UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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July 23, 2004

Southern Nuclear Operating Company, Inc.
ATTN: J. Gasser, Jr., Vice President
Vogtle Electric Generating Plant
P. O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT - NRC INTEGRATED INSPECTION REPORT 05000424/2004004 AND 05000425/2004004

Dear Mr. Gasser:

On June 26, 2004, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Vogtle Electric Generating Plant (VEGP), Units 1 and 2. The enclosed integrated inspection report documents the inspection results, which were discussed on July 6, 2004, with Mr. William Kitchens and other members of your staff.

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

This report documents an NRC-identified finding concerning a failure to perform an adequate containment closeout inspection following the Unit 2 refueling outage. This finding has potential safety significance greater than very low significance and will remain open pending additional review to determine the risk significance. This finding does not present an immediate safety concern because the debris identified by the inspectors was removed from containment. In addition, this report documents three NRC-identified findings and two self-revealing findings of very low safety significance (Green). Two of these findings were determined to involve violations of NRC requirements. However, because of their very low safety significance, and because they had been entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. Additionally, two licensee-identified violations, which were determined to be of very low safety significance, are listed in this report. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Vogtle.

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

Sincerely,

/RA/

Brian R. Bonser, Chief
Reactor Projects Branch 2
Division of Reactor Projects

Docket Nos.: 50-424, 50-425
License Nos.: NPF-68, NPF-81

Enclosure: Inspection Report 05000424/2004004
and 05000425/2004004
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-424, 50-425
License Nos.:	NPF-68, NPF-81
Report Nos.:	05000424/2004004 and 05000425/2004004
Licensee:	Southern Nuclear Operating Company, Inc. (SNC)
Facility:	Vogtle Electric Generating Plant
Location:	7821 River Road Waynesboro, GA 30830
Dates:	March 28, 2004 - June 26, 2004
Inspectors:	J. Zeiler, Senior Resident Inspector T. Morrissey, Resident Inspector (RI) Katherine Green-Bates, RI, Turkey Point (Section 1R08) Kirk Maxey, Reactor Inspector (Section 1R02) Melanie Maymi, Reactor Inspector (Section 1R07) Adam Nielsen, Health Physicist (Sections 2OS1 and 4OA1) Eldan Testa, Senior Health Physicist (Sections 2OS2 and 2PS2)
Approved by:	Brian R. Bonser, Chief Reactor Projects Branch 2 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000424/2004-004, 05000425/2004-004; 03/28/2004 - 06/26/2004; Vogtle Electric Generating Plant, Units 1 and 2; Operator Workarounds, Refueling and Outage Activities, Access Control to Radiologically Significant Areas, Event Followup, and Other.

The report covered a three-month period of inspection by resident inspectors and announced inspections by two reactor inspectors and regional health physics inspectors. One apparent violation with the safety significance to be determined, two Green non-cited violations (NCVs), and three Green findings, were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing finding for inadequate feedpump turbine control valve work instructions was identified which resulted in the loss of feedwater flow control and a subsequent manual reactor trip.

  This finding is greater than minor because it affected the procedure quality attribute of the Initiating Events cornerstone and affected the cornerstone objective by resulting in a reactor trip. The finding was determined to be of very low safety significance because it did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiator, it did not contribute to a loss of mitigation equipment functions, and it did not increase the likelihood of a fire or internal/external flood (Section 4OA3.1).

- Green. A self-revealing NCV of Technical Specification (TS) 5.4.1.a was identified for failure to follow the Unit 2 operating procedure to disable the Auxiliary Feedwater (AFW) actuation signal prior to breaking condenser vacuum.

  This finding is greater than minor because it affected the human performance attribute of the Initiating Events cornerstone and affected the cornerstone objective, in that, it caused an unplanned engineered safety features actuation. The finding is of very low safety significance because it did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiator, did not contribute to a reactor trip with the loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood. The direct cause of this finding involved the cross-cutting area of Human Performance (Section 4OA3.3).

Cornerstone: Mitigating Systems

- Green. An NCV of TS 5.4.1.a was identified by the inspectors for failure to maintain adequate Unit 1 and Unit 2 Nuclear Service Cooling Water (NSCW) system operating procedures.

  This finding is greater than minor because it affected the Mitigating Systems cornerstone attribute of configuration control and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences by rendering the automatic NSCW heat removal function inoperable. This finding is of very low safety significance because the duration did not exceed the 72 hour allowed outage time for one inoperable NSCW train and it did not represent an actual loss of service water safety function (Section 1R16).

- TBD. An apparent violation of TS 5.4.1.a was identified by the inspectors having potential safety significance greater than very low significance for failure to perform an adequate Unit 2 containment closeout inspection in accordance with approved procedures prior to entering Mode 4, Hot Shutdown.

  This finding is greater than minor because it affected the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective, in that, the failure to perform an adequate containment closeout inspection resulted in debris left in containment that could have resulted in inadequate net positive suction head for both trains of the Residual Heat Removal system in the recirculation phase during a design basis loss of coolant accident. This finding will remain open pending completion of a final significance determination (Section 1R20).

- Green. A finding was identified by the inspectors for failure to perform a timely and appropriate operability assessment to address a common cause equipment degradation identified with the AFW discharge control valves.

  The failure to perform a timely and appropriate operability evaluation for the common cause valve degradation is greater than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance because, although the motor driven AFW discharge control valves with the missing cotter pins were considered degraded, the pilot plug assembly retaining nuts for all the valves were still held in place by the disrupted metal on the valve stem threads, therefore the immediate functional capability of the valves was not actually impacted. The direct cause of this finding involved the cross-cutting area of Problem Identification and Resolution (Section 4OA5.1).

Cornerstone: Occupational Radiation Safety

- Green. A finding was identified by the inspectors for inadequate control of keys to Very High Radiation Areas (VHRAs).

  This finding is greater than minor because if left uncorrected the issue could become a more significant safety concern, in that, someone could gain unauthorized access to a VHRA. The finding is of very low safety significance because there was no overexposure, there was no evidence of unauthorized access into a VHRA, and the licensee's ability to assess dose was not compromised (Section 2OS1).

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 began the period in Mode 3, Hot Standby, following a manual reactor trip on March 27. The unit was restarted on March 29 and returned to 100 percent Rated Thermal Power (RTP) on April 1. The unit operated at essentially 100 percent RTP for the remainder of the inspection period.

Unit 2 operated at essentially 100 percent RTP until April 18, when the unit was shutdown for a planned refueling outage. The unit was restarted on May 16, and attained 100 percent RTP on May 31. The unit operated at essentially 100 percent RTP for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed the licensee's safety evaluation for Design Change Package (DCP) No. 97-V2N0062, which replaced the existing Unit 2 Train A Engineered Safety Features (ESF) sequencer logic circuitry with a digital control system. The purpose of this review was to determine whether the licensee had appropriately concluded that the change could be accomplished without obtaining a license amendment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial Walkdowns. The inspectors performed partial walkdowns of the following three systems to verify correct system alignment while redundant or backup equipment was inoperable. The inspectors checked for correct valve and electrical power alignments by comparing positions of valves, switches, and breakers to the procedures and drawings listed in the Attachment. Additionally, the inspectors reviewed the condition report (CR) database to verify that equipment alignment problems were being identified and appropriately resolved.

- 2A Containment Spray (CS) system while 2B CS system was out of service
- 1A Motor Driven Auxiliary Feedwater (MDAFW) system while 1B MDAFW was out of service
- 2A Safety Injection (SI) system while 2B SI system was out of service

Complete Walkdowns. The inspectors conducted a detailed review of the accessible portions of the Unit 2 SI system. The inspectors used procedures 11105-2, Safety Injection System Alignment, 13105-2, Safety Injection System, and drawings 2X4DB119, 2X4DB120, and 2X4DB121 to verify adequate system alignment, electrical power availability, labeling, hangers and support installation, and support systems status. The inspectors also reviewed system health reports, maintenance rule monthly reports, CRs, and outstanding maintenance work orders (MWOs) to verify that alignment and equipment discrepancies were being identified and appropriately resolved.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

Fire Area Tours. The inspectors toured the following nine plant areas to verify the licensee was controlling combustible materials and ignition sources as required by procedures 92015-C, Use, Control, and Storage of Flammable/Combustible Materials, and 92020-C, Control of Ignition Sources. The inspectors assessed the observable condition of fire detection, suppression, and protection systems and reviewed the licensee's fire protection Limiting Condition for Operation log and CR database to verify that corrective actions for degraded equipment were being identified and appropriately resolved. The inspectors also reviewed the licensee's fire protection program to verify the requirements of Updated Final Safety Analysis Report (UFSAR) Section 9.5.1, Fire Protection Program, and Appendix 9A, Fire Hazards Analysis, were met. Documents reviewed are listed in the Attachment.

- Unit 1, Train B cable spreading room
- 2A CS pump room
- 1A MDAFW pump and Unit 1 Turbine Driven Auxiliary Feedwater (TDAFW) pump rooms
- 2A SI pump room
- Unit 2 Normal Charging pump and 2A High Head Safety Injection pump rooms
- 2A and 2B 4.16 kilovolt (kV) emergency switchgear rooms
- Unit 1 and Unit 2, Control Building Level B
- 1B Nuclear Service Cooling Water (NSCW) building
- Unit 1, Train B Component Cooling Water (CCW) pump room

Fire Drill Observation. On May 26, the inspectors observed an unannounced fire drill conducted in the Unit 1 Control Building, Electrical Equipment Room B60, involving a ventilation fan motor fire (Drill Number 2004-Q2-1). The inspectors assessed the adequacy of the fire drill and fire brigade response using licensee procedures 92000-C, Fire Protection Program; 92005-C, Fire Response Procedure; 92030-C, Fire Drill Program; 92772-1, Zone 72 Control Building Level B Fire Fighting Preplan; 3302-1, Control Building ESF Ventilation System; and, 17103A-C, Annunciator Response Procedures for Fire Alarm Computer. The inspectors evaluated fire brigade performance to verify that they responded to the fire in a timely manner, donned proper protective clothing, used self-contained breathing apparatus, and had equipment necessary to control and extinguish the fire. The inspectors assessed the adequacy of the fire brigade's fire fighting strategy including entry into the fire area, communications, search and rescue, and fire equipment usage.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

Biennial Review: The inspectors reviewed inspection records, performance test results, MWOs, and other documentation to ensure that heat exchanger (HX) deficiencies that could mask or degrade performance were identified and corrected. The test procedures and records were also reviewed to verify that they were consistent with Generic Letter 89-13 licensee commitments and Electric Power Research Institute (EPRI) Heat Exchanger Performance Monitoring Guidelines. The following three risk significant heat exchangers were reviewed.

- Residual Heat Removal (RHR) pump motor coolers
- Essential Chilled Water (ECW) system chiller condensers
- Auxiliary Component Cooling Water (ACCW) heat exchangers.

The inspectors reviewed ACCW HX fouling factor and required flow test completed procedures, HX inspection procedures and MWOs, and Eddy Current Test (ECT) results including tube plugging margins. Additionally, ECW chiller inspection MWOs, ECT results including tube plugging margins, and chiller flow control valve instrument calibration MWOs were also reviewed. The inspectors also reviewed RHR pump motor cooler required flow and steady state temperature test completed procedures, inspection work orders, and motor cooler flow trending graphs. These documents were reviewed to verify that test results were consistent with design acceptance criteria, testing methodology and assumptions were adequate, inspection methods and performance of the HXs under the current maintenance frequency was adequate, and to verify minimum flow requirements and HX design basis were being maintained.

The inspectors also reviewed general health of the NSCW system via review of design basis documents, system health reports, chemistry control procedures, material replacement equivalency determinations and a Request for Engineering Assistance, and discussions with the NSCW system engineer. CRs were reviewed for potential common cause problems and problems which could affect system performance to confirm that the licensee was entering problems into the corrective action program and initiating appropriate corrective actions. These CRs included actions regarding ECW chiller flow control valves instrument loop calibration procedure issues, and CRs regarding various pump cooler degraded flow conditions due to blocked orifices. In addition, the inspectors conducted a walk-down of all selected HXs and major components for the NSCW system to assess general material condition and to identify any degraded conditions of selected components. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI)

a. Inspection Scope

The inspectors observed inspection activities and reviewed the documentation and selected supporting records for ISI work activities conducted during Unit 2 refueling outage. The inspection activities, documentation, and supporting records were reviewed for compliance with the Technical Specifications (TS), American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, and other appropriate industry and NRC guidance and standards.

The inspectors reviewed nondestructive examination activities, including ultrasonic (UT) and magnetic particle (MT), on risk important components, and reviewed associated records. The inspectors reviewed evaluations for continued service associated with recordable indications from the last Unit 2 refueling outage. The inspectors reviewed weld examinations on pressure boundary components. The inspectors reviewed component repairs and replacements to ensure they met ASME code requirements. The inspectors observed data collection and reviewed the results of steam generator (SG) tube examinations. The inspectors reviewed licensee actions following the identification of a new degradation mechanism (circumferential cracking at the tubesheet), to ensure that scope expansion was appropriate, tubes were properly screened for in-situ pressure testing, and other corrective actions taken to address the flaws and determine the root cause were appropriate.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors evaluated operator performance on June 22 during licensed operator simulator training associated with Requalification Segment 20043. The simulator scenario began with a reactor power ascension interrupted by a loss of one then both 4.16 kV emergency buses. The inspectors specifically assessed the following areas:

- correct use of abnormal and emergency operating procedures including 18001-C, Primary Systems Instrumentation Malfunction; 18031-C, Loss of Class 1E Electrical Systems; and 19100-C, ECA-0.0 Loss of All AC Power
- ability to implement appropriate event reporting and emergency plan actions in accordance with procedures 91001-C, Emergency Classification and Implementing Instructions, and 91002-C, Emergency Notifications
- ability to identify and implement appropriate TS actions
- clarity and formality of communications in accordance with procedure 10000-C, Conduct of Operations
- proper control board manipulations including critical operator actions
- quality of supervisory command and control
- effectiveness of post-evaluation critique

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the following two equipment problems and associated licensee CRs to evaluate the effectiveness of the licensee's handling of equipment performance problems and to verify the licensee's maintenance efforts met the requirements of 10 CFR 50.65 (the Maintenance Rule) and procedure 50028-C, Engineering Maintenance Rule Implementation. The reviews included adequacy of the licensee's failure characterization, establishment of performance criteria or 50.65 (a) (1) performance goals, and adequacy of corrective actions. Other documents reviewed during this inspection included control room logs, system health reports, the maintenance rule database, and MWOs. Also, the inspectors interviewed system engineers and the maintenance rule coordinator, to assess the accuracy of identified performance deficiencies and extent of condition.

- Reactor Vessel Level Indication System level instrument failure (CR 2003003492)
- Inverter 1BD1I12 failure (CR 2004000682)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the following five risk significant and emergent MWOs to verify plant risk was properly assessed by the licensee prior to conducting the activities. The inspectors reviewed risk assessments and risk management controls implemented for these activities to verify they were completed in accordance with procedure 00354-C, Maintenance Scheduling, and 10 CFR 50.65(a)(4). The inspectors also reviewed the CR database to verify that maintenance risk assessment problems were being identified at the appropriate level, entered into the corrective action program, and appropriately resolved.

- 2B CS system outage (MWOs 20400242 and 20300952)
- 1B MDAFW pump/motor coupling alignment check (MWO 10400060)
- 2B SI system outage (MWOs 20202991, 20300811, 20300950, and 20300810)
- Troubleshoot trip of 2A Emergency Diesel Generator (EDG) (MWO 20401476 and CR 2004001961)
- Unit 1, NSCW pump 5 system outage (MWOs 10302303, 10300601, and 10402059)

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Plant Evolutions

a. Inspection Scope

For the four non-routine plant evolutions described below, the inspectors observed the operating crew's performance, reviewed operator logs, control board indications, and plant computer data to verify that operator response was in accordance with the associated plant procedures.

- March 29, Unit 1 reactor restart in accordance with procedure 12003-C, Reactor Startup (Mode 3 to Mode 2)
- April 17 -18, Unit 2 shutdown for refueling outage in accordance with procedures 12004-C, Power Operation (Mode 1), and 12005-C, Reactor Shutdown to Hot Standby (Mode 2 to Mode 3)
- May 5, response to failure of 4.16 kV emergency bus to properly load shed during 2A Engineered Safety Features Actuation System (ESFAS) testing in accordance with procedure 18031-C, Loss of Class 1E Electrical Systems
- May 16, Unit 2 reactor startup in accordance with procedure 12003-C, Reactor Startup (Mode 3 to Mode 2)

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following six evaluations to verify that they met the requirements of procedure 00150-C, Condition Reporting and Tracking System. This scope included a review of the technical adequacy of the evaluations, the adequacy of compensatory measures, and the impact on continued plant operation.

- Control room normal heating ventilation and air conditioning (HVAC) outside air damper found out-of-position (CR 2004001633)
- Effect of electrical and/or hydraulic transient on 2A EDG and its associated ESF loads (CR 2004002015)
- 2A Control Room Emergency Filtration System (CREFS) start time did not meet acceptance criteria (CR 2004002066)
- 2A CS pump high differential pressure during testing (CR 2004002240)
- Piping downstream valves 2FV-0520 and 2FV-0530 below minimum action level thickness (CRs 2004001604 and 2004001613)
- 1A EDG ESF HVAC supply dampers cycling open and shut (CR 2004002670)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed the licensee’s list of identified operator workarounds and burdens to determine whether any items would adversely affect the operators’ ability to implement abnormal or emergency operating procedures. Additionally, the inspectors reviewed Unit 1 and Unit 2 control room logs, caution tag logs, abnormal configuration logs, MWOs, and the clearance and tagging database, to identify any abnormal plant equipment configurations that might be considered operator workarounds, and to verify the licensee was identifying and documenting operator workarounds in accordance with procedure 10025-C, Work Around Program.

b. Findings

Introduction. A Green non-cited violation (NCV) was identified by the inspectors for failure to maintain adequate Unit 1 and Unit 2 NSCW system operating procedures in accordance with TS 5.4.1.a, which resulted in rendering the Unit 1, Train B NSCW inoperable when the NSCW Tower Bypass Valve hand switch was placed in the “Open Bypass” position without appropriate administrative controls to return the switch to its normal position during a design basis accident.

Description. On June 1, 2004, the licensee placed the Unit 1, Train B NSCW Tower Spray Header Bypass Valve in “Open Bypass” per procedure 13150-1, Nuclear Service Cooling Water System. The switch remained in this configuration for approximately five hours during planned weld repair activities on degraded NSCW tower debris screens. In this configuration, all of the NSCW return cooling water was bypassed from the tower spray header and returned directly to the NSCW basin. The spray header provides the heat removal function during a design basis accident. The inspectors noted that the licensee had not considered this configuration an operator workaround nor a condition that rendered the NSCW train inoperable.

UFSAR Section 9.2.1.1.1, Design Bases, item G, states “the NSCW system is designed to perform its cooling function following a LOCA, automatically and without operator action, assuming a single failure coincident with a loss of offsite power.” With the NSCW spray header bypass valve in the Open Bypass position, the NSCW return temperature control circuitry, is defeated. The valves do not automatically reposition on a Safety Injection actuation. Therefore, during an actual design basis loss of coolant accident (LOCA), the operators would need to reposition the switch in order for the tower spray cooling function to be accomplished. Based on starting from the maximum allowed operating NSCW basin temperature of 90 degrees F, the licensee calculated the switch would need to be repositioned within 30 minutes in order to prevent exceeding the NSCW maximum design temperature of 95 degrees F. The inspectors reviewed alarm response, abnormal operating, and emergency operating procedures and determined that these procedures did not provide instructions for restoring the NSCW system to its proper configuration. The inspectors determined this configuration rendered the NSCW train inoperable.

Analysis. This finding is greater than minor because it affected the Mitigating Systems cornerstone attribute of configuration control and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences by rendering the automatic NSCW heat removal function inoperable. This finding is of very low safety significance (Green) because the duration did not exceed the 72 hour allowed outage time for one inoperable NSCW train and it did not represent an actual loss of service water safety function.

Enforcement. TS 5.4.1.a requires written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Item 3.m, requires procedures be maintained for operating the safety-related service water system. Contrary to this, procedures 13150-1/-2, Nuclear Service Cooling Water System, Revision 32/29, were not properly maintained, in that, the procedures allowed defeating the automatic NSCW cooling design function on June 1 for about five hours without considering the system inoperable or providing adequate administrative controls to ensure necessary manual operator actions would be implemented to ensure the system would operate as designed during a design basis accident. Because the finding is of very low safety significance and the licensee initiated procedural changes to address the issue, this violation is being treated as an NCV, consistent with Section VI.A of the NRC

Enforcement Policy: NCV 05000424, 425/2004004-01: Inadequate NSCW Operating Procedure.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed DCP No. 97-V2N0062, Safety Features Sequencer Control Logic Upgrade, to verify it met the requirements of procedure 58007-C, Design Change Packages. The review was conducted to verify that the modification did not degrade the system design bases, licensing bases, or performance capability, and that plant risk was not increased unnecessarily during implementation of the modification. Documents reviewed are listed in the Attachment. Additional review of the safety evaluation for this DCP was conducted by a regional reactor inspector and is discussed in Section 1R02 of this report.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors either observed post-maintenance testing or reviewed the test results for the following five maintenance activities to verify that the testing met the requirements of procedure 29401-C, Work Order Functional Tests, for ensuring equipment operability and functional capability was restored. The inspectors also reviewed the test procedures to verify the acceptance criteria was sufficient to meet the TS operability requirements.

- Repair Unit 1, Loop 3 Feedwater Regulatory Valve, FV-0530 (MWO 10401276)
- 1B MDAFW pump/motor coupling alignment check (MWO 10400060)
- 2B SI system outage (MWOs 20202991, 20300811, 20300950, and 20300810)
- Unit 1, NSCW pump 5 system outage (MWOs 10302303, 10300601, and 10402059)
- Unit 1, CCW pump 5 system outage (MWO 10301709)

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

Unit 2 Refueling Outage (2R10)

a. Inspection Scope

The inspectors reviewed the licensee's 2R10 Pre-Outage Schedule Risk Assessment Report, dated March 23, 2004, and the 2R10 Refueling Outage Schedule, Revision A, to confirm the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing the outage plan. During the refueling outage, the inspectors observed and monitored licensee controls over the outage activities listed below. Documents reviewed are listed in the Attachment.

- Outage related risk assessment monitoring
- Controls associated with shutdown cooling, reactivity management, reduced and midloop inventory activities, electrical power alignments, containment closure and integrity, and spent fuel pool cooling
- Implementation of equipment clearance activities
- Refueling activities
- Reactor mode changes
- Reactor heatup and repressurization
- Containment cleanup
- Reactor initial startup and reactor physics testing
- Reactor power ascension and related testing

b. Findings

Introduction. A finding with potential safety significance greater than Green was identified by the inspectors for failure to adequately perform a Unit 2 containment building inspection prior to entry into Mode 4, Hot Shutdown.

Description. On May 11, 2004, prior to entering Mode 4, the licensee completed a containment inspection in accordance with procedure 14900-C, Containment Exit Inspection. The purpose of the inspection was to identify and remove any loose debris that remained from the refueling outage that could be transported to the containment emergency sumps. Following the inspection, the licensee continued with plant startup entering Mode 4 and Mode 3, Hot Standby. The inspectors then performed a containment walkdown to verify the licensee's containment inspection had been properly performed. During this walkdown, the inspectors found additional loose debris (e.g., Chem wipes, loose insulation filler material, paper tags, plastic tie-wraps, ziplock bag, pieces of red duct tape, pieces of absorbent material and an assortment of other small paper and plastic items) that were not identified during the licensee's containment inspection. All items of concern were removed at the time the inspectors exited containment with the exception of the loose insulation filler material. This loose insulation material was later removed by the licensee. The inspectors also found several areas of degraded/damaged fibrous insulation in the vicinity of the steam generators.

The licensee initiated CRs 2004002140, 2148, 2154 and Request for Engineering Review (RER) 2004-V0275, to evaluate the impact the debris found could have on the containment emergency sumps. The results of the licensee's evaluation indicated the quantity of the debris found could block the sump screens such that the RHR pumps would be unable to perform their intended safety function during recirculation phase of a design basis LOCA.

Analysis. This finding was greater than minor because it affected the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was also determined to have potential safety significance greater than very low significance because the condition could have resulted in the loss of the RHR recirculation safety function.

Enforcement. TS 5.4.1.a requires, in part, that written procedures be implemented covering activities listed in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, which includes procedures for startup and changing modes of operation that impact emergency core cooling systems (ECCS). Procedure 14900-C, Containment Exit Inspection, Revision 12, step 5.1, requires visual inspection of accessible areas of containment and removal of debris that could be transported to the containment emergency sumps. Contrary to the above, licensee personnel performing containment closeout inspections completed on May 11, 2004, failed to identify and remove loose debris that could be transported to the containment emergency sumps. The finding does not present not an immediate safety concern because the debris identified was removed from containment. Pending completion of a final significance determination, this finding is identified as an Apparent Violation (AV): AV 05000425/2004004-02, Failure to Adequately Perform Containment Closeout Inspection.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the following seven surveillance test procedures and either observed the testing or reviewed test results to verify that testing was conducted in accordance with the procedures and that the acceptance criteria adequately demonstrated that the equipment was operable. Additionally, the inspectors reviewed the CR database to verify that the licensee had adequately identified and implemented appropriate corrective actions for surveillance test problems.

Surveillance Tests

- 28210-C, Main Steamline Code Safety Valve Setpoint Verification (Unit 2, loops 1 and 4)
- 14667-2, Train B Diesel Generator and ESFAS Test
- 14666-2, Train A Diesel Generator and ESFAS Test
- LPPT-GAE/GBE-01, Low Power Physics Test Program with Dynamic Rod Worth Measurement

In-Service Tests

- 14897-2, ECCS Accumulator Inservice Check Valve Test
- 14721-2, ECCS Subsystem Flow Balance and Check Valve Refueling Inservice Test

Containment Isolation Valve Tests

- 14378-2, Containment Penetration No. 78 Containment Sump Pumps Discharge Local Leak Rate Test (for valve 2HV-0781)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors evaluated the following three Temporary Modifications (TMs) and associated 10 CFR 50.59 screenings against the system design basis documentation and UFSAR to verify that the modifications did not adversely affect the safety functions of important safety systems. Additionally, the inspectors assessed whether the modifications were developed and implemented in accordance with procedure 00307-C, Temporary Modifications.

- TM 2004-V2T037, Isolate one cooling coil bank on Unit 2 Containment Cooler 2
- TM 2004-V2T035, Install temporary vibration monitoring on Unit 2 letdown line
- TM 2004-VAT027, Provide temporary alternate cooling from plant firewater system to both unit's turbine plant cooling water pump bearings

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

Access Controls. During the weeks of April 26 and May 3, 2004, licensee activities for monitoring workers and controlling access to radiologically significant areas were evaluated. The inspectors reviewed procedural guidance and directly observed implementation of administrative and physical controls; appraised radiation worker and technician proficiency in implementing radiation protection program activities; and assessed worker exposures to radiation and radioactive material.

Radiological postings and material labeling were directly observed during tours of the Unit 1 (U1) and Unit 2 (U2) auxiliary buildings, U2 containment, and outside radioactive material storage areas. The inspectors took independent surveys in these areas to verify posted radiation levels and to compare with current licensee survey records. During the plant tours, control of Locked High Radiation Area (LHRA) and Very High Radiation Area (VHRA) keys and the physical status of locked doors were examined. In addition, the inspectors observed radiological controls for non-fuel items stored in the spent fuel pools (SFPs). The inspectors reviewed selected parts of eight Health Physics (HP) related procedures, evaluated five radiation work permits (RWPs), and discussed current access control program implementation with HP supervisors.

During the onsite inspection, radiological controls for jobs related to Unit 2 Refueling Outage Ten (2R10) work activities in High Radiation Areas (HRAs) were observed and discussed. The inspectors attended pre-job briefings for SG work, cavity decontamination, and RHR check valve breach. The inspectors observed the work activities either directly or by closed-circuit television and evaluated workers' adherence to RWP guidance and HP Technician (HPT) proficiency in providing job coverage. Controls for monitoring and limiting exposure to airborne radioactive material were reviewed, as applicable. The inspectors evaluated electronic dosimeter (ED) alarm setpoints for consistency with radiological conditions in and around the work areas and reviewed the use of extremity dosimetry for areas with high dose rate gradients around the RHR check valve. In addition, the inspectors quizzed workers on RWP requirements and expected response to dosimeter alarms.

The inspectors evaluated worker exposures through review of data associated with discrete radioactive particle and internal contamination events. The inspectors reviewed and evaluated personnel contamination event logs for 2R10, whole-body count (WBC) records, dose calculations, and procedural guidance.

HP program activities were evaluated against 10 CFR Part 20; TS Section 5, Administrative Controls; Regulatory Guide RG 8.38, Control of Access to High and Very High Radiation Areas in Nuclear Power Plants; and approved licensee procedures. Licensee guidance documents, records, and data reviewed are listed in Section 2OS1 of the report Attachment.

Problem Identification and Resolution. Five CRs and one audit associated with HPT and radworker practices, radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with HP supervisors. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve selected issues in accordance with procedure NMP-GM-002-GL02, Corrective Action Program Details and Expectations Guideline. Specific documents reviewed are listed in the report Attachment.

b. Findings

Introduction. A Green finding was identified by the inspectors for the failure to maintain adequate administrative control over keys to VHRAs.

Description. On April 29, 2004, during preparations for a plant tour to observe LHRA controls, the inspectors were shown where keys to LHRAs and VHRAs were stored in the HP foreman's office. Upon examination of the VHRA key box, the inspectors noted that the box appeared to be unlocked and proceeded to challenge the locking mechanism by pulling on the handle. The box was not locked and it opened with little effort, thereby exposing the VHRA keys inside.

The inspectors also noted that, even if the VHRA key box was properly locked, the licensee's program for controlling the keys was deficient. For example, the lock on the box consisted of a simple latch and catch configuration, similar to the lock on a desk drawer, and did not provide an appropriate level of security. In addition, the process for obtaining the key to the lockbox was inadequate. The key was locked in an HP supervisor's desk; however, the key to that lock was on a large key ring with many other keys in an unlocked desk in the HP foreman's office.

Although VHRAs with radiation fields greater than 500 Rad/hr at one meter did exist during the time that the box was unlocked, no evidence could be found that any unauthorized access had been made to a VHRA. An inventory performed immediately after discovery confirmed that all VHRA keys were currently inside the key box. In addition, all VHRA doors were checked and verified to be in a locked position.

Analysis. The inspectors determined that this finding is a performance deficiency because licensees are expected to implement programs designed to preclude unauthorized access to VHRAs, including appropriate control of keys to these areas. This finding is more than minor because if left uncorrected the issue could become a more significant safety concern in that someone could gain unauthorized access to a VHRA by taking a key from the poorly controlled lockbox. The identified issue was assessed using the Occupational Radiation Safety SDP. Based on the fact that there was no overexposure, there was no evidence of unauthorized access into a VHRA, and the licensee's ability to assess dose was not compromised, the finding was determined to be of very low safety significance (Green). The inspectors noted that immediate licensee corrective actions, such as moving the VHRA keys from the wall-mounted lockbox to a secure safe and putting the key to the safe on the person of the HP Superintendent (or designee), helped to mitigate the potential consequences of this issue.

Enforcement. The inspectors determined that no violation of TS or 10 CFR Part 20 requirements occurred because other administrative controls were in place to deter unauthorized access to VHRAs, such as superintendent approval for key check-out and the use of teledosimetry. This finding has been entered into the licensee's corrective action program as CR 2004001876 and is identified as finding (FIN): FIN 05000424, 425/2004004-03: Failure to Implement Adequate Administrative Control Over Keys to VHRAs.

2OS2 As Low As Reasonably Achievable (ALARA)

a. Inspection Scope

ALARA. The inspectors reviewed ALARA program guidance and its implementation for ongoing 2R10 activities. Development of dose expenditure goals for selected outage tasks estimated to exceed one person-rem were reviewed and discussed with site management. The inspectors reviewed applicable ALARA Committee meeting details, HP self-assessments, post-job reports, and ALARA Planning Work Sheets associated with the following 2R10 activities:

- RWP 04-2004, Install/Remove Scaffold Containment
- RWP 04-2302, Eddy current test, Steam Generators Nos. 2 & 3
- RWP 04-2410, Ultrasonic Fuel Cleaning
- RWP 04-2505, Rx Head and Under-vessel inspection and cleaning
- RWP 04-2609, DCP 20V2N041 S/G Eddy Current Platform
- RWP 04-2610, Remove Replace Letdown Heat Exchanger

The inspectors reviewed and discussed dose rate and cumulative dose expenditure data trends associated with selected systems, equipment, and tasks. For selected outage tasks, the inspectors compared current dose rate and dose expenditure results with data used in planning estimates and with data from the previous 2R9 refueling outage. The inspectors evaluated selected data associated with dose reduction initiatives including shutdown chemistry and cleanup, planning and sequencing of work activities, dose estimation techniques and concurrent reduced ED alarm set points, system equipment flush controls, temporary shielding, and cobalt reduction initiatives for valve replacement.

Knowledge of ALARA program guidance and staff proficiency in program implementation was assessed through observation of selected work activities, comparison of estimated and current dose expenditure data for selected tasks, and discussions of selected outage tasks with responsible supervisors and managers.

Program implementation and results were reviewed against the facility's ALARA work plans; the UFSAR; 10 CFR Part 20 requirements; and procedural guidance documented in Section 2OS2 of the report Attachment.

Problem Identification and Resolution. Licensee CR documents associated with dose reduction initiatives and ALARA activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure NMP-GM-002-GL02 and procedure 00150-C, Condition Reporting and Tracking System. Specific CR documents reviewed and evaluated are listed in Section 2OS2 of the report Attachment.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization. The inspectors evaluated the operability and installed configuration of selected liquid and solid radioactive waste (radwaste) processing systems and equipment. Inspection activities included document review, interviews with plant personnel, and direct inspection of processing equipment and piping in the Radwaste Processing Facility (RPF) and the Auxiliary Radwaste Building (ARB).

The inspectors directly observed material condition and installed configurations for selected RPF equipment and piping. The system engineer was interviewed regarding Process Control Program (PCP) equipment function and operability. Radwaste operators were queried to assess knowledge of resin sluicing and dewatering operations. The material condition and licensee identification of ARB radwaste processing equipment abandoned in place were observed and discussed during tours.

For selected radwaste material sent to licensed processing or burial facilities between January 1, 2003 and May 7, 2004, the inspectors reviewed and discussed waste stream sampling activities; licensee gamma spectroscopy data; and offsite vendor sample sizes and associated detection capabilities for 10 CFR Part 61.55 analyses. The licensee's sampling method for waste stream analyses and the use of scaling factors for hard-to-detect nuclides were assessed. Waste characterizations for dry active waste (DAW) and selected drums containing spent filters and trash were reviewed in detail. The most recent audit of the offsite Part 61.55 vendor's quality program was reviewed and evaluated. In addition, DAW waste stream radionuclide data were reviewed for consistency in radionuclide composition.

Program implementation was evaluated against Part 61.55; licensee PCP requirements; the Branch Technical Position on Radioactive Waste Classification; UFSAR Section 11.4, Solid Waste Management System, and the approved procedures listed in Section 2PS2 of the report Attachment.

Transportation. The inspectors evaluated the licensee's activities related to transportation of radioactive material. The evaluation included review of shipping records and procedures, assessment of worker training and proficiency, and direct observation of shipping activities.

The inspectors assessed selected shipping-related procedures and practices for compliance with applicable regulatory requirements. Records and surveys for selected shipments made during calender year 2004 were reviewed for consistency with licensee procedures and for completeness and accuracy. On May 6, 2004, the inspectors observed the preparation and loading of a resin cask shipment to an offsite volume reduction facility (Shipment No. RVRS-04-018). The inspectors interviewed the responsible technician regarding packaging and vehicle radiation and contamination control limits. In addition, the inspectors directly observed selected radiation surveys associated with the prepared shipment.

Transportation program guidance and implementation were reviewed against requirements detailed in 10 CFR Part 71, 49 CFR Parts 170-189, and applicable licensee procedures listed in Section 2PS2 of the report Attachment.

Problem Identification and Resolution. Licensee CRs and self-assessments associated with PCP and transportation activities were reviewed and evaluated. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with Procedure 00150-C, Condition Reporting and Tracking System. Specific CR documents reviewed and evaluated are listed in Section 2PS2 of the report Attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors sampled licensee submittals for the PIs listed below to verify the accuracy of the PI data reported during the indicated period. The PI definitions and guidance contained in procedures 00163-C, NRC Performance Indicator Preparation and Submittal; 50025-C, Reporting of Mitigating System Performance Indicator Unavailability; and NEI 99-02, Regulatory Assessment Indicator Guideline, Rev. 2, were used to verify the basis in reporting for each data element.

Mitigating Systems Cornerstone

- Safety System Functional Failures

The inspectors reviewed Licensee Event Reports (LERs), operator log entries, monthly operating reports, monthly PI summary reports, and NRC inspection reports for the PI data submitted by the licensee during the period from April 2003 through March 2004, for both Unit 1 and Unit 2.

Barrier Integrity Cornerstone

- Reactor Coolant System (RCS) Activity
- RCS Leak Rate

The inspectors reviewed completed radiochemistry data sheets from procedure 35110-C, Chemistry Control of the Reactor Coolant System, operating logs, leakage calculation results obtained from procedures 14905-1,-2, RCS Leakage Calculation (Inventory Balance), and the licensee's monthly PI Summary reports for the PI data submitted by the licensee during the period from April 2003 through March 2004, for both Unit 1 and Unit 2.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

The inspectors reviewed three CR records for HRAs, LHRAs, VHRAs, and unplanned exposure occurrences for the period of September 2003 through April 2004, to ensure that non-conformances were properly classified as PIs. The inspectors also reviewed electronic dosimeter alarm logs and monthly PI Reports for the same period. Documents reviewed are listed in the Attachment.

Public Radiation Safety Cornerstone

- Radiological Environmental Technical Specifications (RETS)/ODCM Radiological Effluent Occurrences

The inspectors reviewed the Radiological Control Effluent Release Occurrences PI results for the Public Radiation Safety Cornerstone from September 2003 through April 2004. For the assessment period, the inspectors reviewed data reported to the NRC, procedural guidance for reporting PI information, effluent release records, and three CRs documented in Section 4OA1 of the report Attachment. In addition, the inspectors reviewed monthly PI reports and records of compensatory actions taken for out-of-service effluent monitors.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

1. Daily Screening of Corrective Action Items

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by either attending daily screening meetings that briefly discussed major CRs, or accessing the licensee's computerized corrective action database and reviewing each CR that was initiated.

b. Findings and Observations

No findings of significance associated with the reviewed issues were identified.

2. Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, the inspectors performed a review of the licensee's corrective action program and associated documents to identify any trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered trends in human performance errors. The inspectors' review considered the six month period of August 2003 through January 2004, although some examples expanded beyond those dates when the scope of potential trends was warranted. Documents reviewed included licensee quarterly corrective action trending reports, engineering system health monitoring reports, major plant issues reports, department self-assessment activities, and Quality Assurance Department audit reports.

b. Findings and Observations

No findings of significance associated with the reviewed issues were identified.

3. Cross Cutting Aspects of Findings

Section 4OA5.1 describes a finding for failure to perform a timely and appropriate operability evaluation for a generic AFW discharge control valve equipment degradation. The direct cause of this finding involved the cross-cutting area of Problem Identification and Resolution.

4OA3 Event Follow-up

1. (Closed) LER 05000424/2004001-00: Manual Reactor Trip Following Loss of Main Feedwater Pump Speed Control

a. Inspection Scope

The inspectors reviewed the LER, CR 2004001185, and Event Report 1-2004-03 to verify that the cause of the March 27, 2004, Unit 1 reactor trip was identified and that corrective actions were reasonable. The manual reactor trip was caused by the feedwater pump turbine control valve sticking open resulting in the inability to properly control feedwater flow. The inspectors reviewed the plant parameters to verify that timely notifications were made in accordance with 10 CFR 50.72, that licensee staff properly implemented the appropriate plant procedures, and that plant equipment performed as required.

b. Findings

Introduction. A Green self-revealing finding was identified for failure to have adequate feedpump turbine control valve work instructions.

Description. The feedpump turbine control valve stuck open due to misalignment of the hydraulic cylinder shaft with bushings in the cylinder cover. The licensee determined that the technique for ensuring proper alignment of the upper and lower bushings on the hydraulic cylinder shaft was not incorporated into site instructions when the valve was rebuilt during the previous refueling outage.

Analysis. This finding is more than minor because it affected the procedure quality attribute of the Initiating Events cornerstone and affected the cornerstone objective by resulting in a reactor trip. The finding was determined to be of very low safety significance because it did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiator, it did not contribute to a loss of mitigation equipment functions, and it did not increase the likelihood of a fire or internal/external flood.

Enforcement. The inspectors determined there was no violation of regulatory requirements because it occurred on non-safety-related secondary plant equipment. This finding is in the licensee's corrective action program as CR 2004001185 and is identified as FIN 05000424/2004004-04, Inadequate Feedpump Turbine Control Valve Work Instruction.

2. (Closed) LER 05000424, 425/2004002-00: Closure of Control Room Air Damper Results in Technical Specification Non-Compliance

On April 22, 2004, during a shift turnover control room walkdown, the Unit 1 Shift Supervisor found the Control Room Normal HVAC Outside Air Damper, AHV-12153, closed. This condition prevented air flow past both Unit 1 control room air intake radioactive gas monitors rendering them inoperable. The Unit 2 radioactive gas monitors were already inoperable due to maintenance activities. The licensee determined that the four monitors were inoperable for approximately 17 hours. With all four monitors inoperable, TS 3.3.7, Condition P, required that a CREFS train in each unit be placed in service within 1 hour otherwise TS 3.0.3 is applicable which requires the units to be placed in Mode 3 within 7 hours. The licensee determined that cycling of a cell switch during maintenance activities on Unit 2 AB05 switchgear closed the outside damper. This issue was entered in the licensee's corrective action program as CR 2004001633. No additional findings of significance were identified by the inspectors. The regulatory significance of this issue is discussed in Section 4AO7.

3. (Closed) LER 05000425/2004001-00: Auxiliary Feedwater System Actuation While Breaking Condenser Vacuum

a. Inspection Scope

The inspectors reviewed the LER and CR 2004001501, which documented this event in the licensee's corrective action program, to verify the cause of the April 18, 2004, Unit 2 unplanned ESF actuation (i.e., AFW system actuation) was identified and that corrective actions were reasonable. The inspectors reviewed the plant parameters to verify that timely notifications were made in accordance with 10 CFR 50.72, that licensee staff properly implemented the appropriate plant procedures, and that plant equipment performed as required.

b. Findings

Introduction. A Green self-revealing NCV was identified for failure to follow the unit operating procedure.

Description. The licensee determined that the root cause of the event was the failure of the control room operators to adequately sequence the work activities associated with the shutdown of the unit for starting the planned refueling outage. Simultaneous shutdown activities distracted the operators who failed to follow the unit operating procedure to disable the AFW actuation signal prior to breaking condenser vacuum.

Analysis. This finding is greater than minor because it affected the human performance attribute of the Initiating Events cornerstone and affected the cornerstone objective, in that, it caused an unplanned ESF actuation. The finding is of very low safety significance (Green) because it did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiator, did not contribute to a reactor trip with the loss of mitigation equipment functions, and did not increase the likelihood of a fire or internal/external flood. The direct cause of this finding involved the cross-cutting area of Human Performance.

Enforcement. TS 5.4.1.a requires that written procedures be implemented covering the activities listed in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, which includes procedures for placing the plant in cold shutdown. Procedure 12006-C, Unit Cooldown to Cold Shutdown, Revision 63, Section E4.3, requires the AFW actuation signal be blocked prior to breaking condenser vacuum. Contrary to the above, operations personnel performing the activity on April 18, 2004, failed to block the AFW actuation signal prior to breaking condenser vacuum. Because the finding was of very low safety significance and has been entered into the licensee's corrective action program as CR 2004001501, this violation is being treated as an NCV in accordance with Section VI.A of the NRC Enforcement Policy: NCV 05000425/2004004-05, Failure to Follow Unit Operating Procedure.

4OA4 Cross Cutting Aspects of Findings

Section 4OA3.3 describes a finding for failure to follow the unit operating procedure when breaking condenser vacuum resulting in an unexpected ESF actuation. The direct cause of this finding involved the cross-cutting area of Human Performance.

4OA5 Other

1. (Closed) URI 05000424, 425/2003005-01: Review Results of Licensee Auxiliary Feedwater Discharge Control Valve Inspections for Previously Identified Pilot Plug Assembly Concerns

a. Inspection Scope

This item was unresolved pending review of the licensee's operability evaluation and internal valve inspections for a common cause AFW valve degradation issue.

b. Findings

Introduction. A Green finding was identified by the inspectors for failure to perform a timely and appropriate operability evaluation for a common cause failure mechanism that potentially affected all the AFW discharge control valves (DCVs) on both units.

Description. During initial review of CR 2003003157 in December 2003, the inspectors were concerned the licensee had not initiated an operability evaluation to assess the impact of an identified common cause degradation associated with the October 2003 failure of AFW DCV 1HV5137. This degradation involved the potential flow induced metal fatigue failure of a cotter pin designed to secure the pilot plug assembly retaining nut to the valve stem. The licensee identified this degradation as the cause of the October 2003 failure of DCV 1HV5137, as well as a similar failure in 1989 with 2HV5132. In both failures, the missing cotter pin had allowed the retaining nut to unscrew from the valve stem resulting in the loose pilot plug assembly parts blocking AFW flow. While this failure mechanism had been the subject of a 1988 valve vendor advisory notice, the licensee could not find any evidence of having visually inspected the AFW DCVs in direct response to the advisory. The inspectors concluded that all the AFW DCVs on both units were potentially susceptible to this degradation. The licensee's planned corrective actions for the October 2003 valve failure included disassembly and visual inspection of the cotter pin condition on all other AFW DCVs during the March 2005 Unit 1 refueling outage and April 2004 Unit 2 refueling outage. On December 5, 2003, considering the severity of the adverse safety consequences associated with the potential failures, the inspectors expressed concerns to the licensee regarding both the need to perform an operability evaluation for the common cause degradation and timeliness of their valve inspection plans.

As a result of the inspectors' concerns, on December 17, 2003, the licensee visually inspected the internals of Unit 1 MDAFW valves 1HV5132 and 1HV5134. While the cotter pins to both valves were found missing, the retaining nuts were still in place and appeared to be restrained by disrupted metal on the valve stem threads that was created by the vibrating movement of the cotter pin in its slot prior to its failure. On December 22, 2003, the licensee completed an operability evaluation for the remaining AFW DCVs not yet inspected on both units and concluded the remaining valves were operable. The inspectors reviewed the evaluation and determined the licensee's conclusions were poorly justified. One of the licensee's main points for operability was the lack of evidence that the valves had ever been disassembled at the site to the extent of disturbing the condition of the vendor installed cotter pins. The licensee assumed that vendor installed cotter pins were not susceptible to potential failure. However, this failed to explain why at least one of the Unit 1 AFW DCVs with no record of cotter pin disturbance had a missing cotter pin. While resolving this and other weaknesses in the licensee's operability evaluation, the licensee informed the inspectors of their decision to accelerate their inspection schedule for the remaining AFW DCVs.

By April 2004, the licensee had inspected and repaired (by staking the retaining nuts to the stem) all the remaining AFW DCVs. The licensee found missing cotter pins on all of the remaining MDAFW valves, however, as with the two previous valve inspection results, the retaining nuts were still in place and restrained from backing off by disrupted metal on the valve stem threads. The TDAFW DCV cotter pins were all found intact, which was consistent with the licensee's assumption that the TDAFW system was less susceptible to cotter pin failure due to significantly less operating time than the MDAFW system. The inspectors considered the additional licensee inspection and repair actions adequate to resolve the common cause degradation issue.

<u>Analysis</u>. The failure to perform a timely and appropriate operability evaluation for the common cause valve degradation is greater than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance because, although the MDAFW DCVs with the missing cotter pins were considered degraded, the pilot plug assembly retaining nuts for all the valves were still held in place by the disrupted metal, therefore the immediate functional capability of the valves was not actually impacted. The direct cause of this finding involved the cross-cutting area of Problem Identification and Resolution.

<u>Enforcement</u>. The inspectors determined that the failure to perform a timely and appropriate operability evaluation did not constitute a violation of regulatory requirements. This finding is identified as FIN 05000424, 425/2004004-06: Failure to Perform Timely and Appropriate Operability Evaluation of AFW Valve Degradation.

2. <u>(Discussed) NRC Temporary Instruction (TI) 2515/150, Rev. 2, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009) (Unit 2)</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the licensee's inspection activities related to the Unit 2 reactor pressure vessel (RPV) head and vessel head penetration (VHP) nozzle inspection activities in response to NRC Order EA-03-009, issued February 11, 2003. In the licensee's response to NRC Bulletin 2002-02, they calculated the effective degradation years based on time and head temperature which placed both units in the "Low Susceptibility" ranking for nozzle leakage potential. In accordance with NRC Order EA-03-009, the licensee conducted a bare metal visual examination of the Unit 2 reactor head during the current refueling outage. The inspection included interviews with visual inspection (VT) examination personnel, review of VT procedures, assessment of VT personnel training and qualifications, and observation/assessment of VT examinations. Specifically, the inspectors reviewed or observed the following:

- observed portions of in-process bare metal remote video VT inspection of VHP Nozzle Nos. 1, 4, 5, 7, 8, 12, 16, 20, 26, 33, 34, 40, 46, 52, 58, 60, 62, 64, 65, 66, 70, 71, 72, 72, and 78 (including space around nozzle)

- observed portions of the bare metal video tape inspections for selected nozzles

- reviewed licensee procedures and inspection results of visual examinations to identify potential boric acid leaks from pressure-retaining components above the reactor vessel head

- independently assessed the condition of the reactor vessel head by direct visual observation looking for loose debris, insulation, dirt, boron deposits, and boric acid corrosion

b. Findings and Observations

In accordance with the requirements of TI 2515/150, the inspectors evaluated and answered the following questions:

(1) Was the examination performed by qualified and knowledgeable personnel?

Yes. The inspectors determined that the VT examinations were performed by trained and ASME VT-2 Level III qualified inspection personnel. The examiners were experienced and were required to have four hours of additional specific industry developed training associated with inspecting VHPs for leakage.

(2) Was the examination performed in accordance with demonstrated procedures?

Yes. The inspectors verified that the bare metal inspections were conducted in accordance with procedures ES-MISN-V-738, Visual Examination of Reactor Vessel Head Penetrations and Base Material (Remote and Direct), 84008-C, RPV Alloy 600 Material Inspection and Reports, and 85020-C, Visual Inspection.

(3) Was the examination able to identify, disposition, and resolve deficiencies and capable of identifying the primary water stress corrosion cracking or vessel head corrosion phenomena described in NRC Order EA-03-009?

Yes. The inspectors concluded that the reactor head access, available lighting, remote imagery equipment capabilities, and procedural imagery resolution requirements were adequate to detect leakage from VHPs and identify boric acid deposits and corrosion. The visual examination was capable of identifying primary water stress corrosion cracking through evidence of leakage from a VHP.

(4) What was the condition of the reactor head?

The licensee's remote camera visual examination was conducted under the reflective mirror insulation and the as-found head condition was generally free of debris, dirt, or large boron deposits. Some slight boron surface residue was observed in various areas that the licensee knew to be from previous leakage of vessel head mechanical connections during the previous operating cycle; however, this did not interfere with the examination and had not resulted in any boric acid corrosion concerns.

(6) Could small boron deposits, as described in Bulletin 2001-01, be identified and characterized?

Yes. The inspectors determined that the visual clarity and color of the video inspection process allowed for effective identification and characterization of boron deposits as described in Bulletin 2001-01.

(7) What material deficiencies were identified that required repair?

None. The licensee did not identify any leaking VHPs nor any boric acid corrosion as a result of their examinations.

(8) What impediments to effective examination were present?

None. The visual examination of the reactor head included 100 percent circumferential coverage of each VHP and its associated annulus region and all areas of the reactor vessel head surface except for a small inaccessible area where the control rod drive mechanism shroud support structure and reflective metal insulation meet the vessel head. This area was estimated to be less than one percent of the vessel head surface and not significant for determining if head wastage was present.

(9) What was the basis for the temperatures used in the susceptibility ranking calculation?

The licensee used 560°F as the reactor vessel head temperature for both units. Based on previous discussions with licensee personnel, this value was obtained from Westinghouse and was the Vogtle reactor T-cold design temperature.

(10) Did procedures exist to identify potential boric acid leaks from pressure-retaining components above the reactor vessel head?

Yes. Licensee procedures 84008-C, RPV Alloy 600 Material Inspection and Reports, 85020-C, Visual Inspection, and 14864-1/2, Containment General Leak Inspection, provided adequate instructions for conducting visual inspections during each refueling outage to identify potential boric acid leaks from pressure-retaining components above the reactor vessel head. The licensee did not identify any indications of boric acid leakage from pressure-retaining components above the RPV head.

3. <u>(Closed) TI 2515/152, Rev. 1, Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02) (Unit 2)</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the licensee's inspection activities related to the Unit 2 reactor vessel lower head penetrations in response to NRC Bulletin 2003-02, Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity, in accordance with NRC TI 2515/152, Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02, dated August 21, 2003). The inspection included review of VT procedures, assessment of VT personnel training and qualifications, and observation and assessment of VT examinations. Specifically, the inspectors reviewed or observed the following:

<u>Bare Metal Visual Examination</u>:

- visually inspected the lower penetration nozzles and head from one location where insulation was removed

- observed approximately half of the in-process bare metal remote video VT inspection

- reviewed a sampling of pictures taken from the video inspection of the remaining half of the inspection that was not observed

b. <u>Findings and Observations</u>

The inspectors determined that the VT examinations were performed by a trained and ASME VT-2 Level III qualified examiner. The examiner was experienced and had additional training in inspecting the lower head penetrations. The inspectors verified the adequacy of procedure ES-MISN-V-738, Visual Examination of Reactor Vessel Head Penetrations and Base Material (Remote and Direct), used to conduct the examination. The inspectors verified the inspection was performed in accordance with procedures 85020-C, Visual Inspection, and 84008-C, RPV Alloy 600 Material Inspections and Reports.

The licensee performed VT examinations of all 58 nozzle penetrations and the lower head. The VT-2 inspection included inspection of 100 percent of the circumference of each nozzle and was capable of identifying any pressure boundary leakage as described in the bulletin and any lower head corrosion. There were no impediments identified that would impact VT examination. Based on observation of the inspection process, the inspectors concluded that deficiencies were properly identified and resolved.

The inspectors observed that the visual clarity, resolution and color of the video inspection process allowed for effective visual examination of the vessel lower head surface and 100% circumferential coverage of each head penetration and its associated annulus region. The visual inspection was capable of identifying small debris or boric acid deposits as a result of primary water stress-corrosion cracking through evidence of leakage from a penetration. No leakage was identified from any of the vessel lower head penetrations.

The examination involved a remote visual inspection and video taping of the lower reactor vessel head surface in the area of the 58 nozzles as well as the circumference of each nozzle. The inspection was conducted using a crawler mounted video camera positioned on the lower head insulation below the reactor vessel.

There were no significant examples of leakage sources, insulation, debris, dirt, or other physical impediments that prevented a complete visual examination. There was no evidence of any boric acid deposits originating at the interface between the vessel and the penetrations or from pressure-retaining components located above the lower head. The vessel lower head was generally free of debris, dirt, or boron deposits and required no cleaning. No material deficiencies were found that required repair.

4. (Closed) TI 2515/153, Reactor Containment Sump Blockage (NRC Bulletin 2003-01) (Unit 2)

a. Inspection Scope

The inspectors reviewed the licensee's activities in response to NRC Bulletin 2003-01, Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors, dated June 9, 2003. The inspection included review of the licensee's 60 day Bulletin response letter, review of interim compensatory measures implemented to reduce the potential risk due to post-accident debris blockage on emergency sump recirculation, and walkdown of the Unit 2 containment prior to restart from the current refueling outage to identify if any sources of potential debris existed that could impact the containment recirculation sump performance. The inspectors assessed whether the licensee either (1) performed a plant-specific evaluation of the ECCS and CS system recirculation functions for impact of post-accident debris blockage effects, or (2) effectively implemented reasonable compensatory measures.

b. Findings and Observations

By letter dated August 7, 2003, the licensee responded to NRC Bulletin 2003-01 and described interim compensatory measures that had been implemented or were planned. These compensatory measures were designed to provide interim actions in order to reduce the risk which may be associated with potentially degraded or nonconforming ECCS and CS recirculation functions until a more detailed evaluation could be completed to verify conformance with applicable regulatory requirements. The inspectors verified the following compensatory measures identified in the licensee's response had been implemented.

(1) Operator training on indications of and response to sump clogging:

The inspectors reviewed Emergency Operating Procedure (EOP) 19111-C, ECA 1.1 Loss of Emergency Coolant Recirculation, and verified that initial and continuing operator training contained guidance associated with the procedure for dealing with a complete loss of ECCS and CS recirculation capability. The inspectors reviewed Standing Order C-2003-5, issued by the Operations Department on July 24, 2003, which provided additional guidance to the operators emphasizing the need for continuous monitoring of ECCS recirculation plant parameters to identify degrading conditions and actions to be taken if sump blockage is encountered. Also, the inspectors verified that additional sump blockage guidance was provided to all licensed operators in the continuous operator training (Segment #20041) that was completed in February 2004.

(2) Procedural modifications, if appropriate, that would delay the switchover to containment sump recirculation:

The licensee determined that existing guidance contained in EOP 19111-C and the additional guidance provided in Standing Order 2003-5 provided adequate guidance for dealing with a complete loss of sump recirculation. However, the licensee planned to review any future Westinghouse Owners Group EOP recommendations, if issued, to determine if any Vogtle specific procedural changes would be appropriate.

(3) Ensuring that alternate water sources are available to refill the Refueling Water Storage Tank (RWST) or to otherwise provide inventory to inject into the reactor core and spray into the containment atmosphere:

The licensee determined that no additional changes were necessary to existing plant procedures designed to refill the RWST or provide cooling to the reactor core or containment. The inspectors reviewed the following procedures and verified that they contained adequate guidance for providing alternate borated and unborated water sources to refill the RWST or inject into the reactor core and containment:

- 19111-C, ECA 1.1 Loss of Emergency Coolant Recirculation
- 13701-2, Boric Acid System
- Severe Accident Management Guideline 8 (SAG-8), Flood Containment

(4) More aggressive containment cleaning and increased foreign material controls:

The licensee determined that adequate procedural guidance currently existed for containment cleaning and foreign material control. The inspectors reviewed the following procedures in order to verify proper implementation during the recent Unit 2 refueling outage.

- 00254-C, Foreign Material Exclusion and Plant Housekeeping Programs
- 00303-C, Containment Entry
- 00309-C, Control of Unattended Temporary Materials in Containment in Modes 1-4
- 12000-C, Post Refueling Operations (Mode 6 To Mode 5)
- 12001-C, Unit Heatup to Hot Shutdown (Mode 5 To Mode 4)
- 14864-2, Containment General Leak Inspection
- 14900-C, Containment Exit Inspection

The inspectors performed a routine walkdown of the Unit 2 containment prior to plant restart on May11, 2004, following the licensee's containment cleanup activities to verify that debris was not left that could affect the performance of the containment sumps. During this walkdown, the inspectors identified materials (e.g., Chem wipes, insulation, paper tags, plastic tie-wraps, ziplock bag, pieces of red duct tape, pieces of absorbent material and an assortment of other small paper and plastic items) that were not removed by the licensee's cleanup activities. The inspectors also found several areas of degraded/damaged insulation in the vicinity of the steam generators. The licensee documented these items in CRs 2004002140, 2154 and 2148. The loose debris was removed and the degraded insulation was either repaired or removed prior to plant restart. The licensee subsequently determined that during a design basis LOCA, the material left in containment and the additional insulation debris generated from the damaged insulation, could have caused the loss of the recirculation function of both trains of RHR due to sump blockage. This issue is discussed in more detail in Section 1R20 of this report.

(5) Ensuring containment drainage paths are unblocked:

The inspectors reviewed procedures 12000-C and 12001-C and verified that they contained adequate instructions for ensuring that the reactor cavity drains were properly opened prior to the plant entering Mode 4 following a refueling outage to ensure that the drainage path to the containment sump was unblocked. In addition, the inspectors verified the procedures were properly implemented during the current Unit 2 refueling outage.

(6) Ensuring sump screens are free of adverse gaps and breaches:

The inspectors reviewed procedure 14903-2, Containment Emergency Sump Inspection, and verified that it contained adequate guidance for identifying adverse gaps and breaches. In addition, the inspectors inspected the Unit 2 containment sumps during the current refueling outage and found no issues of significance.

5. <u>(Discussed) TI 2515/154, Spent Fuel Material Control and Accounting at Nuclear Power Plants</u>

a. <u>Inspection Scope</u>

The inspectors completed Phase I and Phase II of TI 2515/154, Spent Fuel Material Control and Accounting at Nuclear Power Plants.

b. Findings

No findings of significance were identified.

6. (Discussed) TI 2515/156, Offsite Power System Operational Readiness

a. Inspection Scope

The inspectors collected data from licensee maintenance records, event reports, corrective action documents and procedures and through interviews of station engineering, maintenance, and operations staff, as required by TI 2515/156. The data was gathered to assess the operational readiness of the offsite power systems in accordance with NRC requirements such as Appendix A to 10 CFR Part 50, General Design Criterion 17; Criterion XVI of Appendix B to 10 CFR Part 50, TS for offsite power systems; 10 CFR 50.63; 10 CFR 50.65(a)(4), and licensee procedures. Documents reviewed for this TI are listed in the Attachment.

b. Findings

No findings of significance were identified. Based on the inspection, no immediate operability issues were identified. In accordance with TI 2515/156 reporting requirements, the inspectors provided the required data to the NRC Headquarters staff for further analysis. This TI will remain open pending further completion of that analysis.

4OA6 Meetings, Including Exit

On July 6, 2004, the resident inspectors presented the inspection results to Mr. William Kitchens and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee Identified Violations

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

- TS 5.4.1.a requires written procedures be established covering the activities listed in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, which includes procedures for performing maintenance activities on safety-related systems. Contrary to the above, procedure 27731-C, 480 Volt Switchgear Cubicle/Transformer Maintenance, Revision 25.1, was inadequate in that it did not address actions to be taken when operating a cell switch during Unit 2 AB05 switchgear cleaning. The cycling of the cell switch resulted in isolating air flow to the two Unit 1 radioactive gas monitors rendering them inoperable. The finding was determined to be of very low safety significance because it represented a degradation of only the radiological barrier function provided for the control room. It did not degrade the barrier function of the control room against smoke or toxic atmosphere. This issue was identified in the licensee's corrective action program as CR 2004001633.

- TS 5.7 requires special controls for LHRA (i.e., radiation areas > 1R/hr at 30 cm). Contrary to this, on October 2, 2003, two workers encountered an unposted and uncontrolled LHRA inside the U1 bioshield and received dose rate alarms on their electronic dosimeters. The workers responded appropriately and left the area, however, a follow-up survey by HP showed a maximum dose rate of 2 R/hr at 30 cm near a cavity drain line valve. This finding is of very low safety significance because there was no substantial potential for overexposure and the licensee's ability to assess dose was not compromised. This issue has been entered in the licensee's corrective action program as CR 2003002792.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

W. Bargeron, Plant Support Assistant General Manager
W. Burmeister, Manager Engineering Support
D. Carter, Superintendent Chemistry
J. Dixon, Superintendent Health Physics
S. Douglas, Manager Operations
K. Holmes, Manager Training and Emergency Preparedness
W. Kitchens, Nuclear Plant General Manager
I. Kochery, Health Physics & Chemistry Manager
T. Tynan, Assistant General Manager Operations

Other licensee employees contacted included office, operations, engineering, maintenance, chemistry/radiation, and corporate personnel.

NRC personnel:

B. Bonser, Chief, Region II Reactor Project Branch 2
W. Rodgers, Senior Risk Analyst, Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000424, 425/2004004-01	NCV	Inadequate NSCW Operating Procedures (Section 1R16)
05000424, 425/2004004-03	FIN	Failure to Implement Adequate Administrative Control Over Keys to VHRAs (Section 2OS1)
05000424/2004004-04	FIN	Inadequate Feedpump Turbine Control Valve Work Instruction (Section 4OA3.1)
05000425/2004004-05	NCV	Failure to Follow Unit Operating Procedure (Section 4OA3.3)
05000424, 425/2004004-06	FIN	Failure to Perform Timely and Appropriate Operability Evaluation of AFW Valve Degradation (Section 4OA5.1)

Opened

05000425/2004004-02	AV	Failure to Adequately Perform Containment Closeout Inspection (Section 1R20)

Closed

05000424/2004001-00	LER	Manual Reactor Trip Following Loss of Main Feedwater Pump Speed Control (Section 4OA3.1)
05000424, 425/2004002-00	LER	Closure of Control Room Air Damper Results in Technical Specification Non-Compliance (Section 4OA3.2)
05000425/2004001-00	LER	Auxiliary Feedwater System Actuation While Breaking Condenser Vacuum (Section 4OA3.3)
05000424, 425/2003005-01	URI	Review Results of Licensee Auxiliary Feedwater Discharge Control Valve Inspections for Previously Identified Pilot Plug Assembly Concerns (Section 4OA5.1)
2515/152 (Docket 50-425)	TI	Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02) (Section 4OA5.3)
2515/153 (Docket 50-425)	TI	Reactor Containment Sump Blockage (NRC Bulletin 2003-01) (Section 4OA5.4)

Discussed

2515/150 (Docket 50-425)	TI	Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009) (Section 4OA5.2)
2515/154	TI	Spent Fuel Material Control and Accounting at Nuclear Power Plants (Section 4OA5.5)
2515/156	TI	Offsite Power System Operational Readiness (Section 4OA5.6)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures

11115-2, Containment Spray System Alignment
13115-2, Containment Spray System
11610-1, Auxiliary Feedwater System Alignment
13610-1, Auxiliary Feedwater System
11105-2, Safety Injection System Alignment
13105-2 Safety Injection System

Section 1R05: Fire Protection

Procedures

92820-1, Zone 120 - Control Building - Level 2 Fire Fighting Preplan
92704-2, Zone 4 - Auxiliary Building - Level D, Containment Spray Pump A Fire Fighting Preplan
92856-1, Zone 156 - Auxiliary Feedwater Pumphouse Fire Fighting Preplan
92857A-1, Zone 157A - Auxiliary Feedwater Pumphouse Train C Fire Fighting Preplan
92732-2, Zone 32 - Auxiliary Building - Level B, SI Pump, Train A Fire Fighting Preplan
92721-2, Zone 21 - Auxiliary Building - CVCS NCP Room Fire Fighting Preplan
92720-2, Zone 20 - Auxiliary Building - CVCS Pump Room Train A Fire Fighting Preplan
92791-2, Zone 91 - Control Building - Level A Fire Fighting Preplan
92792-2, Zone 92 - Control Building - Level A Fire Fighting Preplan
92860B-1, Zone 160 - NSCW Pumphouse - Train B Fire Fighting Preplan
92737-1, Zone 37 - Auxiliary Building Level A CCW Pumps Train B Fire Fighting Preplan

Section 1R07: Heat Sink Performance

Procedures

23830-C, Process Instrument Loop Channel Calibration, Rev. 17.1
35363-C, Chemistry Control of the NSCW System, Rev. 1
83305-C, Heat Exchanger Testing/Maintenance Program, Rev. 7.1
83306-C, CCW and ACCW Heat Exchanger Testing, Rev. 7
83308-C, Testing of Safety-Related NSCW System Coolers, Rev. 29
83309-C, Safety-Related Heat Exchanger Inspection, Rev. 6

Condition Reports

2001000543, 2002003548, 2002003582, 2003001497, 2003002595, 2004000388, 2004000410, 2004000810, 2004001458, 20103194

Completed Procedures

83309-C, Attachment 1, RHR Motor Cooler 2A & 2B Inspection, completed 04/18/01
83309-C, Attachment 1, ESF Chiller 2A & 2B Condenser Section Inspection, completed 10/16/02 and 10/22/02
83309-C, Attachment 1, ACCW 2A & 2B Inspection, completed 02/06/95 & 06/10/03

Completed Surveillances Task Sheets

116695, 54 Month NSCW Flow Test of Train A Auxiliary Component Cooling Water Heat Exchanger per Procedure 83306-C, completed 04/08/01

116696, 54 Month NSCW Flow Test of Train B Auxiliary Component Cooling Water Heat Exchanger per Procedure 83306-C, completed 04/08/01
128059, 54 Month Temperature Monitoring of Train A RHR Pump Motor Coolers per Procedure 83308-C, completed 11/02/02
128060, 54 Month Temperature Monitoring of Train B RHR Pump Motor Coolers per Procedure 83308-C, completed 11/02/02
140165, Quarterly A-Train Nuclear Service Cooling Water Check Valve Inservice Test per Procedure 14830-2, completed 11/09/03

Completed MWOs
10203101, 10300652, 20103136, 20103194

Miscellaneous
REA (Request for Engineering Assistance) 02-VAA068, NSCW Tower Recommended Replacement Fill, 02/28/03

ED (Equivalency Determination) SC-03-0-0005, Resolite Tred Safe Panels for Use as Replacement Fill Material in the NSCW Towers, 12/03/03

RHR Pump Motor Coolers 2-1205-P6-001M01 and 2-1205-P6-002-M01 NSCW Flow Trends, 1995-2004

Commitments CO0019128, CO0031141, CO0031352, and CO0034688

System Health Report, Nuclear Service Cooling Water System 1202, 4th Quarter 2002 - 4th Quarter 2003

Section 1R17: Permanent Plant Modifications
Procedures
13540-2, Safety Features Sequencer System
T-ENG-04-03, DCP 97-V2N0062 Functional Test Procedure for 2A Sequencer

Maintenance Work Order
20300993, DCP 97062 Installation

Condition Reports
2004002015, During testing 2A Sequencer failed to perform a load shed as designed

Section 1R20: Refueling and Outage Activities
Procedures
00254-C, Foreign Material Exclusion and Plant Housekeeping Programs
00309-C, Control of Unattended Temporary Material in Containment in Modes 1-4
11899-2, RCS Draindown Configuration Checklist
12000-C, Post Refueling Operations (Mode 6 to Mode 5)
12001-C, Unit Heatup to Hot Shutdown (Mode 5 to Mode 4)
12002-C, Unit Heatup to Normal Operating Temperature and Pressure
12003-C, Reactor Startup (Mode 3 to Mode 2)
12004-C, Power Operations (Mode 1)
12005-C, Reactor Shutdown to Hot Standby (Mode 2 to Mode 3)
12006-C, Unit Cooldown to Cold Shutdown
12007-C, Refueling Operations (Entry into Mode 6)
12009-C, RCS Vacuum Refill
13005-2, Reactor Coolant System and Refueling Cavity Draining
14210-2, Containment Building Penetrations Verification - Refueling

14406-2, Boron Injection Flow Path Verification - Shutdown
14900-C, Containment Exit Inspection
18004-C, Reactor Coolant System Leakage
18019-C, Loss of Residual Heat Removal
18030-C, Loss of Spent Fuel Pool Level or Cooling
27504-C, Equipment Hatch Emergency Closure
23985-2, RCS Temporary Water Level System
29540-C, Risk Assessment Monitoring
29542-C, Shutdown Risk Management
93300-C, Conduct of Refueling Operations
93663-C, Verification of Core Loading Pattern

Section 2OS1: Access Control To Radiologically Significant Areas

Procedures, Guidance Documents, and Manuals
00008-C, Plant Lock and Key Control, Rev. 13.1
00930-C, Radiation and Contamination Control, Rev. 22
43000-C, Radiation and Contamination Surveys, Rev. 17.1
43005-C, Establishing and Posting Radiation Controlled Areas and High Radiation Area Access Control, Rev. 24
43007C, Issuance, Use, and Control of Radiation Work permits, Rev. 20
43014-C, Special Radiological Controls, Rev. 30
43300-C, Personnel Decontamination, Rev. 23
46017-C, Control and Monitoring of Materials in Radiation Controlled Areas, Rev. 28
NMP-GM-002-GL02, Corrective Action Program Details and Expectations Guideline, Ver. 2.0.

Records and Data
RWP No. 04-2601, CM/PM VLVS Motors X-mitters, Misc Activities U2 AB/FHB, Rev. 0
RWP No. 04-2302, Eddy Current Testing on U2 S/Gs # 2 & 3, Rev. 1
RWP No. 04-2306, S/G Primary Side Tube Work S/G 2 & 3, Rev. 1
RWP No. 04-0109, Work and Surveillances in HRAs, CAs, and airborne areas, Rev. 0
RWP No. 04-2403, Decon of the Upper and Lower Rx Cavity
Radiological Survey No. 65410, CTMT Spray Encapsulated Vessel (2FHBC2), 4/22/04
Radiological Survey No. 64433, Valve Gallery CVCS Mixed Bed Demin (2AXA72), 3/29/04
Radiological Survey No. 65570, U2 Containment 210' Elevation, 4/24/04
Radiological Survey No. 65211, U2 Undervessel, 4/20/04
Radiological Survey No. 66190, U2 S/G Manway Platform #2, 5/3/04
Radiological Survey No. 65774, U2 S/G 2 Cold Leg, 4/27/04
Personnel Contamination Event Log and WBC records, 4/20/04 - 5/2/04

Audits and Condition Reports
Audit No. OP02-03/11, QA Audit of Health Physics and Radiation Protection, 9/15/03
CR Nos. 2003002522, 2003002792, 2003003617, 2004001876, and 2003001924

Section 2OS2: As Low As Reasonably Achievable (ALARA)

Procedures, Instructions, and Guidance Documents
00920-C, Radiation Exposure Limits and Administrative Guidelines, Rev. 14
00930-C, Radiation and Contamination Control, Rev. 22
00910-C, VEGP ALARA Program, Rev. 13.2
41001-C, ALARA Job Review, Rev. 15.041006-C, Temporary Shielding, Rev. 18.0
Vogtle Electric Generating Plant 2003 Annual ALARA Report 3/29/04
Vogtle Electric Generating Plant 1R11 Refueling Outage ALARA Report September 28-October 23, 2003
Vogtle Electric Generating Plant Major Issues Status Report April 09, 2004 Edition

Records, Worksheets, and Surveys
Radiological Survey No. 18361 Under Vessel U1
Radiological Survey No. 65211 Under Vessel U2
Radiological Survey No. 29939 Under Vessel U2
Radiological Survey No. 65772 S/G 2 Channel Head Cold Leg U2
Radiological Survey No. 65773 S/G 2 Channel Head Hot Leg U2
Radiological Survey No. 65776 S/G 3 Channel Head Cold Leg U2
Radiological Survey No. 65775 S/G 3 Channel Head Hot Leg U2

Audits and Condition Reports
CR Nos. 2004001854, 2004001927, and 2004000045
Dose Reduction Response Report, November 17, 2003
Dose Reduction Self Assessment Report, November 17, 2003
QA Audit of Health Physics and Radiation Protection, OPO2-03/11, September 15, 2003

Section 2PS2: Radioactive Material Processing and Transportation
Procedures, Instructions, Guidance Documents
Process Control Program, Rev. 8
46004-C, Shipment of Radioactive Material, Rev. 17.1
46023-C, Dry Active and Wet Waste Sorting and Segregation, Rev. 3.0
46100-C, 10 CFR 61 Waste Classification Sampling Program, Rev. 4.1
46104-C, Shipment of Radwaste to a Licensed Waste Processor, Rev. 7.0
46106-C, Waste Classification Resin Shipments, Rev. 5.1
Drawing AX4DB124-2 Rev. 5 P&I Diagram Radwaste Processing Facility
Drawing AX4DB124-3 Rev. 4 P&I Diagram Radwaste Processing Facility
Drawing AX4DB124-4 Rev. 3 P&I Diagram Radwaste Processing Facility
Drawing AX4DB124-5 Rev. 1 P&I Diagram Radwaste Processing Facility

Records, Worksheets, and Surveys
Shipment No. 04-001 DAW/RESIN Date Shipped 01/16/04
Shipment No. 04-002 DAW/RESIN Date Shipped 01/23/04
Shipment No. 04-003 DAW Date Shipped 01/30/04
Shipment No. 04-004 DAW/FILTERS Date Shipped 02/02/04
Shipment No. 04-005 DAW/FILTERS Date Shipped 02/09/04
Shipment No. 04-006 DAW/RESIN Date Shipped 02/20/04
Shipment No. 04-018 RESIN Date Shipped 05/07/04

Condition Reports
2003003293, 2003003673, 2004000335, and 2004000503

Section 4OA1: Performance Indicator Verification
Records
Radiological Survey No. 58708, Inside U1 Bioshield Area, 9/29/03
Electronic Dosimeter Alarm Logs, 5/3/03 - 5/3/04
2003 Annual Effluent Release Report
Gaseous Release Permit No. 40105.025.038.G, 5/1/04
Liquid Release Permit No. 40064.003.018.L, 5/3/04
OOS Effluent Monitor Compensatory Action Records, September 2003 - April 2004
Monthly PI Reports, September 2003 - February 2004

Condition Reports
2003002712, 2003002966, 2003002805, 2004001934, 2004000061, and 2004000229

<u>Section 4OA5: Other</u>

<u>Procedures, Guidance Documents and Miscellaneous documents</u>:
18017-C, Abnormal Grid Disturbances/Loss of Grid
19100-C, ECA - 0.0, Loss of All AC Power
13419-C, Diesel Generator Extended AOT
00354-C, Maintenance Scheduling
14230-1/2, A.C. Source Verification
VEGP - UFSAR Chapter 8, Electric Power
Vogtle Electric Generating Plant-Scoping Manual - Offsite Power and High-Voltage Switchyard
Power Quality Guide AX3BB02-00001 for Vogtle Electric Generating Plant
Southern Company NERC Control Area Readiness Audit, March 23-25, 2004
LER 050-425/90-006-00, Loss of Offsite Power Leads to Site Area Emergency