance is in accordance with the guidance of the Nuclear Energy Institute 99-02 document's acceptance criteria.

- June 1, 2005, emergency response organization drill with simulated offsite release initiated from the Unit 1 simulator and activating the Technical Support Center, Emergency Operations Facility, and Operations Support Center.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS2 As Low as is Reasonably Achievable (ALARA) Planning and Controls (71121.02)

a. Inspection Scope

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

- Current 3-year rolling average collective exposure

- Six work activities from previous work history data which resulted in the highest personnel collective exposures

- Site-specific trends in collective exposures, plant historical data, and source-term measurements

- Site-specific ALARA procedures

- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements

- Intended versus actual work activity doses and the reasons for any inconsistencies

- Dose rate reduction activities in work planning

- Postjob (work activity) reviews

- Assumptions and basis for the current annual collective exposure estimate, the methodology for estimating work activity exposures, the intended dose outcome, and the accuracy of dose rate and man-hour estimates

- Method for adjusting exposure estimates, or replanning work, when unexpected changes in scope or emergent work were encountered

- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding

- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

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

- Resolution through the corrective action process of problems identified through postjob reviews and postoutage ALARA report critiques

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

- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

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

- Special reports related to the ALARA program since the last inspection

The inspector completed 12 of the required 15 samples and 4 of the optional samples.

b. Findings

Introduction. The inspector reviewed a self-revealing NCV of 10 CFR 20.1501(a) resulting from the licensee's failure to evaluate radiological hazards. The violation had very low safety significance.

Description. On March 20, 2005, as workers on the spent fuel pool bridge started to inspect fuel assemblies, dose rates increased unexpectedly, causing the electronic dosimeters of two workers to alarm with a maximum dose rate of 220 millirems per hour. A radiation protection technician responded to the alarms and evacuated the workers. During a review of the occurrence, the licensee determined that the dose rates rose as the first fuel assembly was moved near the cask loading pit because the pit was not filled with water as it usually was during this operation. The licensee calculated that the maximum accessible dose rate was 1,400 millirems per hour. The empty cask loading pit was not identified by the job planners and precautions were not considered. The licensee determined the root cause was "Procedures associated with fuel handling do not provide instructions, limits, and precautions that are adequate to prompt a comprehensive evaluation of shielding requirements during spent fuel handling activities."

Analysis. The failure to evaluate the radiological hazards associated with the lack of water shielding in the cask loading pit is a performance deficiency. The finding is more than minor because it is associated with the occupational radiation safety cornerstone attribute of exposure control and affected the cornerstone objective, because not adequately evaluating the radiological hazards could lead to inadequate radiological controls. Since this occurrence involved workers' unplanned, unintended dose or potential for such a dose that could have been significantly greater as a result of a single minor, reasonable alteration of circumstances, this finding was evaluated with the occupational radiation safety significance determination process. The inspector determined that the finding was of very low safety significance (Green) 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. This issue involved human performance crosscutting aspects associated with an inadequate job planning procedure.

Enforcement. 10 CFR 20.1501(a) requires that each licensee make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and that are reasonable under the circumstances to evaluate the extent of radiation levels, concentrations or quantities of radioactive materials, and the potential radiological hazards that could be present. Pursuant to 10 CFR 20.1003, a "survey" means an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. 10 CFR 20.1201(a) states, in part, that the licensee shall control the occupational dose to individual adults. The licensee violated the requirements of 10 CFR 20.1501(a) when it did not evaluate the extent of radiation levels in order to assure compliance with 10 CFR 20.1201(a). This finding was entered into the licensee's CAP (CR ANO-2-2005-00730). Because the failure to perform a radiological survey is of very low safety significance and has been entered into the licensee's CAP, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000368/2005003-03, Failure to evaluate radiological hazards.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

.1 Annual Sample Review

a. Inspection Scope

The inspectors chose one issue for more in-depth review to verify that licensee personnel had taken corrective actions commensurate with the significance of the issue. The issue and the basis for its selection is described below:

- On March 20, 2005, a fuel handling team moved a fuel assembly into an area of the Unit 2 spent fuel pool adjacent to the cask loading pit which was drained, resulting in unplanned exposure to members of the fuel handling team. This sample was chosen due to the licensee's upcoming dry fuel loading campaign planned later in the summer and recent events involving improperly posted radiological areas.

When evaluating the effectiveness of the licensee's corrective actions for this issue, 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 one sample.

b. Findings

No findings of significance were identified.

.2 Cross-References to Problem Identification and Resolution (PI&R) Findings Documented Elsewhere

Section 4OA3 documents a condition where operations, engineering, and management personnel did not properly identify, prioritize, nor evaluate a condition adverse to quality regarding the pressurizer proportional heaters for many years.

Section 4OA3 documents a condition where the licensee failed to correct a significant condition adverse to quality in a manner timely enough to preclude RCS pressure boundary leakage.

Section 4OA5 documents a condition where licensee personnel failed to perform an evaluation related to repetitive failures of submerged safety-related cables.

.3 Semiannual Trend Review

a. Inspection Scope

On June 23, 2005, the inspectors completed a semiannual review of licensee internal documents, reports, and audits to identify trends that might indicate the existence of more significant safety issues. The inspectors reviewed the following:

- System health indicators
- Temporary alterations
- CRs
- Work requests
- Maintenance rule failures

b. Findings

- During the first 6 months of 2005, licensee personnel documented eight (three on Unit 1 and five on Unit 2) instances of foreign material in the station's spent fuel pools. In addition, licensee personnel documented three instances where administrative controls which were corrective actions from CR ANO-C-2003-1025 were not being followed. None of these instances actually challenged plant safety, but the number of documented instances was indicative of a need for improved control of foreign material in Units 1 and 2 spent fuel pools. Licensee management was aware of this performance issue and have implemented corrective actions as identified in CR ANO-C-2005-0427.

.4 ALARA Inspection

Section 2OS2 evaluated the effectiveness of the licensee's PI&R processes regarding exposure tracking, higher than planned exposure levels, and radiation worker practices. The inspector reviewed the corrective action documents listed in the attachment against the licensee's PI&R program requirements. No findings of significance were identified.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report (LER) 05000368/2004-003-00. Entry into an Operational Mode Prohibited by Technical Specification due to Inoperable Pressurizer Proportional Heaters

a. Inspection Scope

The inspectors reviewed the LER, corrective action documents, Unit 2 station operating logs, plant procedures, and licensing memoranda. This review verified that the cause of the pressurizer proportional heater breakers tripping open during the September 27 through October 2, 2004, forced outage was identified and corrective actions were appropriate. The inspectors also reviewed the corrective action database for other past failures related to the proportional heater breakers. An additional NCV that resulted from the inspection on the pressurizer proportional heaters is documented in NRC Inspection Report 05000313/2004005; 05000368/2004005, Section 1R12.1.

b. Findings

Introduction. A Green self-revealing NCV of Unit 2 Technical Specification 3.0.4 was reviewed by the inspectors when the licensee made an inappropriate mode change without all required equipment being operable.

Description. On September 30, 2004, the licensee found pressurizer proportional Heater 2PP6 Breakers 1 and 3 tripped free as part of a containment closeout checklist walkdown. An operator received permission from the shift manager to shut the breakers and monitor if they immediately opened. The operator shut the breakers, noted they stayed shut, and reported back to the shift manager. The shift manager then proceeded on with the plant startup and changed from Mode 4 to Mode 3. The oncoming shift manager questioned the efforts done to determine that the proportional heaters were operable. Upon recognizing that no troubleshooting efforts were performed, the oncoming shift manager, along with the assistant operations manager, ordered the 18-month surveillance test to ensure that the proportional heaters were operable per Technical Specification 3.4.4. Upon completion of the test, the licensee discovered that the two breakers had tripped again, resulting in that heater bank not meeting the Technical Specification requirement of 150 kilowatts. Upon investigation, the licensee found two loose connections on Breaker 3. Breaker 1 is physically located above Breaker 3 so it is susceptible to conductive and radiative heat from Breaker 3. CR ANO-2-2004-1716 documents the licensee's root cause evaluation report and the failure modes analysis, identifying the most probable cause of the failures to be repeated thermal cycles on the breaker to bus bar connections caused by the variable current output to the heater elements.

In addition, when performing the 18-month surveillance test on the other bank of heaters, the licensee discovered that 2PP5 Breaker 2 was also tripped, documented in CR ANO-2-2004-1727. While no loose connections were observed on Heater 2PP5 Breaker 2, the bus connections were enlarged to match that of the other breaker connections, see NRC Inspection Report 05000313/2004005; 05000368/2004005, Section 1R12.1, for further information. On February 7, 2004, a similar event occurred during a forced plant shutdown when the licensee found Heater 2PP6 Breaker 1 tripped free, documented in CR ANO-2-2004-0223. Again, the licensee action was to shut the breaker and verify it remained shut before continuing on with the plant startup. This was a missed opportunity by the licensee to identify this inadequate operability judgement.

It had been the long-standing practice of the licensee to re-close the breakers and declare them operable without any further investigations. Procedure 1015.036, "Containment Building Closeout," specifically lists the pressurizer proportional heater breakers as items to check before closing out containment. This step was placed in the procedure as a result of the repeated tripping of the breakers and allowed for shutting the breaker if found in the tripped condition. Procedure 2107.001, "Electrical System Operations," contains guidance on what actions are required if a breaker is found in the tripped condition, but did not specifically address the type of breaker used for the pressurizer proportional heaters. However, the general guidance in Procedure 2107.001 was if a breaker is found tripped, the cause of the trip must be known and corrected, or permission from the shift manager obtained under emergency conditions, before it can be reset. The licensee's alarm response procedure generally directs that the breaker can be reset once if it was found in the tripped condition. Finally, the procedure for operability and reportability, Procedure 1015.047, "Condition Reporting Operability and Immediate Reportability Determinations," requires that a tripped breaker have some type of testing performed before declaring it operable. This all lends itself to the licensee having conflicting and inadequate procedures and guidance for how to respond to pressurizer proportional heater breaker trips. This has been identified as a weakness in operability determinations and has been captured in CR ANO-C-2004-1791. The licensee replaced the power panels and the breakers for the pressurizer proportional heaters during current Refueling Outage 2R17 to address the operable but degraded classification of the breakers.

Analysis. The inspectors determined this finding was greater than minor because it affected the mitigating systems cornerstone objective of ensuring the availability and reliability of the pressurizer proportional heaters, such that, if left uncorrected, both banks of pressurizer proportional heaters could have become inoperable. Additionally, a mode change was performed without all required equipment (pressurizer heaters) being operable. Using the Phase 1 worksheets in Manual Chapter 0609, “Significance Determination Process,” the finding was determined to have very low safety significance because other mitigating systems were available and it did not affect the likelihood of an external initiating event. This issue involved PI&R crosscutting aspects in that operations, engineering, and management personnel did not properly identify, prioritize, nor evaluate a condition adverse to quality associated with pressurizer heater breakers.

Enforcement. Unit 2 Technical Specification 3.0.4 requires that entry into an operational mode shall not be made when the conditions of the limiting condition for operation are not met, and the associated action requires a shutdown if they are not met within a specified time interval. Contrary to the above, on September 30, 2004, the licensee entered Mode 3 without having both banks of pressurizer proportional heaters capable of suppling 150 kilowatts each per Technical Specification 3.4.4. Thus not all the required equipment was operable. Because of the very low safety significance and because the licensee included this condition in their CAP as CR ANO-2-2004-1716, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000368/2005003-04, Inappropriate Mode Change Without All Required Equipment Being Operable.

.2 (Closed) LER 05000368/2005-001-00. RCS Pressure Boundary Leakage due to Primary Water Stress Corrosion Cracking of Pressurizer Heater Sleeves

Between March 9-15, 2005, the licensee identified a total of 10 leaking pressurizer penetrations (nine pressurizer heater sleeves and one previously repaired Alloy 600 nozzle). A detailed review of the results of the inspection is documented in Section 4OA5.2 of this report. In addition, the inspectors identified three violations of minor significance not subject to enforcement action in accordance with Section IV of the NRC’s Enforcement Policy. The first issue resulted from not identifying a leaking nozzle during the qualified inspector inspection. The leak was subsequently identified by a technician conducting a survey for installation of temporary shielding when he observed indications of wetting. The licensee was planning to issue an operating experience report to the industry about wetting as an indication of a leak as outlined in CR ANO-2-2005-0607. The second issue resulted from the final as-left diameter of X-1 penetration being slightly larger than the design called for, resulting in the calculation having to be revised. This issue was captured in CR ANO-2-2005-1309. The third issue involved the MNSA installation drilling, bolting, and torquing not satisfying design requirements as documented in CRs ANO-2-2005-0627 and -1269. Finally, this pressure boundary leakage is an additional example of a previously issued NCV for ineffective corrective actions to prevent primary water stress corrosion cracking. The final corrective actions for the previously identified NCV will not be completed until a subsequent refueling outage. For further information of this previously dispositioned violation, see NRC Inspection Report 05000313/2004002; 05000368/2004002, Section 4OA3.2, NCV 05000368/2004-02, "Ineffective Corrective Actions to Prevent Recurrence of primary water stress-corrosion cracking of Alloy 600 Material." This LER is closed.

4OA4 Crosscutting Aspects of Findings

Cross-Reference to Human Performance Findings Documented Elsewhere

Section 1R20 describes a condition where an inadequate operations procedure led to RCP seal damage, necessitating an additional reduced inventory period during a refueling outage. The same finding also documents incomplete communications between site engineers and the seal vendor in the preparation of an operability evaluation which led the licensee to an incorrect conclusion regarding seal integrity.

Section 1R22 describes a condition where Unit 1 operators failed to follow a surveillance procedure which created an increased probability of a reactor trip transient initiator by removing cooling water to the main condensate pump motors.

Section 2OS2 describes a condition where unexpectedly high dose rates were encountered during spent fuel movement due to an inadequate job planning procedure.

4OA5 Other Activities

.1 Reactor Vessel Head and Head Inspections (TI 2515/150)

a. Inspection Scope

The inspection procedure requires the observation of visual examination or review of postexamination videotape of the upper head penetrations. The inspectors performed a bare metal visual walkdown of the Unit 2 reactor pressure vessel head to assess its overall condition. The reactor pressure vessel head was absent of any appreciable foreign material. The inspectors also performed a 100 percent review of the licensee's video tape recording of the bare metal visual inspection of all the penetrations on the reactor pressure vessel head. The inspectors verified that activities performed on the reactor pressure vessel head penetrations were consistent with the requirements of NRC Order EA-03-009, "Reactor Pressure Vessel Head Inspection Requirements for Pressurized Water Reactors." 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. The CRs that the inspectors reviewed are listed in the attachment.

The procedure requires that, if the licensee is performing nonvisual nondestructive examination of the reactor vessel head, the inspectors should review a sample of these examinations. TI 2515/150 requires review of 10 percent of vessel head nozzle volumetric examinations and 5-10 percent of nozzle and/or J-groove surface examinations. The inspectors reviewed volumetric and surface examinations of nine control element drive mechanisms of 89 nozzles, including the J-groove weld surface. The inspectors also verified that examination methods used were capable of identifying stress corrosion cracking. The inspectors observed the ultrasonic and eddy-current examinations of two control element drive mechanism penetrations. The inspectors verified that activities performed on the vessel upper head penetrations were consistent with licensee commitments.

The inspectors observed two nondestructive examinations performed at the vessel head from remote video feeds at the collection and analysis stations. A vendor (Wesdyne International) performed the nondestructive automated examinations for the incore instrument and control element drive mechanism nozzles. The inspectors examined ultrasonic and eddy-current data from 9 of 81 control element drive mechanism nozzles. The inspectors verified that qualified personnel performed the examinations in accordance with approved procedures. Examinations reviewed are listed in the attachment.

The inspectors reviewed the certification records for personnel performing the automated ultrasonic and eddy-current examinations and data analysis performed on the control element drive mechanism.

The procedure requires review of one or two examinations from the previous outage with recordable indications from surface and volumetric examinations that have been accepted by the licensee for continued service if applicable. There were no volumetric or surface examinations with recordable indications on the reactor pressure vessel penetrations from the previous outage.

The procedure requires review of one or two ASME Section XI Code repairs done as a result of volumetric and surface examinations. There were no repairs on the reactor vessel head to review this Refueling Outage 2R17.

The procedure requires review of 5-10 percent of the reactor pressure vessel head bare metal visual examination and 3-5 vessel head penetration nozzle examinations. The inspectors performed a bare metal visual walkdown of the reactor pressure vessel head to assess its overall condition, generally quite clean. The inspectors also performed a 100 percent review of the licensee’s video tape recording of the bare metal visual inspection of all the penetrations on the reactor pressure vessel head. The inspectors verified that: (1) a clear 360E observation of the nozzles was completed; (2) no evidence of cracking or boric acid crystals were present; (3) there were no boron deposits, debris, or insulating material which masked the ability to identify the existence of boric acid; and (4) there were no structural interferences which impeded the ability to complete the bare metal visual inspections. The inspectors determined that the licensee had procedures in place to identify leakage from pressure retaining components located above the reactor pressure vessel head. The inspectors reviewed several CRs related to problems encountered during the examination process. The inspectors determined that the licensee’s threshold for initiating CRs was low, thereby, capturing most deficiencies identified, and that corrective actions were being appropriately addressed. The CRs that the inspectors reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.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 2 to determine whether the inspections by the licensee are consistent with the licensee's response to 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. The licensee submitted LER 2005-001-00 to document the RCS pressure boundary leakage from the pressurizer nozzles, see Section 4OA3 for a discussion of minor issues. The licensee's ultimate corrective action repair plan is the replacement of the pressurizer with a new pressurizer made with Alloy 690 during the next refueling outage (2R18).

(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.)

There were two separate types of individuals involved with the inspection, responsible engineers and Level II VT-2 qualified inspectors. The responsible engineers used to complete Procedure 2311.009, "ANO Unit 1 and Unit 2 Alloy 600 Inspection," 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. The Level II VT-2 qualified inspectors are governed by Procedure CEP-NDE-0112, "Program Section for Certification of Visual Testing Personnel," Revision 1. All ANO VT-2 inspectors are qualified to Level II. Level II VT-2 inspector training consists of the following: (1) vision testing for acuity and color; (2) codes, standards, and specifications; (3) terms and definitions; (4) visual examination equipment; (5) reporting, recording, and evaluation of inspection results; (6) procedure training applicable to VT examinations; (7) minimum level of experience hours; (8) specific exam to demonstrate knowledge of ASME Section XI and plant procedures; and (9) performance evaluations not to exceed every 12 months.

(b) Performed in accordance with demonstrated procedures?

Yes, Procedure 2311.009, "ANO Unit 1 and Unit 2 Alloy 600 Inspections," Revision 7, and its attachments, is the one previously utilized that identified heater sleeve leakage. However, this was the first time that this revision had been used, it incorporated lessons learned from the Unit 1 upper and lower head inspections during 2004. Unfortunately, two heater nozzles were missing from the procedure, CR ANO-2-2005-0566. See Section 4OA7 and Item d below for more information.

(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. The CRs that the inspectors reviewed are listed in the attachment.

(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 vertically from the bottom of the vessel surrounding the surge nozzle. For the initial walkdown the insulation was still in place and signs of white staining were present on the outside of the insulation and between the insulation and the bottom of the pressurizer. The white substance was seen in the general area around the 0E azimuth and did not have the appearance of boron crystals; it had the appearance of being light and fluffy. The substance was determined to be lithium borate and was attributed to the leakers that were identified during the inspection, CR ANO-2-2005-0607. For the bare metal visual examination, the as-found condition when the insulation was removed was generally clean. The inspectors visually observed the licensee perform the initial walkdown with the insulation in place, as well as take samples of the white substance. Additionally, the inspectors also accompanied the Level II VT-2 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 Level II VT-2 inspector and by a responsible engineer. The penetrations that could not be declared satisfactory at the time of the inspection were documented by digital camera snapshots, CR ANO-2-2005-0506. The inspectors performed an independent direct visual inspection to review the categorization of the licensee's inspection results and to verify the accuracy of the digital snapshots to the as-found condition. After this inspection had occurred and the bottom of the pressurizer had been cleaned and decontaminated, a health physics technician surveying for temporary shield installation discovered an additional leaking nozzle, indications of wetting, that had been previously declared satisfactory, CRs ANO-2-2005-0569 and -0607.

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

The nozzles that were inspected were directly inspected 360E around the circumference. However, one heater nozzle and two instrument lines were not definitively inspected prior to cleaning and decontamination, CR ANO-2-2005-0566. Another nozzle was also not inspected at the same time as the above mentioned, but since this nozzle had a MNSA it was also covered by Procedure 5120.243, "Unit 2 - Post Outage Pressure Test," and as a result did receive a Level II VT-2 inspection, CR ANO-2-2005-0489. The licensee concluded that the three missed penetrations did receive some level of review, by conducting interviews with personnel that performed the inspections on the bottom of the pressurizer, just not at the level of a Level II VT-2 inspection.

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

Yes, the licensee did in fact identify small boron deposits. The licensee also performed subsequent followup examinations to determine the characterization of the flaws. See Item h below for more information.

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

The licensee discovered nine leaking heater nozzles and one previously repaired Alloy 600 plug. The licensee determined these deficiencies by direct inspection through the observance of boric acid deposits or indications of wetting. The licensee's corrective action plan is documented in CR ANO-2-2005-0607 and was based on the fact that only axial cracks in the 10 leaking penetrations were observed. The inspectors reviewed the ultrasonic and eddy-current examination data to ensure that the indications were appropriately classified, see Section 1R08.02.01 and Item h below for more information.

(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. However, issues between the ultrasonic and eddy-current data collection and analysis and the temper bead repair of the previously repaired Alloy 600 plug is discussed in Item h below.

(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 inspectors observed the nondestructive examinations of four pressurizer heater sleeves and reviewed the examination results of nine other heater sleeves, including the four that were observed. These nine pressurizer heater sleeves were identified during various walkdowns as leakers and entered into the licensee's CAP. The licensee initially attempted to perform ultrasonic examinations of the heater sleeves, but had difficulty obtaining usable data due to dimensional and distortion problems with the probe and penetrations. The licensee was able to obtain data on three heater sleeves, but the other six heater sleeves were not able to be obtained and analyzed. The licensee decided to use eddy-current examination on all nine heater sleeves to ensure that the flaws were appropriately categorized. The licensee performed eddy-current examinations on these nine heater sleeves and confirmed the presence of through-wall defects. Two nozzles had multiple through-wall defects. These defects were repaired with an MNSA clamp. Penetration X-1 could not be ultrasonic or eddy-current examined due to its distortion; however, the licensee was able to perform liquid penetrant examinations on it during the repair process. This penetration was repaired using the temper bead process, which was approved by the NRC during this refueling outage (2R17). During this repair, the licensee encountered some difficulties with the machining process. Because of the nozzle being slightly distorted, the boring of the old sleeve resulted in a small sliver of material remaining that had to be removed. In addition, the as-left dimension of the new sleeve was slightly larger than originally planned. The inspectors verified that work was performed in accordance with site procedures and work instructions and that the licensee entered the issues into their CAP as CRs ANO-2-2005-0507, -0720, -0842, -0846, and -1309.

(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 find any leaks on the existing installed MNSAs during the shutdown walkdown, but they did find some streaking and residue on two of the six, CR ANO-2-2005-0489. For the normal operating temperature and pressure walkdown, the licensee found no leaks on the MNSAs. The inspectors independently performed a hot walkdown and also observed no leaks. The licensee did, however, have problems with the installation process for the nine new MNSAs that were installed, CRs ANO-2-2005-0627 and -1269. These issues included: (1) alignment of the mounting holes around the nozzle, (2) the 'Go' gauge and the 'No-Go' gauge indicating the mounting moles were not within tolerance, and (3) initially overtorquing 14 of the 15 MNSAs.

Note: Throughout the documentation of the pressurizer examination, a Level II VT-2 inspection refers to the fact that an individual qualified to the level of a Level II VT-2 inspector performed the inspection, not that an actual Level II VT-2 inspection occurred.

b. Findings

No findings of significance were identified.

.3 Licensee Strike Contingency Plans (Inspection Procedure 92709)

The inspectors evaluated the adequacy of the licensee's strike contingency plan and verified plan implementation prior to the May 19, 2005, adoption of the negotiated labor agreement by the security force.

b. Findings

No findings of significance were identified.

.4 Transportation of Reactor Control Rod Drives in Type A Packages (TI 2515/161)

a. Inspection Scope

This area was inspected to verify that the licensee's radioactive material transportation program complies with specific requirements of 10 CFR Parts 20 and 71 and Department of Transportation regulations contained in 49 CFR Part 173. The inspector interviewed licensee personnel and determined the licensee had undergone refueling/defueling activities between January 1, 2002, and present, but it had not shipped irradiated control rod drives in Department of Transportation Specification 7A Type A packages.

b. Findings

No findings of significance were identified.

.5 Operational Readiness of Offsite Power (TI 2515/163)

The inspectors collected data pursuant to TI 2515/163, "Operational Readiness of Offsite Power." The inspectors reviewed the licensee's procedures related to General Design Criteria 17, "Electric Power Systems;" 10 CFR 50.63, "Loss of All Alternating Current Power;" 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants;" and the Technical Specifications for the offsite power system. The data was provided to the Office of Nuclear Reactor Regulation for further review. Documents reviewed for this TI are listed in the attachment.

.6 Problem Identification and Resolution

(Closed) Unresolved Item 05000313;368/2005009-04, "Untimely Corrective Measures to Address Repetitive 4160 VAC Cable Failures"

Introduction. The team documented a Green self-revealing, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, because the licensee failed to correct a 4160 Vac cable failure mechanism (a significant condition adverse to quality). In addition, the licensee failed to properly address industry operating experience on the same topic. The cables were submerged in water but they were not designed for submergence. Consequently, several 4160 Vac service water pump and fire pump cables failed in service between 1993 and 2003, when all the cables were replaced.

Description. On February 2, 2003, Service Water Pump 2P-4C tripped because the feeder cable shorted to ground. The feeder cable was located in an underground vault and was submerged in water. The cables were not designed for submergence. Prior to the trip, the licensee had experienced four similar failures, including Unit 2 Service Water Pump A in 1993, Unit 1 Service Water Pump C in 1995, Unit 1 Service Water Pump C in 1999 (in 1995 the licensee only replaced half of the cable and in 1999 the remaining half failed), and Fire Water Pump P-6A in 2001.

The inspectors also noted that the NRC had previously issued generic correspondence to address this problem - Information Notice 2002-12, "Submerged Safety-Related Electrical Cables," dated April 21, 2002. The licensee did not adequately address the information notice and did not take actions to replace all the cables that had not yet failed. Subsequently, the fifth 4160 Vac cable failed in service on February 2, 2003.

As a result of the 2003 Unit 2 Service Water Pump C failure, the licensee performed a root cause determination, CR-ANO-C-2003-0067, and concluded that black ethylene propylene rubber was susceptible to a phenomena called water treeing. A Sandia National Laboratory report published in 1996, "Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations," documented the same phenomena. In addition, the licensee identified the following contributing causes: (1) corrective action for previously identified problems/events were not adequate to prevent recurrence; management failed to take meaningful corrective action for past events; and (2) previous industry operating experience was not effectively used to prevent problems. The licensee subsequently replaced the remaining 4160 Vac safety-related black ethylene propylene cables with cables that were designed for the environment. The licensee determined that no other groups of safety-related cables were impacted.

Analysis. The failure to take measures to prevent recurrence for a significant condition adverse to quality was a performance deficiency. This finding was more than minor because it affected the initiating events and mitigating system cornerstone objectives of limiting the likelihood of initiating events and ensuring the availability of systems that mitigate plant accidents. Based on the results of a significance determination process using Manual Chapter 0609, Appendix A1, Phase 1 work sheet, this finding screened to a Phase 2 by satisfying the criteria of affecting two cornerstones. Using Appendix A, "Technical Basis for at Power Significance Determination Process," of Manual Chapter 0609, and the Phase 2 work sheets from the "Risk-Informed Inspection Notebook for Arkansas Nuclear One - Unit 2," the finding screened as potentially greater than Green.

A Region IV senior reactor analyst performed a Phase 3 significance determination. Based, in part, on the following assumptions, the senior reactor analyst determined that the issue was of very low risk significance (Green):

- The increased risk of failure was assumed for all unrepaired service water pump cables on Units 1 and 2.

- The exposure period was 1 year prior to the failure of Service Water Pump 2P-4C on February 2, 2003, plus 6 days for restoring the pump to an operable condition.

- No other risk significant plant equipment was affected by the finding during the exposure period.

The entire significance determination can be found in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS) - search on Accession Number ML051650079. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

This issue had crosscutting aspects associated with problem identification and resolution in that the licensee failed to adequately evaluate the condition.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, requires, for a significant condition adverse to quality, that the licensee determine the cause of the condition and take corrective measures to preclude repetition. Contrary to the above, in 1993 when the first safety-related 4160 Vac service water system cable failed (a significant condition adverse to quality), the licensee did not determine the cause for the failure and did not take effective measures to preclude repetition. Consequently, until recently, when all the affected cables were replaced, 4160 Vac cables continued to have failures. Because the violation was of very low safety significance and was entered into the licensee's corrective action program (CR ANO-C-2003-0067), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000368/2005009-04).

4OA6 Meetings, Including Exit

The inspectors presented the results of the inservice inspection effort to Mr. Cliff Eubanks, General Manager, Plant Operations, and other members of your staff on March 25, 2005. Licensee management acknowledged the inspection results. The inspectors noted that, while proprietary information was reviewed, none would be included in this report.

On May 10, 2005, the inspector discussed the radioactive material transportation inspection with Mr. D. Moore, Radiation Protection Manager. The inspector verified that no proprietary information was provided during the inspection.

On June 23, 2005, the inspector presented the ALARA inspection results to Mr. C. Eubanks, General Manager, Plant Operations, and other members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

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 June 30, 2005. 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 violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

10 CFR Part 50, Appendix B, Criterion V, states, in part, that activities affecting quality shall be accomplished in accordance with prescribed instructions. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. On March 15, 2005, the licensee identified, while performing Refueling Outage 2R17 Alloy 600 Pressurizer VT-2 bare metal visual inspections for boric acid leakage (in accordance with requirements of Procedure 2311.009-K, "Unit 2 Pressurizer A-600 Bottom Visual Inspection"), that one pressurizer heater sleeve (F2), and two level instrument nozzles associated with Valves 2RC-4627C and 2RC-4627G, did not receive a VT-2 inspection as required prior to vessel bottom cleaning and decontamination. The licensee determined that Heater Sleeve F2 was not listed on the procedure inspection sheet, but the two pressurizer instrument nozzles were on the inspection sheet. The licensee determined that instrument nozzles associated with Valves 2RC-4627C and 2RC-4627G were missed because the VT-2 inspectors thought that these penetrations

were outside of the inspection scope. This condition is described in the licensee's CAP in CR ANO-2-2005-0566. This finding is of very low safety significance because, even though these penetrations were not on the inspection sheet, they did receive an inspection during the licensee's Alloy 600 walkdown prior to cleaning and decontamination of the bottom of the pressurizer.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Bennett, Project Manager, Licensing
B. Berryman, Manager, Unit 1 Operations
M. Briley, Technical Specialist IV, Non-Destructive Examination
T. Brown, Supervisor, Radiation Protection
J. Browning, Acting Manager, Planning and Scheduling
S. Cotton, Manager, Training
J. Eichenberger, Manager, Corrective Actions and Assessments
C. Eubanks, General Manager, Plant Operations
N. Finney, Technical Specialist IV, Non-Destructive Examination
J. Forbes, Vice President, Arkansas Nuclear One
A. Hawkins, Licensing Specialist
J. Hoffpauir, Manager, Maintenance
R. Holeyfield, Manager, Emergency Planning
D. James, Acting Director, Nuclear Safety Assurance
W. James, Manager, Alloy 600 Group
J. Kowalewski, Director, Engineering
D. Lomax, Manager, Dry Fuels
D. Meatheany, Senior Technical Specialist IV
J. Miller, Manager, Systems Engineering
D. Moore, Manager, Radiation Protection
K. Nichols, Manager, Design Engineering
G. Parks, Supervisor, Non-Destructive Examination
R. Partridge, Manager, Technical Support
M. Paterak, Technical Specialist IV
S. Pyle, Licensing Specialist
C. Reasoner, Manager, Engineering Programs and Components
T. Roliniak, Specialist, Radiation Protection
J. Rudder, Acting Manager, Unit 2 Operations
R. Scheide, Licensing Specialist
T. Smith, Specialist, Radiation Protection
C. Tyrone, Manager, Quality Assurance
B. Williams, Director, Reactor Vessel Head/Steam Generator Replacement Project

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000368/2005003-01	AV	Inadequate procedure leads to RCP Seal Damage (Section 1R20)

Opened and Closed

05000368/2005003-02	NCV	Failure to Follow a Service Water Surveillance Procedure (Section 1R22)
05000368/2005003-03	NCV	Failure to Evaluate Radiological Hazards (Section 2OS2)
05000313/2005003-04	NCV	Inappropriate Mode Change Without All Required Equipment Being Operable (Section 4OA3)
05000368/2005003-05	NCV	Untimely corrective actions to address repetitive 4160 Vac Cable Failures (Section 4OA5)

Closed

05000368/2004-003-00	LER	Entry into an Operational Mode Prohibited by Technical Specification due to Inoperable Pressurizer Proportional Heaters (Section 4OA3)
05000368/2005-001-00	LER	RCS Pressure Boundary Leakage due to Primary Water Stress Corrosion Cracking of Pressurizer Heater Sleeves (Section 4OA3)
05000368/2005009-04	URI	Untimely corrective actions to address repetitive 4160 Vac Cable Failures (Section 4OA5)

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 1R04: Equipment Alignment

Plant Drawings

NUMBER	TITLE	REVISION
M-31	Piping & Instrument Diagram Makeup and Purification, Sheet 1	107
M-2204	Piping & Instrument Diagram Condensate and Feedwater, Sheet 2	79
M-2206	Piping & Instrument Diagram Steam Generator Secondary System, Sheet 1	142

M-2232	Piping & Instrument Diagram Safety Injection System, Sheet 1	115
M-2236	Piping & Instrument Diagram Containment Spray System, Sheet 1	91

Plant Procedures

NUMBER	TITLE	REVISION
1104.002	Makeup and Purification System Operation	57
2104.004	Shutdown Cooling System	30
2106.006	Emergency Feedwater System Operations	56

Section 1R05: Fire Protection

Engineering Calculation

85-E-0053-34, Revision 12

Plant Documents

Arkansas Nuclear One Fire Hazards Analysis Report, Revision 9
Unit 1 Prefire Plans 1A-372-100-N, Revision 2
Unit 2 Prefire Plans 2B-354-2073-DD, Revision 4

Plant Drawings

FP-103 Fire Zones Intermediate Floor Plant Elev. 368' and 372', Sheet 1, Revision 24
FP-105 Fire Zone Plan Below Grade Elev. 335', Sheet 1, Revision 18
FP-2102 Fire Zone Operating Floor Plan Elev. 386', Sheet 1, Revision 32
FP-2104 Fire Zone Ground Floor Plan Elev. 354', Sheet 1, Revision 29

Section 1R06: Flood Protection Measures

Engineering Calculation

83-E-0033-15, Revision 0

Engineering Report

92-R-0024-01, Revision 0

Section 1R07: Heat Sink Performance

Engineering Report

ER-ANO-2005-0168-00, Revision 0

Plant Procedure

NUMBER	TITLE	REVISION
2311.001	Shutdown Cooling Heat Exchanger Performance Test	5

Section 1R08: 1R08 Inservice Inspection Activities

Section 2.01: Performance of Nondestructive Examination Activities Other than Steam Generator Tube Inspections, PWR Vessel Upper Head Penetrations Inspections, Boric Acid Control

SYSTEM/COMPONENT ID	WELD/PENETRATION	EXAM METHOD
Reactor Coolant	2-16-004	Ultrasonic
Reactor Coolant	2- 27-065*	Ultrasonic
Reactor Coolant	2- 27-066*	Ultrasonic
Reactor Coolant	2- 43-023*	Ultrasonic
Safety Injection	2- 22-004	Ultrasonic
Safety Injection	2- 22-005	Ultrasonic
Safety Injection	2-25-037	Ultrasonic

*Observed examination

Inservice Inspection NDE

1 Level III Ultrasonic Testing, Washington Group
3 Level II Ultrasonic Testing, Washington Group
2 Level II Ultrasonic Testing, Structural Integrity Associates, Inc.
1 Level III Ultrasonic Testing, Structural Integrity Associates, Inc.
1 Level III Ultrasonic Testing, Entergy

Section 2.04: Steam Generator Tube Inspection Activities

PROCEDURE	TITLE	REVISION/CHANGE
2305.055	SG Leak Test	001-010
5120.519	SG In-Situ Testing	001-02-0
STD-FP-1997-8053	Field Procedure for In-Situ Testing of 11/16" SG Tubes	4

SYSTEM/COMPONENT ID	WELD/PENETRATION	EXAM METHOD
Steam Generator A	Tube 30-90	Eddy-current
Steam Generator A	Tube 70-169	Eddy-current
Steam Generator A	Tube 71-170	Eddy-current
Steam Generator A	Tube 72-169	Eddy-current
Steam Generator B	Tube 102-25	Eddy-current
Steam Generator B	Tube 109-92	Eddy-current

Inspector Certifications Reviewed

Steam Generator ET Inspectors

3 Level II
2 Level IIA Quality Data Analyst
5 Level III Quality Data Analyst

The inspectors also verified that the technician performing bobbin data collection for Steam Generator A was on the qualified list.

Miscellaneous

EPRI Document, Steam Generator Examination Guidelines, Revision 6
Steam Generator Integrity Assessment Guidelines, Revision 1
FR-ANO-2005-0082-00, ANO Unit 2, Steam Generator Degradation Assessment

Section 2.05: Identification and Resolution of Problems

CRs

ANO-2-2005-0344	ANO-2-2005-0566	ANO-2-2005-0924
ANO-2-2005-0419	ANO-2-2005-0847	ANO-2-2005-0941
ANO-2-2005-0489	ANO-2-2005-0876	ANO-2-2005-0953
ANO-2-2005-0506	ANO-2-2005-0914	ANO-2-2005-0961
ANO-2-2005-0507	ANO-2-2005-0915	ANO-2-2005-1118

Section 1R11: Licensed Operator Requalification Program

Training Scenario

ASPGLOR050401, “Functional Recovery,” Revision 0

Section 1R12: Maintenance Effectiveness

CRs

ANO-2-2003-0674	ANO-2-2004-0292
ANO-2-2003-1306	ANO-2-2005-1588
ANO-2-2003-1598	ANO-2-2005-1765

Miscellaneous

Maintenance Rule Database, Unit 2, Containment Spray System
Maintenance Rule Database, Unit 2, Reactor Building System

Work Orders

00025585 01	50245540 01
50245531 01	51000316 01

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

CRs

ANO-C-2005-0857

Section 1R15: Operability Evaluations

CRs

ANO-1-2005-0653	ANO-2-2005-1053	ANO-2-2005-1576
ANO-2-2004-1516	ANO-2-2005-1099	ANO-2-2005-1678
ANO-2-2005-0406	ANO-2-2005-1116	
ANO-2-2005-1048	ANO-2-2005-1505	

Plant Procedures

2202.006, “Loss of Feedwater,” Revision 6

Work Orders

00066102 01

Section 1R16: Operator Workarounds

Miscellaneous

Ops Impact Concerns Database

Section 1R19: Postmaintenance Testing

Work Orders

00025585 01
00038792 01
50965936 01
50966845 01
50966860 01
50976879 01
51000316 01

Section 1R22: Surveillance Testing

CRs

ANO-C-2004-1707
ANO-C-2005-0338
ANO-C-2005-0341
ANO-C-2005-0342
ANO-C-2005-0362
ANO-C-2005-0368
ANO-C-2005-0391

Engineering Report

91-R-2013-01, Revision 12

Plant Procedures

NUMBER	TITLE	REVISION
2311.002	Unit 2 Service Water System Flow Test	14
5120.425	In-Place Testing of the Unit 1 Control Room Filtration System	6

Miscellaneous

2001 ASHRAE Handbook of Fundamentals, Section 14
ACGH Industrial Ventilation, 17th Edition, Section 9
OSHA Technical Manual, Section III: Chapter 3

Work Orders

00034802 01
50966664 01

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents

ANO-2-2005-00553	ANO-2-2005-01307
ANO-2-2005-00730	ANO-2-2005-01487
ANO-2-2005-00847	ANO-C -2005-00626
ANO-2-2005-01040	

Audits and Self-Assessments

QA-14-2005-ANO-1, Quality Assurance Report of Radiation Protection

Radiation Work Permit Packages

NUMBER	TITLE
2005-2413	Reactor Building Coordinator Activities
2005-2420	Scaffolding and Insulation
2005-2442	Steam Generator Inspection and Repair
2005-2443	Steam Generator Secondary Side Work
2005-2481	2P-32C Motor Replacement
2005-2907	Pressurizer Repairs

Procedures

NUMBER	TITLE	REVISION
ENS-RP-105	Radiation Work Permits	6
RP-105	Radiation Work Permits	5
RP-110	ALARA Program	2

ALARA Committee Minutes

March 9, 2005	March 18, 2005	March 31, 2005
March 10, 2005	March 21, 2005	April 2, 2005
March 14, 2005	March 24, 2005	April 4, 2005
March 15, 2005	March 28, 2005	April 5, 2005
March 17, 2005	March 29, 2005	April 6, 2005

Section 4OA2: Identification and Resolution of Problems

CRs

ANO-C-2002-0500	ANO-2-2005-0084	ANO-C-2005-0324
ANO-C-2003-1025	ANO-2-2005-0551	ANO-C-2005-0422
ANO-1-2004-0531	ANO-2-2005-0574	ANO-C-2005-0427
ANO-1-2004-0911	ANO-2-2005-0730	ANO-C-2005-0869
ANO-2-2004-1487	ANO-2-2005-0759	ANO-C-2005-0896
ANO-1-2005-0079	ANO-2-2005-1007	ANO-C-2005-0975
ANO-1-2005-0359	ANO-2-2005-1387	ANO-C-2005-0991
ANO-1-2005-0512	ANO-2-2005-1397	ANO-C-2005-1394
ANO-1-2005-0888	ANO-C-2005-0194	

Section 4OA3: Event Followup

CRs

ANO-C-2004-1791	ANO-2-2002-1354	ANO-2-2004-1727
ANO-C-2004-2116	ANO-2-2004-0223	ANO-2-2004-1728
ANO-C-2004-2208	ANO-2-2004-1709	ANO-2-2004-1735
ANO-2-1994-0255	ANO-2-2004-1713	ANO-2-2004-1793
ANO-2-1998-0117	ANO-2-2004-1716	ANO-2-2004-1867
ANO-2-1998-0141	ANO-2-2004-1726	ANO-2-2004-1961

Miscellaneous

LIC-04-032, Technical Specification Actions for ANO-2 Pressurizer Proportional Heaters
LIC-04-045, Technical Specification Requirements for ANO-2 Pressurizer Proportional Heaters
LIC-04-046, Application of ANO-2 EDG TSs with Regard to the Pzr Proportional Heaters
LIC-04-047, TS AOT Reset During Failure of Redundant Components
Maintenance Rule Database, Unit 2 RCS
Maintenance Rule Database, Unit 2 480 V Load Centers & Motor Control Centers

Plant Procedures

NUMBER	TITLE	REVISION
1015.036	Containment Building Closeout	11
1015.047	Condition Reporting Operability and Immediate Reportability Determinations,	1
2102.001	Plant Pre-Heatup and Precritical Checklist,	51
2102.002	Plant Heatup	52

2107.001	Electrical System Operations	49
2305.016	Remote Feature Periodic Testing	16
2307.009	Pressurizer Proportional Heater Checkout	6

Work Orders

00052582 01	50254407
00052582 04	50254408
00052582 05	50261965
00052582 06	50278646
50248377	

Section 4OA5: Other Activities

.1 Reactor Vessel Head and Head Inspections (TI 2515/150)

PROCEDURE	TITLE	REVISION/CHANGE
2311.009	ANO Unit 1 and Unit 2 Alloy 600 Inspection	007-00
2311.009G	Unit 2 RPV Closure Head A-600 Visual Inspection	008-00
2311.009L	Units 1 & 2 Pressure Retaining Component Above RPV Head	007-00

SYSTEM/COMPONENT ID	WELD/PENETRATION	EXAM METHOD
Reactor Vessel Head	Control Element Drive Mechanism 2	Ultrasonic/Eddy-current
Reactor Vessel Head	Control Element Drive Mechanism 6	Ultrasonic/Eddy-current
Reactor Vessel Head	Control Element Drive Mechanism 12	Ultrasonic/Eddy-current
Reactor Vessel Head	Control Element Drive Mechanism 15	Ultrasonic/Eddy-current
Reactor Vessel Head	Control Element Drive Mechanism 19	Ultrasonic/Eddy-current
Reactor Vessel Head	Control Element Drive Mechanism 33	Ultrasonic/Eddy-current
Reactor Vessel Head	Control Element Drive Mechanism 44	Ultrasonic/Eddy-current
Reactor Vessel Head	Control Element Drive Mechanism 62	Ultrasonic/Eddy-current
Reactor Vessel Head	Control Element Drive Mechanism 80	Ultrasonic/Eddy-current

Inspector Certifications Reviewed

Eight Level II Ultrasonic Testing Inspectors
Four Level III Ultrasonic Testing Inspectors

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

SYSTEM/COMPONENT ID	WELD/PENETRATION	EXAM METHOD
Pressurizer	New Pressurizer Heater Sleeve 1	Radiography
Pressurizer	New Pressurizer Heater Sleeve 2	Radiography
Pressurizer	New Pressurizer Heater Sleeve 3	Radiography
Pressurizer	Penetrations G-3*	Eddy-current
Pressurizer	Penetrations X-3*	Eddy-current
Pressurizer	Penetrations V-1*	Eddy-current
Pressurizer	Penetrations J-2*	Eddy-current
Pressurizer	Penetrations P-1	Eddy-current
Pressurizer	Penetrations P-2	Eddy-current
Pressurizer	Penetrations U-3	Eddy-current
Pressurizer	Penetrations H-4	Eddy-current
Pressurizer	Penetrations C-4	Eddy-current

*Observed examination

CRs

ANO-2-2005-0419	ANO-2-2005-0627	ANO-2-2005-1180
ANO-2-2005-0451	ANO-2-2005-0639	ANO-2-2005-1269
ANO-2-2005-0489	ANO-2-2005-0720	ANO-2-2005-1274
ANO-2-2005-0569	ANO-2-2005-0842	ANO-2-2005-1309
ANO-2-2005-0607	ANO-2-2005-0846	

Miscellaneous Documents

NUMBER	TITLE	REVISION /DATE

CEP-NDE-0112	Program Section for Certification of Visual Testing (VT) Personnel	1
CEP-NDE-0641	ANO Unit 2, 2T1 Nozzle X-1 Liquid Penetrant Examination	March 13, 2005
CNRO-2002-00012	Entergy Operations, Inc., Use of Mechanical Nozzle Seal Assemblies, Relief Request	March 15, 2002
CNRO-2005-00016	Request for Alternative ANO2-R&R-003 Proposed Alternative to ASME Requirements for Weld Repairs	March 16, 2005
CNRO-2005-00017	Supplemental Information for Request for Alternative ANO2-R&R-003 Proposed Alternative to ASME Requirements for Weld Repairs	March 18, 2005
CNRO-2005-00019	Response to Request for Additional Information Concerning ANO2-R&R-003 Proposed Alternative to ASME Requirements for Weld Repairs	March 24, 2005
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	May 28, 2004
0CAN030502	Inservice Inspection (ISI) Summary Report Unit 2	March 25, 2005
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
Welding Services Inc.	Liquid Penetrant Inspection Report X-1, Reference 101707-2	March 25, 2005

Plant Procedures

NUMBER	TITLE	REVISION
2311.009	ANO Unit 1 and Unit 2 Alloy 600 Inspections	7
2311.009J	Unit 2 Pressurizer A-600 Upper Piping Visual Inspection	7
2311.009K	Unit 2 Pressurizer A-600 Bottom Visual Inspection	7
5120.243	Unit 2 - Post Outage Pressure Test	10
5120.243	Unit 2 - Post Outage Pressure Test	11

.5 TI 2515/163, "Operational Readiness of Offsite Power"

Corporate Procedures

NUMBER	TITLE	REVISION
EN-LI-102	Corrective Action Process	1
EN-LI-108	Event Notification and Reporting	0
EN-PL-145	Notifications of Off-Normal Situations,	7

Plant Procedures

NUMBER	TITLE	REVISION
1104.036	Emergency Diesel Generator Operation	42
1107.001	Electrical System Operations,	60
1015.047	Condition Reporting Operability and Immediate Reportability Determinations	1
1202.007	Degraded Power	6
1202.008	Blackout	7
1203.012B	Annunciator K02 Corrective Action	26
1203.025	Natural Emergencies	19
1203.037	Abnormal ES Bus Voltage	4
2104.036	Emergency Diesel Generator Operations	49
2107.001	Electrical System Operations	49
2202.007	Loss of Offsite Power	6
2202.008	Station Blackout	6
2203.012A	Annunciator 2K01 Corrective Action	24

Miscellaneous

Plant Procedures

NUMBER	TITLE	REVISION/ DATE

COPD-024	Risk Assessment Guidelines	15
OPS-145	Shift Turnover Checklist Extended EDG Outage,	3
OPS-146	Extended EDG Outage Coordinator Checklist	2
OSP-013	Coordination	June 15, 1994
OSP-014	Power Availability Requirements for Arkansas Nuclear One	August 8, 2001
	Switchyard and Transmission Interface Agreement	June 1, 1999

LIST OF ACRONYMS

ACW	auxiliary cooling water
ALARA	as low as reasonably achievable
ANO	Arkansas Nuclear One
ASME	American Society of Mechanical Engineers
AV	apparent violation
CAP	corrective action program
CFR	*Code of Federal Regulations*
CR	condition report
EPRI	Electric Power Research Institute
LER	licensee event report
MNSA	mechanical nozzle seal assembly
NCV	noncited violation
PI&R	problem identification and resolution
RCP	reactor coolant pump
RCS	reactor coolant system
RTP	rated thermal power
SSC	structure, system, and component
TI	temporary instruction