UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

January 28, 2005

Carolina Power and Light Company
ATTN: Mr. James Scarola
Vice President - Harris Plant
Shearon Harris Nuclear Power Plant
P. O. Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000400/2004006

Dear Mr. Scarola:

On December 31, 2004, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Shearon Harris reactor facility. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 6, 2005 with you 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.

Based on the results of this inspection, two NRC-identified and two self-revealing findings of very low safety significance (Green) were identified. Three of these findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations (NCVs), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. In addition, one licensee-identified violation which was determined to be of very low safety significance is listed in Section 40A7 of the enclosed report. If you deny any NCV in the enclosed report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report to the 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 the Shearon Harris facility.

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) components 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/

Paul E. Fredrickson, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Docket No.: 50-400
License No.: NPF-63

Enclosure: NRC Inspection Report 05000400/2004006
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-400
License No:	NPF-63
Report No:	05000400/2004006
Licensee:	Carolina Power and Light Company
Facility:	Shearon Harris Nuclear Power Plant, Unit 1
Location:	5413 Shearon Harris Road New Hill, NC 27562
Dates:	September 26, 2004 - December 31, 2004
Inspectors:	R. Musser, Senior Resident Inspector C. Welch, Acting Senior Resident Inspector P. O'Bryan, Resident Inspector G. MacDonald, Senior Project Engineer, (Section 1R12) W. Loo, Senior Health Physicist, (Sections 2OS1, 2OS2, 2PS2, 4OA1) H. Gepford, Health Physicist, (Sections 2OS1, 2OS2) J. Fuller, Reactor Inspector, (Section 1R08, 4OA5)
Approved by:	P. Fredrickson, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000400/2004-06, 09/26/2004 - 12/31/2004; Shearon Harris Nuclear Power Plant, Unit 1; Refueling and Outage Activities, Other Activities.

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

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A self-revealing non-cited violation of Technical Specification 6.8.1 was identified for a failure to adequately implement maintenance procedures for electrical maintenance on the 1A-SA switchgear during a refueling outage. During relay calibration, technicians failed to adequately tape the relay leads in order to prevent the leads from short-circuiting and causing the loss of the IA-SA vital bus and ‘A’ residual heat removal pump. The ‘A’ residual heat removal pump was in service, providing core cooling, at the time the bus was lost.

  This finding is more than minor because it affected the Mitigating Systems Cornerstone safety function of core decay heat removal and increased the likelihood that a loss of decay heat removal would occur due to the loss of power to the 1A-SA bus. NRC Inspection Manual Chapter 0609, Appendix G was used to evaluate this finding. Phase 2 and 3 analyses determined that this finding is of very low safety significance (Green) because decay heat removal was only temporarily interrupted, power to the 1A-SA bus was restored automatically by the 1A EDG, the ‘A’ RHR train was restarted promptly (four minutes), and the ‘B’ RHR train was continuously available for decay heat removal if it was needed. The finding was also related to the cross-cutting area of human performance because the performance deficiency was identified as the failure of maintenance personnel to adequately tape the lifted leads. (Section 4OA5.3)

- Green. The inspectors identified a finding involving the management of maintenance activities during a refueling outage resulting in an unnecessary increase in risk of losing the decay heat removal key safety function. Work conducted during the refueling outage unnecessarily increased the risk of a loss of core shutdown cooling by conducting intrusive electrical maintenance on the 1A-SA vital electrical bus while the ‘A’ residual heat removal pump was being used for core cooling. Concurrently, the reactor coolant system was depressurized, rendering steam generators unavailable for natural circulation cooling and time to core boiling was relatively low.

This finding is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of configuration control of shutdown equipment used to mitigate the consequences of accidents, and the objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Therefore, the issue was assessed using the Significance Determination Process (SDP). NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Safety SDP" figure 1 and checklist 3 were applicable for the phase 1 evaluation of this issue. Although the finding increased the likelihood that a loss of decay heat removal would occur, the event's significance is bounded by the Phase 2 and 3 evaluations performed in response to NCV 05000400/2004006-02. The evaluation determined that the interruption of decay heat removal event was of very low safety significance (Green). Therefore, this finding, which involved increasing the likelihood of an interruption, is also of very low safety significance (Green). (Section 4OA5.4)

- Green. A self-revealing non-cited violation of Technical Specification 6.8.1 was identified for failure to adequately implement work control procedures, leading to draining approximately 1250 gallons of contaminated water from the refueling water storage tank and the volume control tank to the reactor auxiliary building during a refueling outage. The water drained due to an inadequate boundary clearance hung for work on a chemical and volume control system valve.

  This finding is more than minor because, if left uncorrected, it would become a more significant safety concern due to the potential to damage plant equipment, and drain the refueling water storage tank. The finding was associated with the configuration control attribute of the Mitigating Systems Cornerstone. NRC Inspection Manual Chapter 0609, Appendix G was used to evaluate this finding. The finding did not require quantitative analysis and was of very low safety significance because it did not affect the ability of the licensee to maintain shutdown event mitigation capability. The finding was also related to the cross-cutting area of human performance because failure to adhere to both a valve isolation and a clearance boundary procedure contributed to initiating the draindown. (Section 1R20.3)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, for failure to promptly identify and correct a condition adverse to quality. The condition adverse to quality was the presence of flow paths in the top of the containment recirculation sump structures which bypassed the containment sumps' screens and had the potential to adversely impact emergency core cooling system (ECCS) performance during containment recirculation.

This finding is more than minor because it affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events (loss-of-coolant-accident) to prevent undesirable consequences (core damage). The finding was associated with the design control attribute of the cornerstone. NRC Inspection Manual Chapter 0609, Appendix A was used to evaluate this finding. The finding is considered to be of very low safety significance (Green) because the bypass flow paths did not result in a loss of safety function. This finding was also related to the cross-cutting area of problem identification and resolution because the condition of the sumps had not been properly identified and corrected by the licensee during previous containment walkdowns. (Section 4OA5.2)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Shearon Harris began the inspection period at full rated thermal power. Operation at or near rated power continued until October 15, when a shutdown for Refueling Outage 12 commenced. Refueling activities (Mode 6) occurred from October 19 - 31. On October 18, electrical power to the A safety-related electrical switchgear was lost and caused a 4 minute interruption in shutdown cooling (1R14). An NRC Special Inspection was conducted in response to the event. The inspection and associated findings are documented in Inspection Report 05000400/2004009.

On November 7, while attempting to raise power and place the main turbine on-line, control room operators responded to a lowering level in C steam generator (S/G), and established manual control of the C S/G level using motor driven auxiliary feedwater (MDAFW) pump A. The unit entered cold shutdown on November 9 to repair the C S/G main feedwater isolation valve. Inspection identified that one of the C feedwater isolation valve's seats was extensively damaged, the stem sheared, and the disc wedged between the seats. Further inspections of A and B feedwater isolation valves revealed similar seat damage.

After repair to the feedwater isolation valves, power operations (Mode 1) were established November 17, with operation at full rated thermal power reached on November 19, 2004.

The unit continued to operate at rated thermal power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

After the licensee completed preparations for seasonal low temperature, the inspectors walked down the cold weather protection for the emergency service water and the chemical and volume control systems including the refueling water storage tank, the reactor makeup tank, the emergency service water pump structure, and the service water intake structure. These systems were selected because their safety-related functions could be affected by adverse weather. The inspectors reviewed documents listed in the Attachment, observed plant conditions, and evaluated those conditions using criteria documented in Procedure AP-301, "Adverse Weather."

The inspectors reviewed AR #111396, "1A and 1B air compressor cold weather protection" to verify that the licensee identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns:

The inspectors performed four partial system walkdowns to verify the selected train or system was properly aligned and capable of performing its specified safety function. The proper alignment was verified by comparing the "as found" position of critical valves, breakers, and control switches to the required position specified in the procedures and drawings listed in the Attachment. The walkdowns were performed at a time of increased risk, either the redundant train was out-of-service for maintenance or the train was just re-aligned following maintenance. The spent fuel pool cooling (SFP) system, for the A and B pools, was walked down following the full core off-load due to the increased risk associated with a loss of SFP cooling. A sample of manual containment isolation valves inside containment were walked down to verify the valves had been properly repositioned and the pipe caps reinstalled following local leak rate testing.

- Turbine driven auxiliary feedwater, following maintenance on October 6.
- Spent fuel pool (SFP) cooling for the A and B pools, on October 24, following full core off-load.
- Manual containment isolation valves associated with test connection points for local leak rate testing, on November 6.
- The high head safety injection flow paths and cold leg accumulators, on November 6, following restoration from outage activities.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

For the eight areas identified below, the inspectors reviewed the licensee's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures, to verify that those items were consistent with Final Safety Analysis Report (FSAR) Section 9.5.1, Fire Protection System, and FSAR Appendix 9.5.A, Fire Hazards Analysis. The inspectors walked down accessible portions of each area and reviewed results from related surveillance tests, to verify that conditions in these areas were consistent with descriptions of the applicable FSAR sections. Documents reviewed are listed in the Attachment.

- Switchgear ventilation room B (1-A-5-HVB)
- Switchgear room B (1-A-SWGRB)
- 221' level inside containment (1-C-1-CHFB, 1-C-1-CHFA, 1-C-1-RCP-1A, 1-C-1-RCP-1B, 1-C-1-RCP-1C, and 1-C-1-BAL)
- 236' level inside containment (1-C-1-RCP-1A, 1-C-1-RCP-1B, 1-C-1-RCP-1C, and 1-C-1-BAL)
- 261' level inside containment (1-C-3-EPA, 1-C-3-EPB, 1-C-1-BAL, 1-C-1-RCP-1A, 1-C-1-RCP-1B, 1-C-1-RCP-1C, and 1-C-1-BAL)
- 236' Fuel Handling Building (5-F-2-FPV1)
- 236' Fuel Handling Building (5-F-2-FPC)
- 236' Fuel Handling Building (5-F-2-FPV2)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

Internal Flooding

The inspectors walked down the turbine building, which contains risk-significant structures, systems, or components (SSCs) which are below flood levels or otherwise susceptible to flooding from postulated pipe breaks, to verify that the area configuration, features, and equipment functions were consistent with the descriptions and assumptions used in FSAR Section 3.6A.6, Flooding Analysis, and in the supporting basis documents listed in the Attachment. The inspectors reviewed the operator actions credited in the analysis, to verify that the desired results could be achieved using the plant procedures listed in the Attachment.

The inspectors reviewed the Action Request (AR) #23331, "FME in 240' Turbine Bldg Sump" to verify that the licensee identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

a. Inspection Scope

ISI Activities

The inspectors observed the calibration of non-destructive examination (NDE) test equipment, reviewed examination results for in-process ISI work activities, reviewed ISI procedures, and reviewed selected ISI records since the last outage, associated with risk significant structures, systems, and components. This was the first outage of the third period of the second ISI interval. The inspection activities, documentation, and supporting records were compared to the requirements specified in the Technical Specifications (TS) and the ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, with no addenda, to verify compliance and to ensure that examination results were appropriately evaluated and dispositioned.

Specifically, NDE activities were reviewed as follows:

Direct Observation

Calibration of Ultrasonic Test (UT) equipment for UT Examination: II-SI-017RC-SW-A3, 6" Elbow to Pipe weld on Safety Injection Line
Calibration of Ultrasonic Test (UT) equipment for UT Examination: II-SI-017RC-SW-A4, 6" Pipe to Elbow weld on Safety Injection Line
Bare Metal Visual (BMV) Inspection - 2 (VT-2): II-PZR-01NSEW-15, PZR BMV
Bare Metal Visual (BMV) Inspection - 2 (VT-2): II-PZR-01NSEW-16, PZR Spray Nozzle to Safe End
Bare Metal Visual (BMV) Inspection - 2 (VT-2): II-PZR-01NSEW-17, PZR Safety Nozzle to Safe End
Bare Metal Visual (BMV) Inspection - 2 (VT-2): II-PZR-01NSEW-18, PZR Safety Nozzle to Safe End
Bare Metal Visual (BMV) Inspection - 2 (VT-2): II-PZR-01NSEW-19, PZR Safety Nozzle to Safe End
Bare Metal Visual (BMV) Inspection - 2 (VT-2): II-PZR-01NSEW-20, PZR Relief Nozzle to Safe End
Liquid Penetrant (PT) Examination: II-PZR-01NSEW-19, PZR Safety Nozzle to Safe End

Record Review

UT: II-SI-017RC-SW-A3, 6" Elbow - Pipe weld on Safety Injection Line
UT: II-SI-017RC-SW-A4, 6" Pipe - Elbow weld on Safety Injection Line
UT: II-BIT-01NTHW-03, Boron Injection Tank Nozzle
UT: II-BIT-01NTHW-04, Boron Injection Tank Nozzle
PT: II-BIT-01NTHW-03, Boron Injection Tank Nozzle
PT: II-BIT-01NTHW-04, Boron Injection Tank Nozzle
MT: 3SW-14-46SB-1, 14" SW Line Preparation for Freeze Seal

VT-2: II-RV-001RVNOZA1-N-01SE, Pipe to Nozzle weld
VT-2: II-RV-001RVNOZAO-N-06SE, Nozzle to Pipe weld
VT-2: II-RV-001RVNOZB1-N-03SE, Pipe to Nozzle weld
VT-2: II-RV-001RVNOZBO-N-02SE, Nozzle to Pipe weld
VT-2: II-RV-001RVNOZC1-N-05SE, Pipe to Nozzle weld
VT-2: II-RV-001RVNOZCO-N-04SE, Nozzle to Pipe weld
VT-1: II-SI-024SI-81 VBB(1-16), Bolted Connection
VT-1: II-SI-027SI-135 VBB(1-16), Bolted Connection

Qualification and certification records for examiners, inspection equipment, and consumables along with the applicable NDE procedures for the above ISI examination activities were reviewed. In addition, samples of ISI issues in the licensee's corrective action program were reviewed for adequacy.

The inspectors reviewed the following recorded indications to ensure that they were dispositioned in accordance with ASME Code requirements:

Bare Metal Visual (BMV) Inspection - 2 (VT-2): II-PZR-01NSEW-19, PZR Safety Nozzle to Safe End
MT: 3SW-14-46SB-1, 14" SW Line Preparation for Freeze Seal

The inspectors reviewed the "Shearon Harris Nuclear Power Plant Unit 1 Inservice Inspection Summary Report" dated August 13, 2003, which states that there were no rejectable indications during the last outage. The inspectors reviewed the NIS-1 Form (report of inservice inspections), and the NIS-2 Forms (report of repairs and replacements) for compliance to ASME Code requirements.

The inspectors reviewed Work Order Packages 625527, 626840, and 625526 for the code repairs of through-wall leaks on the AH-2 and AH-3 Essential Service Water (ESW) headers. The inspectors reviewed repair weld data sheets, the welding procedure specification (WPS), supporting welding procedure qualification records (PQR), material certifications, associated corrective action documentation, and examination results for these weld repairs of ASME Class 2 piping materials.

Boric Acid Corrosion Control (BACC) Inspection

The inspectors reviewed implementation of the licensee's BACC program to verify that commitments made in response to Generic Letter 88-05 and Bulletin 2002-01 were being effectively implemented. The inspectors reviewed the records for a sample of BACC walkdown visual examination activities, to verify that the examiners were adequately identifying and documenting boric acid leakage throughout the plant. The inspectors review ensured that the inspection scope of the BACCP included locations where BA could cause degradation to safety-related components. The inspectors also reviewed a sample of engineering evaluations and associated corrective action documents to evaluate the engineering bases for conclusions regarding apparent cause and severity of discovered leaks, and justification for corrective actions. The engineering evaluations were from the steam generator outage during May 2004.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

Just-in-Time Outage Training

The inspectors observed just-in-time (JIT) training on October 4-5. Observations were made of both operating crews and reactor engineering personnel. Training comprised both classroom and simulator instruction and covered major plant evolutions, including: shutdown, establishing shutdown cooling, solid plant operations, startup, and reactor physics testing. The training also covered industry and plant specific operating experience and made use of data from prior evolutions. The purpose of the inspection was to assess command and control, communications, crew dynamics and the integration of the reactor engineers, as well as to observe the plant simulator's response for various evolutions.

Quarterly Operations Training

On December 7, the inspectors observed licensed-operator performance during licensed operator simulator training for crew D, to verify that operator performance was consistent with expected operator performance, as described in Exercise Guide EOP-SIM-17.82. This training tested the operators' ability to respond to a small break and large break loss of coolant accidents, including high containment pressure. The inspectors focused on clarity and formality of communication, the use of procedures, alarm response, control board manipulations, group dynamics and supervisory oversight. The inspectors observed the post-exercise critique to verify that the licensee had identified deficiencies and discrepancies that occurred during the simulator training. The inspectors reviewed AR #141274, "Unauthorized operator aid in place at the main control board" to verify that the licensee identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the three degraded SSC/function performance problems or conditions listed below to verify the licensee's handling of these performance problems or conditions in accordance with 10CFR50, Appendix B, Criterion XVI, Corrective Action, and 10CFR50.65, Maintenance Rule. Documents reviewed are listed in the Attachment.

- Reactor Vessel Level Indication System (RVLIS) Failures
- Safety Related Battery Charger Trips
- Spent Fuel Pool Cooling Pump Trips On Thermal Overload

The inspectors focused on the following attributes:

- Appropriate work practices,
- Identifying and addressing common cause failures,
- Scoping in accordance with 10 CFR 50.65(b),
- Characterizing reliability issues (performance),
- Charging unavailability (performance),
- Trending key parameters (condition monitoring),
- 10 CFR 50.65(a)(1) or (a)(2) classification and reclassification, and
- Appropriateness of performance criteria for SSCs/functions classified (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified (a)(1).

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- 044487 Breaker 1&4 A33-SA-2C Thermal Overload Trip After Long Run
- 136104 MR Repetitive Functional Failure
- 133645 2&3A SFP Tripped on Overload
- 132679 2&3B Spent Fuel Pump Tripped On Overload
- 135334 RVLIS Train A Failed
- 054982 Safety Battery Charger High Voltage Trip
- 134180 Battery Charger 1B-SB Tripped While In Operation
- 134196 Clarify Technical Specification Applicability
- 130512 1A-SA Battery Charger Failed To Load When Placed In Service
- 105539 System 5230 Functional Failures

b. <u>Findings</u>

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. <u>Inspection Scope</u>

The inspectors reviewed the licensee’s risk assessments, and the risk management actions for the plant configurations associated with the five activities listed below. The inspectors verified that the licensee performed adequate risk assessments, and implemented appropriate risk management actions when required by 10CFR50.65(a)(4). For emergent work, the inspectors also verified that any increase in risk was promptly assessed, and that appropriate risk management actions were promptly implemented. For shutdown activities, the six shutdown safety functions (decay heat removal, electrical power, inventory control, reactivity control, pressure control, and containment) were reviewed and verified based on walkdown of the main control boards, discussions with control room operators, and review of the Key Safety Function Availability Checklists.

- Planned maintenance on the B emergency diesel generator, on September 29, that resulted in an elevated Yellow risk condition.
- Testing on October 17, per OST-1823, that involved the A emergency diesel generator (EDG), the EDG load sequencer, and various emergency core cooling system components.
- A deviation to the key safety function availability checklist, on October 17, was verified acceptable and in accordance with OMP-003, "Outage Shutdown Risk Management." Additionally, crane operations in the vicinity of the refueling water storage tank (RWST) were observed and the high-risk evolution contingency plan for heavy lifts over the RWST were verified.
- Planned testing per OST-1813; on October 19, with a time to boil of less than 30 minutes; of the remote shutdown system that resulted in an elevated orange risk condition.
- Emergent work to inspect the check valve and main feedwater isolation valve for the C S/G on November 10.

The inspectors reviewed AR#143023, "Main Feed Isolation Valve Damage" to verify that the licensee identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events

Loss of Shutdown Cooling

a. Inspection Scope

The inspectors reviewed the control room operators' performance following a loss of shutdown cooling on October 18, 2004. Maintenance activities on a degraded voltage relay caused an inadvertent loss of power to the A train safety-related switchgear and the operating residual heat removal (RHR) pump. The purpose of the inspection was to determine if the operators' response was appropriate to the event and in accordance with procedures and training. To assess the operators' performance, the inspectors reviewed the operator logs, plant computer data, and abnormal operating procedures AOP-025, "Loss of One Emergency AC Bus (6.9KV) or One Emergency DC Bus (125V);" and AOP-020, "Loss of RCS Inventory or Residual Heat Removal While Shutdown." Shutdown cooling restoration was reviewed by a walkdown of the main control board, and review of control board indications and plant computer information. The inspectors also observed control room activities to restore the A train 480 volt safety-related switchgear and the start of a second train of RHR.

b. Findings

No findings of significance were identified related to the operator's response to the event. An NRC Special Inspection for the event is documented in inspection report 05000400/2004009.

Non-routine Plant Evolutions

a. Inspection Scope

The inspectors observed portions of the following non-routine plant evolutions to assess the control room operators' performance during the evolution. The inspection focused on command and control, communications, use of peer and self-checking techniques, procedure adherence, and adherence to technical specification cooldown and heatup requirements.

- GP-006, Normal Plant Shutdown From Power Operation to Hot Standby (Mode 1 to Mode 3).
- GP-007, Normal Plant Cooldown Mode 3 to Mode 5.
- GP-004, Reactor Startup
- GP-005, Power Operation
- EST-923, Initial Criticality and Low Power Physics Testing

b. Findings

No findings of significance were identified.

Feedwater System Malfunction

a. Inspection Scope

The inspectors reviewed the control room operators' response for a lowering level in the C steam generator during power ascension on November 7, 2004. The inspectors reviewed operator logs, plant computer data, and Abnormal Operating Procedure AOP-10, "Feedwater System Malfunctions." The inspectors walked down the main control boards and observed indicated system alignments and plant parameters. The inspectors observed control room activities to restore and control C steam generator level. The purpose of the inspection was to determine if the operators' response was appropriate to the event and in accordance with procedures and training.

The inspectors reviewed AR#143023, "Main Feed Isolation Valve Damage" to verify that the licensee identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed four operability determinations addressed in the ARs listed below. The inspectors assessed the accuracy of the evaluations, the use and control of any necessary compensatory measures, and compliance with the TS. The inspectors verified that the operability determinations were made as specified by Procedure AP-618, "Operability Determinations." The inspectors compared the justifications made in the determination to the requirements from the TS, the FSAR, and associated design-basis documents, to verify that operability was properly justified and the subject component or system remained available, such that no unrecognized increase in risk occurred:

- #142315, “13 entry points for FME into recirculation points”
- #143023, “AOP entry due to C steam generator level decrease”
- #140248, “Boraflex Degradation of PWR Fuel Storage Racks, Rev. 1”
- #132130, “Boraflex Degradation of BWR Fuel Storage Racks, Rev. 2”

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed the cumulative effects of the operator workarounds listed below, to verify that those effects could not increase an initiating event frequency, affect multiple mitigating systems, or affect the ability of operators to respond in a correct and timely manner to plant transients and accidents.

- emergency service water pump seal flow degraded causing control room alarm
- waste processing building chiller vanes in manual
- gross failed fuel detector sample flow not in required range
- normal containment purge trips frequently due to weather changes

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the modification described in Engineering Change 56215, “Control Room Isolation Signal (CRIS) Actuation Logic Changes”, to verify that:

- the modification did not degrade the design bases, licensing bases, and performance capabilities of risk significant SSCs,
- implementing the modification did not place the plant in an unsafe condition, and
- the design, implementation, and testing of the modification satisfied the requirements of 10CFR50, Appendix B.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

For the eight maintenance activities listed below, the inspectors reviewed associated work documents and witnessed the post-maintenance test and/or reviewed the test data, to verify the test and its results adequately demonstrated the operability and functional capability of the effected SSC. The inspectors verified the procedure’s acceptance criteria was consistent with the applicable licensing basis and/or design-basis documents.

- W.O. 00608642, replace B EDG overspeed governor assembly.
- W.O. 00208776, 1SI-327 valve replacement.
- W.O. 00524675, pedestal bearing adjustments and generator alignment on 1B-SB EDG.
- W.O. 00633011, corrective maintenance on valve 1CS-9.
- W.O. 00614577, 1CP-10 valve seat replacement.
- W.O. 00614586, 1CP-4 valve seat replacement.
- W.O. 00491580, replace B ESW pump motor.
- W.O. 00397638, remove, inspect, install 1SW-118.

The inspectors reviewed AR#141462, “Unsat Results For —7 LLRT (1CS-7, 1CS-8, 1CS-9), to verify that the licensee identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

.1 Review of Outage Plan

a. Inspection Scope

Prior to the outage, the inspectors reviewed the outage risk plan and verified the licensee had considered risk, industry operating experience, and prior site-specific problems. The inspectors reviewed Procedure OMP-003, “Outage Shutdown Risk Management,” the R12 Pre-Outage Risk Assessment, the level 2 outage schedule, and the key safety functions availability checklists. The inspectors verified the outage plan appropriately considered the six shutdown critical safety functions (decay heat removal, electric power availability, inventory control, reactivity control, pressure control, and containment).

b. Findings

No findings of significance were identified.

.2 Monitoring of Shutdown Activities

h. Inspection Scope

The inspectors observed operators perform portions of the procedures listed below to shutdown the reactor and establish shutdown cooling. Adherence to TS cooldown limits (TS 3.4.9) was verified based on review of Procedure OMM-13, Cycle and Transient Program Monitoring, data and cooldown plots maintained by the shift technical advisor. The inspectors performed a thorough walkdown of containment and discussed the results of the boric acid walkdown with the program engineer.

- GP-006, Normal Plant Shutdown From Power Operation to Hot Standby (Mode 1 to Mode 3).
- GP-007, Normal Plant Cooldown Mode 3 to Mode 5.
- OST-1036, Shutdown Margin Calculation.
- OPT-1519, Containment Visual Inspection for Boron and Evaluation of Containment Sump Inleakage.
- OMM-013, Cycle and Transient Program Monitoring.

b. Findings

No findings of significance were identified.

.3 Licensee Control of Outage Activities

a. Inspection Scope

The inspectors observed the items or activities described below to verify that the licensee maintained defense-in-depth commensurate with the outage risk control plan for key safety functions and applicable TS when taking equipment out of service. The inspectors reviewed licensee responses to emergent work and unexpected conditions to verify that resulting configuration changes were controlled in accordance with the outage risk control plan, and to verify that control room operators were kept cognizant of the plant configuration.

- clearance activities
- RCS instrumentation
- electrical power
- decay heat removal
- spent fuel pool cooling
- inventory control
- reactivity control
- containment closure

The inspectors performed periodic walkdowns of the main control board and plant equipment and observed the associated instrumentation to verify systems required for decay heat removal were properly aligned and operating.

b. Findings

Introduction. A Green self-revealing non-cited violation (NCV) was identified for the failure to implement maintenance procedures in accordance with TS 6.8.1. Adequate isolation was not provided for the disassembly of chemical and volume control system (CVCS) valve 1CS-243, which resulted in a discharge of approximately 1250 gallons into the reactor auxiliary building.

Description. On October 22, 2004, clearance order 74919 was hung to perform maintenance on valve 1CS-243. To support the maintenance, valve 1CS-240 located upstream of 1CS-243 was closed from the main control board and danger tagged in the closed position. Contrary to Procedure OPS-NGGC-1301 two valve isolation was not established. During the tagging process, the operator questioned whether valve 1CS-240 was fully closed. The operator engaged the valve's local manual operator and verified that the valve was fully closed and then hung the danger tag. Contrary to procedure OP-107, "Chemical and Volume Control System," the 1CS-240 valve handwheel was not locked in position after the manual operator was engaged to prevent the valve from drifting open under system pressure. The tendency of the valve to drift open under system pressure when closed using the manual operator was a known condition.

On October 25, the A charging/safety injection pump (CSIP) was started for post maintenance test. The increased system pressure caused valve 1CS-240 to drift open. When 1CS-240 opened, a flow path from the CVCS to the reactor auxiliary building environment was established through valve 1CS-243 which was disassembled. As a result, 250 gallons of contaminated water was discharged from the volume control tank and approximately 1000 gallons drained from the refueling water storage tank before the flow path was isolated. Four low level personnel contaminations occurred during response to the event.

Analysis. The finding was greater than minor because if left uncorrected it would become a more significant safety concern due to the finding's potential to injure personnel, damage plant equipment, and drain the RWST. The finding was associated with the configuration control attribute of the Mitigating Systems Cornerstone and was assessed per MC-0609 Appendix G for a shutdown plant. The finding was of very low safety significance (Green). Review of Appendix G Table 1 and Checklist 4 of Attachment 1, identified the finding did not require a quantitative assessment and therefore screened to Green, per Figure 1. A quantitative assessment was not required because the finding did not cause a loss of thermal margin, a loss of inventory, or degrade the ability to add inventory to the reactor coolant system. The finding did not degrade the ability to add inventory to the reactor coolant system because the refueling water storage tank retained sufficient water to fulfill it's safety function. The finding was also related to the cross-cutting area of human performance because failure to adhere to both a valve isolation and a clearance boundary procedure contributed to initiating the draindown.

Enforcement. Technical Specification 6.8.1, requires in part that procedures be written and implemented for the activities specified in Appendix A to Regulatory Guide 1.33. Contrary to the above, licensee personnel did not implement procedural requirements when establishing clearance 74919 for maintenance on valve 1CS-243. Specifically, two valve isolation was not provided and valve 1CS-240 was not locked shut following manual closure of the valve to prevent stem movement as required per OP-107, "Chemical and Volume Control System." Because the finding is of very low safety significance and has been entered into the licensee's corrective action program (AR 141473), this violation is being treated as an NCV, consistent with section VI.A of the NRC Enforcement Policy: NCV 05000400/2004006-01, Failure to Provide an Adequate Isolation for the Disassembly of CVCS Valve 1CS-243.

.4 Refueling Activities

a. Inspection Scope

The inspectors observed fuel handling operations (removal, inspection, and insertion) and other ongoing activities, to verify that those operations and activities were being performed in accordance with TS and approved procedures. The inspectors reviewed fuel locations to verify that the locations of the fuel assemblies were tracked, including new fuel, from core offload through core reload. The cycle 13 core configuration was verified using the following techniques: observation from the refueling bridge of a sample of fuel assembly placements in the reactor, review of video records for a sample of fuel assemblies, and review of completed procedure FHP-10, "Core Loading Verification." The results were compared to the core design specified in proprietary report HNP-F/NFSA-0120; Harris Nuclear Plant Unit 1, Cycle 13 Startup and Operations Report.

b. Findings

No findings of significance were identified.

.5 Monitoring of Heatup and Startup Activities

a. Inspection Scope

On a sampling basis prior to changing modes or plant configurations, the inspectors reviewed system lineups and/or control board indications to verify that TS, license conditions, and other requirements, commitments, and administrative procedure prerequisites for the mode or configuration change were met.

Prior to startup, the inspectors inspected the internals of the containment sumps and walked down containment to verify debris had not been left which could affect performance of the sumps and emergency core cooling systems. The inspectors also walked down piping systems in containment during the operating pressure test (EST-201, "ASME System Pressure Test,") to identify potential RCS leakage and reviewed the licensee's own inspection results. Acceptable shutdown margin was verified based on review of Procedure OST-1036, "Shutdown Margin Calculations Modes 1-5," and the Core Operating Limits Report (COLR). Containment integrity was verified based on review of EST-212 Attachment 53, "Combined Type B and Type C Leak Rate Evaluation," and inspection of a sample of containment penetrations and containment isolation valves.

The inspectors observed portions of the initial startup and physics testing. The physics testing results were reviewed and discussed with the reactor engineer to verify that core operating limit parameters were consistent with design and within technical specification requirements.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

Periodically, the inspectors reviewed the items that had been entered into the licensee's corrective action program, to verify that the licensee had identified problems related to outage activities at an appropriate threshold and had entered them into the corrective action program.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the eight surveillance tests identified below, the inspectors witnessed testing and/or reviewed test data, to verify that the systems, structures, and components involved in these tests satisfied the requirements described in the TS and the FSAR, and that the tests demonstrated that the SSCs were capable of performing their intended safety functions.

- OST 1813, “Remote Shutdown System Operability 18-month Interval Modes 5, 6, Defueled.”
- EPT-328, “Safety Injection to Cold Leg/Hot Leg Valves 1SI-3, 1SI-4, 1SI-52, 1SI-86, and 1SI-107 Generic Letter 89-10 MOV Test.”
- EST-221, “Type B Local Leak Rate Test,” performed on the containment equipment hatch following closure for RFO-12.
- EST-923, “Initial Criticality and Low Power Physics Testing”
- EST-724, “Shutdown and Control Rod Drop Test Using Computer”
- EST-813, “Control Rod Drive Mechanism Timing Test Using Computer”
- OST-1112, “Rod Position Indication Test 18 Month Interval Modes 3-5"
- *OST-1076, “Auxiliary Feedwater Pump 1B-SB Operability Test Quarterly Interval Modes 1-4"

*This procedure included inservice testing requirements.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modifications described in the Engineering Changes (ECs) and work orders listed below to verify that the modifications did not affect the safety functions of important safety systems, and to verify that the modifications satisfied the requirements of 10CFR50, Appendix B, Criterion III, Design Control.

- Work Order 428231-02, temporary power to the 1A-SB battery charger.
- Engineering Change (EC) 59439, 1SC-33 valve internals removed.
- Disabling of the main feedwater pump low flow trip (EC 54754)
- Disabling of the condensate booster pump low flow trip (EC54754)

The inspectors reviewed AR#135129, “Failure of ‘A’ Emergency Service Water Screen Wash System” to verify that the licensee identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Controls To Radiologically Significant Areas

a. Inspection Scope

Access Controls The inspectors evaluated licensee guidance for access controls to radiologically significant areas. Selected procedural details for posting, surveying, and access controls to airborne radioactivity, radiation, high radiation area (HRA), locked high radiation area (LHRA) and very high radiation area (VHRA) locations were reviewed and discussed with cognizant licensee representatives. The inspectors evaluated seven radiation work permits (RWPs) used for work in radiologically significant areas associated with refueling outage 12 (RFO-12). The selected RWPs were evaluated for adequacy of access controls and specified electronic dosimeter (ED) alarm set-points against expected work area dose rates and work conditions. Access control procedures for posted LHRA and VHRA locations were reviewed and discussed with selected Radiation Protection (RP) management, supervision, and technicians.

During facility tours, the inspectors evaluated selected radiological postings, barricades, and surveys associated with radioactive material storage areas and radiologically significant areas within the reactor containment building, reactor auxiliary building, waste processing building, and fuel handling building. The inspectors conducted independent dose rate measurements at various building locations, including the spent fuel pool, and compared those results to licensee radiation survey map data. In addition, the inspectors independently assessed implementation of LHRA controls. The inspectors reviewed licensee implementation of inspections to verify the condition of the locked doors and assessed LHRA and VHRA key controls.

During the inspection, the proficiency and knowledge of the radiation workers and RP staff in communicating and applying radiological controls for selected tasks were evaluated. The inspectors attended briefings for work activities associated with RWP 2011 - Reactor Head Lift, RWP 2012 (Task 10) - Upper Internals Set, and RWP 2012 (Task 14) - Control Rod Drive Pull and Replace. Radiological worker and RP technician training/skill levels, procedural adherence, and implementation of RWP specified access controls, including those associated with changing radiological conditions, were observed and evaluated by the inspectors during selected job site reviews and tours within the licensee’s radiological control area. In addition, the inspectors interviewed selected management personnel regarding radiological controls associated with RFO-12 activities.

RP activities were evaluated against FSAR Section 12, Radiation Protection; 10 Code of Federal Regulations (CFR) 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J; and approved procedures. The procedures and records reviewed are listed in the Attachment.

Problem Identification and Resolution Licensee Corrective Action Program (CAP) Nuclear Condition Report (NCR) documents associated with access controls to radiologically significant areas, radiation worker performance, and RP technician proficiency were reviewed and assessed. Ten ARs documented in the Attachment were reviewed and evaluated in detail during inspection of this program area. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with CAP-Nuclear Generation Group Common (NGGC) Procedure - 0200, Corrective Action Program, Revision (Rev.) 7.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA) Implementation of the licensee's ALARA program during the RFO-12 outage was observed and evaluated by the inspectors. The inspectors reviewed ALARA planning, dose estimates, and prescribed ALARA controls for the five outage work tasks expected to incur the maximum collective exposures. Reviewed activities included installation and removal of temporary lead shielding, installation and removal of insulation, decontamination activities, reactor headwork/refueling activities, and incore instrument work. Also, incorporation of planning, established work controls, expected dose rates and dose expenditure into the ALARA pre-job briefings and RWPs for those activities were reviewed. The inspectors also independently verified that selected job site dose rates were consistent with the dose rates recorded on pre-job survey maps for selected containment and auxiliary building work areas and equipment. The inspectors directly observed performance of the reactor head lift, the upper internals lift, setting of the upper internals, and pulling thimbles in the seal table room, while evaluating the licensee's use of engineering controls, low dose waiting areas, and on-the-job supervision.

Selected elements of the licensee's source term reduction and control program were examined to evaluate the effectiveness of the program in supporting implementation of the ALARA program goals. Reviewed areas included primary chemistry shutdown controls, radiation field monitoring and trending, and temporary shielding.

Trends in individual and collective personnel exposures at the facility were reviewed. Records of year-to-date individual radiation exposures sorted by work groups were examined for significant variations of exposures among workers. Exposure tracking during the RFO-12 outage, and records of exposures to declared pregnant workers incurred from May 2003 through October 2004 as well as associated guidance for controlling such exposures, were also reviewed. Trends in the plant's three-year rolling average collective exposure history, outage, non-outage and total annual doses for 1992 through 2003 were reviewed and discussed with licensee representatives.

The licensee's ALARA program implementation and practices were evaluated for consistency with FSAR Chapter 12, Sections 1-5, Radiation Protection; 10 CFR Part 20 requirements; Regulatory Guide 8.29, Instruction Concerning Risks from Occupational Radiation Exposure, February 1996; and licensee procedures. Documents reviewed during the inspection of this program area are listed in the Attachment.

Problem Identification and Resolution The inspectors reviewed NCR documents and audits listed in the Attachment that are related to the ALARA program. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with CAP-NGGC-0200, Corrective Action Program, Rev. 7.

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 licensee methods for processing and characterizing radioactive waste (radwaste). Inspection activities included direct observation of processing equipment for solid and liquid radwaste and evaluation of waste stream characterization data.

Solid and liquid radwaste equipment was inspected for material condition, configuration compliance with the FSAR, and consistency with Process Control Program (PCP) requirements. Inspected equipment included liquid radwaste hold-up tanks; resin transfer piping; abandoned waste evaporators; remote operating equipment for packaging filters, and elements of the Modular Fluidized Transfer Demineralization System. The inspectors discussed system changes, component function, and equipment operability with licensee staff. In addition, procedural guidance for resin transfer was evaluated and compared with current equipment configuration. Reviewed documents are listed in the Attachment.

Licensee radionuclide characterizations for selected waste streams were reviewed and discussed with radwaste staff. For primary resin, radwaste filters, and dry active waste (DAW) the inspectors evaluated analyses for hard-to-detect nuclides and appropriate use of scaling factors. Comparison results between licensee waste stream characterization data and outside laboratory data were reviewed for the period May 2003 - September 2004. For selected shipment records, waste classification calculations were independently performed and the methodology used for resin waste stream mixing and concentration averaging was evaluated. The inspectors also interviewed radwaste staff and reviewed procedural guidance to evaluate the licensee's program for monitoring changing operational parameters.

Radwaste processing activities were reviewed for consistency with the licensee 's PCP, Rev. 8; and UFSAR, Chapter 11, Amendment 52. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 61.55 and guidance provided in the Branch Technical Position on Waste Classification and Waste Form, 1983.

Transportation The inspectors evaluated the licensee's activities related to transportation of radioactive material. The evaluation included direct observation of shipment preparation activities and review of shipping related documents.

The inspectors directly observed transportation activities including the shipment of two containers of DAW, a pressurizer safety relief valve; and receipt of two IF-300 spent fuel casks and snubbers. The inspectors observed placarding of the shipment vehicles and marking and labeling of the shipment packages. For the shipments, the inspectors also reviewed Department of Transportation (DOT) 7A Type A conformance documentation and evaluated whether the receiving licensee was authorized to accept the package. During the IF-300 evolutions, the inspectors evaluated technician proficiency in conducting selected dose rate measurements and obtaining contamination smears. The inspectors also reviewed survey results for selected IF-300 shipments from May 2003 - October, 2004.

As part of the document review, the inspectors evaluated five shipping records for consistency with licensee procedures and compliance with NRC and DOT regulations. The inspectors also reviewed the licensee's procedure for opening, closing, and handling the IF-300 series spent fuel shipping cask and compared the procedure to recommended vendor protocols and certificate of compliance requirements. In addition, training records for five individuals currently qualified to ship radioactive material were checked for completeness and the training curriculum provided to these workers was evaluated. Documents reviewed during the inspection are listed in the Attachment.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Parts 20 and 71, 49 CFR Parts 170-189; as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172 Subpart H.

Problem Identification and Resolution Selected NCR documents associated with radwaste processing and transportation were reviewed and discussed with cognizant licensee representatives. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure CAP-NGGC-0200, Corrective Action Program, Rev. 7. Reviewed documents are listed in Section 2PS2 of the report Attachment.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program

a. Inspection Scope

The inspectors followed up on an event that was an issue of agency-wide concern regarding an uncontrolled release of radioactive materials from the RCA.

b. Findings

No findings of significance were identified; however, a minor violation was identified for failure to control and maintain constant surveillance of licensed material that is not in storage, as prescribed by 10 CFR 20.1802. This self-revealing violation occurred when a worker, upon in-processing at another facility after working at Harris, was determined to have a radioactive particle of approximately 21 nCi on the tongue of his shoe. Because the general quantities reported were near or below the detection capabilities of the personnel monitors used to screen workers exiting the radiologically controlled areas, no performance deficiency occurred. The severity of the violation was screened using Supplement IV of the Enforcement Policy which states that a violation involving an isolated failure to secure, or maintain surveillance over, licensed material in an aggregate quantity that does not exceed 10 times the quantity specified in Appendix C to Part 20 is a minor violation. Since this issue was determined to be a violation of minor significance, it is not subject to enforcement action in accordance with Section IV of the Enforcement Policy. This minor violation is being documented because the uncontrolled release of radioactive material to the public domain is an issue of agency-wide concern.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

To verify the accuracy of the PI data reported during the specified time periods, the inspectors compared the licensee's basis in reporting each data element to the PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Rev. 2.

Mitigating Systems Cornerstone

- Safety System Unavailability, Emergency AC Power
- Safety System Unavailability, Auxiliary Feedwater

For the previous four quarters (fourth quarter of 2003 through the third quarter of 2004), the inspectors reviewed operators logs, records of inoperable equipment, and Maintenance Rule records, to verify that the licensee had adequately accounted for unavailability hours that the subject systems had experienced. The inspectors also reviewed the number of hours those systems were required to be available and the licensee's basis for identifying unavailability hours. In addition, the inspectors interviewed licensee personnel associated with the PI data collection, evaluation, and distribution.

In IR 05000400/2004005 the Reactor Coolant System Leakage PI was reported as being verified from January 1, 2004 through June 30, 2004. This was an error. The Safety System Functional Failure PI was the PI that should have been reported as being verified for this period.

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- #109914, "OMM-016 Guidance Improvement For Tracking EDG Unavailability"
- #114721, "Autolog/NRC KPI Discrepancies"

Occupational Radiation Safety (OS) Cornerstone

The inspectors reviewed Occupational Exposure Control Effectiveness PI data collected from January through September 2004 to evaluate the Occupational Radiation Safety Cornerstone. For the reviewed period, the inspectors assessed CAP records to determine whether HRA, VHRA, or unintended radiation exposures, resulting in TS or 10 CFR 20 non-conformances, had occurred during the review period. In addition, the inspectors reviewed selected personnel contamination event data, internal dose assessment results, and ED alarms associated with dose rates exceeding 1 rem/hr and cumulative dose rates exceeding established set-points from May 2003 through September 2004. Reviewed documents relative to this PI are listed in the Attachment.

Public Radiation Safety (PS) Cornerstone

The inspectors reviewed the Radiological Effluent Technical Specification (RETS) / Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences PI data from May 2003 through September 2004 to evaluate the Public Radiation Safety Cornerstone. For the review period, the inspectors reviewed data reported to the NRC and NCR documents listed in the Attachment. In addition, the inspectors reviewed out-of-service effluent monitor logs and seven effluent release permits.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Annual Sample Review

a. Inspection Scope:

The inspectors reviewed the corrective actions for the degraded performance of the B emergency service water (ESW) pump due to the high safety significance of the ESW system. The degraded condition, captured in AR 00112525 and 00141842, resulted from failure of a component (wear ring) internal to the pump. Inspection focused on the adequacy of the root cause determination and the corrective actions, whether the timeliness to identify and correct the issue was commensurate with its safety significance and ease of discovery, and the extent of condition relative to the installed ESW pumps. The ability to monitor the pumps' performance for signs of degradation and the operating history of the A pump were considered when assessing corrective action timeliness. The inspector also reviewed the ARs' disposition for reportability/operability.

b. Findings:

No findings of significance were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspector's review was focused on repetitive equipment issues, but also considered the results of inspector CAP item screenings, licensee trending efforts, and licensee human performance results. The inspector's review nominally considered the six-month period of July through December 2004, although some examples expanded beyond those dates when the scope of the trend warranted. The review also included issues documented outside the normal CAP in system health reports, self assessment reports, and Maintenance Rule assessments. The specific items reviewed are listed in the Attachment. The inspectors compared and contrasted their results with the results contained in the licensees latest semi-annual trend reports.

The inspectors also evaluated the licensee's trend reports against the requirements of the CAP as specified in CAP-NGGC-0200, Corrective Action Program.

b. Findings and Observations

There were no findings of significance identified. The inspectors observed that the licensee performed adequate trending reviews. The licensee routinely reviewed cause codes, involved organizations, key words, and system links to identify potential trends in the CAP data. The inspectors compared the licensee process results with the results of the inspectors' daily screening and did not identify any discrepancies or potential trends in the CAP data that the licensee had failed to identify.

The inspectors, as well as the licensee, identified a negative trend with regard to component configuration control during the fourth quarter of 2004. This adverse trend was documented in AR 144390. During the fourth quarter, the licensee initiated thirteen action requests with configuration control as the potential cause. The most significant of these action requests was AR 140449, which described a loss of the 1A-SA emergency bus with a brief interruption of shutdown cooling. This event was reviewed in detail and is discussed in NRC Inspection Report Number 05000400/2004009. At the end of the inspection period, the licensee had initiated an apparent cause investigation to determine the cause and establish corrective actions for this negative trend. The inspectors considered the licensee initiatives with respect to this matter appropriate.

.3 Problem Identification and Resolution Cross-Cutting Aspects

The finding described in Section 4OA5.2.b of this report regarding unidentified bypass flowpaths in the containment recirculation sumps has as its primary cause problem identification and resolution in that the condition of the sumps had not been properly identified and corrected by the licensee during previous containment walkdowns.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 0500400/2004-003-00, "Automatic Reactor Trip - Turbine Trip Caused by Failure of a Circuit Card in the Rod Control System"

On May 6, 2004, the reactor automatically tripped from 100% power due to a power range flux rate signal. The flux rate signal was generated in response to the four control rods assemblies in Shutdown Bank C inserting into the core due to a loss of stationary gripper coil current. Stationary gripper coil current was lost due to a failure of a transistor on the stationary gripper regulation card for the Shutdown Bank C in the Rod Control System. The licensee replaced the failed card and determined that the root cause was a random failure of the transistor. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the failed equipment in AR #126304. This LER is closed.

.2 (Closed) LER 0500400/2004-002-00, "Inoperability of Containment Fan Cooler (AH-1)," and subsequent retraction of the LER.

On March 3, 2004, licensee personnel discovered the discharge gravity damper for containment fan AH-1A partially open with the fan out of service. With this damper partially open, a recirculation path for fan AH-1B was created, reducing the cooling flow from fan AH-1B. The licensee submitted an LER pending further investigation and analysis of the operability of the system. The licensee later retracted the original LER because they determined that the containment fan cooler was degraded but not inoperable since it was capable of performing its specified safety functions. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the degraded equipment in AR #118991. This LER is closed.

4OA4 Cross Cutting Aspects of Findings

The finding described in Section 1R20.3 of this report regarding the inadequate boundary for maintenance on a chemical and volume control system valve has as its primary cause human performance because failure to adhere to both a valve isolation and a clearance boundary procedure contributed to initiating the draindown.

The finding described in Section 4OA5.3 of this report regarding the inadequate taping of electrical leads had as its primary cause human performance in that the leads were improperly taped by maintenance technicians.

4OA5 Other Activities

.1 Closed - Temporary Instruction 2515/152, "Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02)."

a. Inspection Scope

The inspectors reviewed Progress Energy's activities in response to Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," as required by Temporary Instruction 2515/152, "Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02)." The inspection included review of station procedures, examiner qualification records, and the licensee's 90-day response to NRC Bulletin 2003-002. The inspectors independently reviewed videos and digital photographs of the Spring 2003 (RFO-11) lower head inspection, discussed observed staining of the lower head with the responsible engineers, and independently inspected the lower head during the Fall 2004 (RFO-12) inspection.

The following input addresses the specific reporting requirements of TI 2515/152:

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

   a) Performed by qualified and knowledgeable personnel?

   Yes. The examiners were certified to level II and/or III. The inspector reviewed the qualification records for the 2003 and 2004 examiners and verified that their qualifications were current and included VT-2 examinations. Harris provided 4 hours of training to the examiners specifically for the 2003 reactor head inspections. The training covered boric acid corrosion, discussed industry events and operating experience, utilized EPRI Report 1006296 concerning boric acid corrosion, and also included visual aids demonstrated to resolve selected 0.158 inch tall test chart characters under conditions similar to the exam conditions.

b) Performed in accordance with demonstrated procedures?

The 2003 and 2004 inspections were not performed using a formal procedure specific to inspection of the lower head penetrations. The 2003 inspection used an informal inspection plan along with a map of the lower head penetrations. Additional guidance was available in the very detailed inspection procedure approved for the upper head inspection. The 2004 inspection relied upon existing station procedures for the boric acid control program and VT-2 exams (EGR-NGGC-0207, "Boric Acid Corrosion Control;" and NDEP-0612, "VT-2 Visual Examination of Nuclear Power Plant Components") along with a map of the lower head penetrations. Review of the documents identified that they provided sufficient guidance.

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

Yes.

d) Capable of identifying pressure boundary leakage as described in the bulletin and/or RPV lower head corrosion?

Yes.

2. Could small boric acid deposits representing RCS leakage, as described in the Bulletin 2003-02, be identified and characterized, if present by the visual examination method used?

   Yes.

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

   Each inspection consisted of a direct bare metal visual (BMV) VT-2 exam by qualified examiners. The 2003 inspection was recorded and digital pictures taken of the areas of concern.

4. How complete was the coverage (e.g., 360^{o} around the circumference of all the nozzles)?

   Access to the lower head was good and allowed for complete inspection of each nozzle 360 degrees in circumference, including inspection of the annulus around each penetration. Additional lighting was provide to aid the inspection.

5. What was the physical condition of the RPV lower head (e.g., debris, insulation, dirt, deposits from any source, physical layout, viewing obstructions)? Did it appear that there are any boric acid deposits at the interface between the vessel and the penetrations?

The lower head's material condition appeared good. There were no visual or physical impairments that hindered the inspection or conditions that required corrective action be taken. The inspectors reviewed the video recording and still pictures taken during the 2003 BMV inspection and discussed the inspection results with the responsible engineer. Staining was observable on the lower head and was attributed to an earlier documented cavity seal leak. The noted staining was light and reported as being easily removed with a damp cloth. Where the stain's flow path crossed or bridged over a penetration, the absence of boron crystals within the penetration's annulus was verified by visual inspection. No indication of leakage was identified at any of the penetrations. The inspectors observed the 2004 examination and independently inspected each penetration. No sign of leakage was observed. It was also reported that no new staining was noted on the lower head. No material deficiencies were noted.

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

   None. There were no conditions that required corrective action be taken.

7. What, if any, impediments to effective examinations, for each of the applied non-destructive examination methods, were identified (e.g., insulation, instrumentation, nozzle distortion)?

   None. Additional lighting was provide to aid the inspection.

8. Did the licensee perform appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components above the RPV lower head?

   Yes. Staining was observable on the lower head and was attributed to an earlier documented cavity seal leak that had been previously repaired. Where the stain's flow path crossed or bridged over a penetration, the absence of boron crystals within the penetration's annulus was verified by visual inspection. No indication of leakage was identified at any of the penetrations.

9. Did the licensee take any chemical samples of the deposits?

   No. Chemical analysis was not performed of the residue.

10. Is the licensee planning to do any cleaning of the head?

    No additional cleaning efforts are planned. The noted staining was light and reported as being easily removed with a damp cloth however, the licensee did not remove all of the stains and after being wiped down a light film returned once the surface dried.

11. What are the licensee's conclusions regarding the origin of any deposits present and what is the licensee's rationale for the conclusions?

    No deposits were noted that were attributable to pressure boundary leakage from the lower head penetrations. The observed staining was attributed to an earlier documented cavity seal leak and where the stain's flow path crossed or bridged over a penetration, the absence of boron crystals within the penetration's annulus was verified by visual inspection. The licensee has indicated their intent to perform non-destructive ultrasonic (UT) exams of the lower head penetrations in 2007 outage.

b. Findings

No Findings of significance were identified.

.2 Closed - Temporary Instruction 2515/153, "Reactor Containment Sump Blockage," (NRC Bulletin 2003-01)."

a. Inspection Scope

The inspectors reviewed the licensee's activities in response to Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized Water Reactors," as required by Temporary Instruction 2515/153, "Reactor Containment Sump Blockage (NRC Bulletin 2003-01)." The inspection included review of the licensee's written response to NRC Bulletin 2003-01 (serial #HNP-03-080), the licensee's response to NRC request for additional information (serial #HNP-04-135), station procedures, training records, and physical inspection of the containment recirculation sumps.

The following input addresses the specific reporting requirements of TI 2515/153:

1. For units that entered refueling outages (RFOs) after August 31, 2002, and subsequently returned to power: Was a containment walkdown to quantify potential debris sources conducted by the licensee during the RFO?

   The licensee completed containment walkdowns in accordance with NEI 02-01 during refueling outage (RFO) 11 in the spring of 2003.

2. Did the walkdowns conducted check for gaps in the sumps' screened flowpath and for major obstructions in containment upstream of the sumps?

   Yes, the licensee did check for screen gaps and major obstructions in containment upstream of the sumps and found none. The inspectors also inspected the sumps during RFO-12 in the fall of 2004. No gaps greater than 1/8" were identified in the screens, however the inspectors identified several gaps in the top of the sump structure greater than 1/8". This is discussed in detail in the findings section below.

3. Are any advanced preparations being made at the present time to expedite the performance of sump-related modifications, in case it is found to be necessary after performing the sump evaluation?

   No sump-related modifications are currently planned.

The inspectors verified the following actions were taken by the licensee, as stated in their response to NRC Bulletin 2003-01 and their response to the NRC request for additional information regarding NRC Bulletin 2003-01:

- Emergency Operating Procedures (EOPs) include instructions to monitor containment wide range and recirculation sump level instruments and instructions to secure affected RHR and containment spray pumps if recirculation sump level will not support continued operation. EOPs were revised to include the plant-specific indications of recirculation sump blockage and mitigating actions including refilling of the refueling water storage tank.

- Additional training on generic recirculation sump safety issues, plant-specific indications and mitigating strategies, and EOP changes was conducted for all licensed operators and selected Accident Assessment Team members.

- The simulator was updated to model containment sump screen blockage.

- The containment closeout surveillance procedure and pre-job brief procedure were revised to emphasize containment cleanliness and that flowpaths to the containment recirculation sump are unobstructed.

- Training was provided for maintenance, maintenance shared resources, and selected contractor personnel prior to work inside containment for RFO-12.

- The containment recirculation sump closeout procedure was revised to verify that sump screens are free of gaps and voids greater than 1/8".

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, for failure to promptly identify and correct a condition adverse to quality. Specifically, the licensee failed to promptly identify and correct flow paths which bypassed the containment sumps' screens and had the potential to adversely impact emergency core cooling system (ECCS) performance during containment recirculation.

Description. On October 31, 2004, during inspection of the containment sumps, the inspectors identified: (1) gaps greater than 1/8 inch along the top of the containment sumps where the sumps butted against the containment liner, (2) gaps greater than 1/8 inch around penetrations through the sump covers, (3) debris (e.g. nut, washers, weld rod pieces, dirt/grit, plastic) on top of the sumps and in close proximity to the open areas, and (4) a ½ inch diameter hole in the B sump cover due to a missing fastener.

The inspectors noted that although the debris in and around the sumps demonstrated failure to conform with station procedures concerning containment cleanliness, the debris probably resulted from the current outage (RFO-12) activities. Because the debris was removed prior to entering a mode which required ECCS recirculation sump operability, the issue is a minor violation of TS 6.8.1, implementation of maintenance procedures.

The design of the containment sump screens is to prevent particles greater than 1/8 inch in diameter from passing into the residual heat removal (RHR) and containment spray (CS) suction piping during the recirculation phase of a loss of coolant accident (LOCA). The location and size of these gaps could have allowed particles in excess of 1/8 inch to enter. Particles entering the suction of the RHR and CS systems could adversely affect the safety-related systems' performance during the recirculation phase of a LOCA.

The licensee's past-operability evaluation determined that the bypass flow paths did not cause a loss of safety function for the containment sumps or adversely affect the operability of the emergency core cooling system (ECCS) components. The gaps are believed to have existed since initial construction. Past-operability was primarily based upon engineering judgement and the following factors: (1) the relatively low recirculation flow velocity (< 0.2 ft/sec) which would allow debris to settle out on the floor before reaching the raised pump suction inlets, (2) the time lapse that occurs prior to initiation of sump recirculation, (3) the maximum expected sump water level in containment post accident, (4) the potential debris generation and transport to the sumps during a LOCA, (5) sump design and internal debris transport, and (6) emergency operating procedure (EOP) guidance. The inspectors reviewed the operability determination, the EOPs, and the applicable UFSAR sections to verify that past operability was adequately justified and that potentially affected ECCS components would have remained available and capable of performing their intended safety functions (1R15).

Analysis. The inspectors considered the licensee's failure to identify and take timely corrective action for containment sump screen bypass issues a performance deficiency. Based on previous NRC generic communications and industry operating experience information relative to containment sump issues, the licensee had several opportunities prior to the October 2004 inspection to have identified and corrected the degraded conditions.

The finding was determined to be more than minor because it affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events (LOCA) to prevent undesirable consequences (core damage). The finding was associated with the design control and human performance attributes of the cornerstone. The finding, evaluated per MC-0609, was determined to be of very low safety significance (Green). The finding affected the Long Term Heat Removal attribute of the Phase 1 screening worksheet for Mitigating Systems and is of low safety significance (Green) because it did not result in a loss of safety function and is not risk significant in response to external events (seismic, flood, and severe weather). This finding was also related to the cross-cutting area of problem identification and resolution because the condition of the sumps had not been properly identified and corrected by the licensee during previous containment walkdowns.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, (Corrective Action) requires in part that conditions adverse to quality be promptly identified and corrected. Contrary to this requirement, the licensee failed to promptly identify and correct flow paths which bypassed both containment sumps' screens and had a potential to adversely impact emergency core cooling during sump recirculation. However, because of the very low safety significance and because the issue was entered into the corrective action program (AR 00142315) and the deficient conditions corrected, this finding is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000400/2004006-02, Failure to Promptly Identify and Correct Flow Paths Which Bypassed the Containment Sumps' Screens

.3 (Closed) URI 05000400/2004009-01: Failure to Follow the Procedure for Taping Leads Lifted From Time Delay Relay 2-1/1711.

Introduction. A Green self-revealing NCV was identified for failure to adequately tape electrical leads as required by a maintenance procedure, which caused a loss of power to the 1A-SA electrical bus.

Description. As discussed in Inspection Report 05000400/2004009, an electrician inadequately insulated electrical leads on October 17, 2004 which had been lifted from a time delay relay in a safety-related switchboard. The leads subsequently shorted on October 18, resulting in a loss of offsite power to one safety bus, with a loss of reactor shutdown cooling for four minutes. Step 7.5.2 of Procedure MST-E0045 requires the technician to label (if necessary), lift, and tape the leads from the relay under test. The technicians failed to adequately tape the leads in order to prevent the leads from shorting out and causing the loss of the vital bus, as discussed in Section 03.02 of Inspection Report 05000400/2004009.

Analysis: This issue was unresolved pending the determination of the safety significance. In this inspection period, the issue was determined to be greater than minor because it affected the Mitigating Systems Cornerstone shutdown safety function of decay heat removal and increased the likelihood that a loss of decay heat removal would occur due to the loss of power to the 1A-SA bus. Therefore, the issued was assessed using the Significance Determination Process (SDP). Manual Chapter 0609, Appendix G, "Shutdown Safety SDP" figure 1 and checklist 3 were applicable for the phase 1 evaluation of this issue. Because the issue increased the likelihood of losing decay heat removal, a quantitative evaluation was required. The quantitative, phase 2 and phase 3 determination was conducted using Appendix G of MC 0609, "Shutdown Safety SDP." Since both RHR trains could be easily placed into service prior to exceeding the time of bulk boiling within the reactor vessel (> 40 minutes), an additional recovery credit from the Phase 2 analysis results was warranted. Consequently, Worksheet 9 of MC0609, Appendix G, was altered to include recovery credit. Therefore, the dominant accident sequence was Loss of RHR (set at zero since there was an actual interruption of decay heat removal), failure to recover decay heat removal prior to boiling using the standby train of RHR, (2 points), failure of RCS injection prior to core damage (1 ECCS train was available - 4 points) and failure to recover the operating train of RHR prior to RCS boiling (this train was returned to service in a matter of minutes - 1 point). This resulted in a numerical result of 7 which equates to a finding of very low safety significance (Green). This finding was also related to the cross-cutting area of human performance because the performance deficiency was identified as the failure of maintenance personnel to adequately tape the lifted leads.

Enforcement: TS 6.8.1 requires in part that written procedures be implemented, including procedures for maintenance that can affect the performance of safety-related equipment. Contrary to the above, on October 17, 2004, Procedure MST-E0045 was not implemented, in that the leads which were lifted from time delay relay 2-1/1711 were not adequately taped. However, because of the very low safety significance and because the issue was entered into the corrective action program (AR 140449) and the deficient conditions corrected, this finding is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000400/2004006-03, Failure to Follow the Procedure for Taping Leads Lifted From Time Delay Relay 2-1/1711. URI 05000400/2004009-01 is closed.

.4 (Closed) URI 05000400/2004009-02: Assessment of Increased Plant Risk

Introduction: A Green finding was identified involving the assessment and management of the maintenance activities conducted during the outage.

Description: As discussed in Inspection Report 05000400/2004009, The inspectors noted that, at the time of the loss of the 1A-SA bus on October 18, 2004 the plant conditions were significantly different than they were at the time the degraded voltage time delay relay testing was initially scheduled to be performed. When the testing was originally scheduled, both RHR pumps were operating, and the RCS was pressurized, allowing the possibility of natural circulation for core cooling using the steam generators. However, when the procedure was actually conducted, the RCS had been depressurized, which complicated the use of the steam generators and natural circulation for core cooling had the RHR system failed. Also, the 'A' RHR pump was in service while testing the 'A' train degraded grid voltage relays during procedure MST-E0045, increasing the chance of a loss of power to the 'A' train shutdown cooling RHR pump. The inspectors also noted that the testing procedure, MST-E0045, was scheduled at a time with relatively little time to core boiling.

This issue was unresolved pending the determination of both the safety significance and enforcement. In this inspection period, the inspectors determined that this issue was a performance finding because the licensee unnecessarily increased the risk of losing the decay heat removal key safety function by planning an intrusive high-risk maintenance activity on the 'A' RHR train in conjunction with several avoidable conditions existing at the time of the maintenance. Specifically, the risk of losing the RHR function was increased and the defense-in-depth for maintaining the function reduced because, 1) the RCS was depressurized, rendering steam generators unavailable for decay heat removal, 2) the time to core boiling was relatively short, and 3) the high-risk maintenance was being performed on the same train that powered the running RHR pump. For this issue, the licensee did not effectively employ the shutdown risk management principles from Procedure OMP-003, "Outage Shutdown Risk Management." For instance, Section 5.1.1 states in part that "The most fundamental risk management action is planning and sequencing of maintenance activities taking into account the insights provided by the risk assessment." Section 5.1.5 states in part that "The primary means of enhancing nuclear safety during shutdown operation is to effectively plan and control outage activities." Section 5.2.1 states in part that "Outage scheduling should be developed through interaction with involved organizations and disciplines to assure that the planning provides Defense in Depth throughout the outage." Although Risk Management Procedure OMP-003 is not required by the site license, the requirements above are fundamental risk-management principles. In addition, the inspectors determined, based on discussion with Region II and Office of Nuclear Reactor Regulation staff, that the licensee's risk assessment required by the Maintenance Rule, 10CFR50.65(a)(4), was adequate. The assessment assumed protected train RHR pump availability for loss of the running RHR pump, thus enveloping loss of one pump for any reason, including this event.

Analysis: This finding is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of configuration control of shutdown equipment used to mitigate the consequences of accidents, and the objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Therefore, the issue was assessed using the Significance Determination Process (SDP). Manual Chapter 0609, Appendix G, "Shutdown Safety SDP" figure 1 and checklist 3 were applicable for the phase 1 evaluation of this issue. Although the finding increased the likelihood that a loss of decay heat removal would occur, the event's significance is bounded by the phase 2 and 3 evaluations performed in response to URI 05000400/2004009-01 (see analysis in previous section). The URI 05000400/2004009-01 evaluations determined that the interruption of decay heat removal event was of very low safety significance (Green). Therefore, this finding, identified as FIN 05000400/2004006-04, Unnecessary Increase in Risk of Losing the Decay Heat Removal Key Safety Function, is of very low safety significance (Green).

Enforcement: No violation of regulatory requirements occurred. URI 05000400/2004009-02 is closed.

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

The inspectors reviewed the licensee's 60-day response to NRC Bulletin 2004-01, dated June 27, 2004. The inspectors verified that the licensee's inspections conducted during this outage were consistent with the licensee's response to BL 2004-01.

The inspectors observed the BMV examination performed on all the welds that fall under the scope of BL 2004-01. BMV examinations were observed on the following welds:

- II-PZR-01NSEW-15, PZR BMV
- II-PZR-01NSEW-16, PZR Spray Nozzle to Safe End
- II-PZR-01NSEW-17, PZR Safety Nozzle to Safe End

- II-PZR-01NSEW-18, PZR Safety Nozzle to Safe End
- II-PZR-01NSEW-19, PZR Safety Nozzle to Safe End
- II-PZR-01NSEW-20, PZR Relief Nozzle to Safe End

All 5 welds have Inconel 82/182 buttering and are susceptible to Primary Water Stress Corrosion Cracking (PWSCC). The inspectors verified that these welds were captured in the licensee's ISI program, and determined that a 100% BMV examination will be scheduled every outage.

The inspectors verified that the visual inspections were conducted by personnel qualified to ASME Section XI, VT-2. Visual inspections were conducted in accordance with a licensee qualified procedure. The inspectors reviewed the procedure used for the BMV examination to ensure that it contained specific instructions related to the identification, disposition, and resolution of deficiencies.

The inspectors verified that the physical conditions of the pressurizer nozzle to safe end connections were clean and accessible for the prescribed inspections, and that there were no problems with debris, insulation, dirt, boron from other sources, physical layout, or viewing obstructions which could interfere with the identification of relevant indications.

The inspectors observed that:

- The visual inspections were by direct visual examination.

- Examiners were able to examine 360° around the circumference of all the nozzles.

- Lighting and access was such that small boron deposits, as described in the Bulletin 2004-01, could have been identified and characterized.

- There were no material deficiencies (i.e., cracks, corrosion, etc.) identified that required repair.

- Other than the expected nozzle-to-safe-end geometry, there were no impediments to effective examinations.

- The VT-2 BMV examinations resulted in one small indication of a stain that was located across dissimilar metal weld II-PZR-01NSEW-19. The licensee attempted to obtain a sample for chemical analysis, but was unable to harvest any amount of the stain. The area was visually examined after cleaning and no evidence of cracking or leakage was noted. The licensee performed a liquid penetrant (PT) examination to evaluate the area where the stain was present, and the PT exam did not result in any indications. The inspectors reviewed the examination documentation, observed the BMV examination and the supplemental liquid penetrant exam to verify compliance to BL 2004-01.

.6 Follow-up to Special Inspection Report 05000400/2004009

Subsequent to the issuance of special inspection report 05000400/2004009, inspectors determined, through additional information provided, that as discussed in Section 03.04.c of Inspection Report 05000400/2004009, the outage schedule for refueling outage (RFO) 12 did not preclude the performance of maintenance activity MST-E0045 simultaneously with the performance of surveillance activity OST-1857. Inspectors have determined that the designation of procedure MST-E0045 in the licensee's outage schedule for the maintenance activity in question was a licensee administrative error and that the maintenance activity was also associated with a different procedure (MST-E0075). Inspectors have verified that the licensee accomplished procedure MST-E0075 as the maintenance activity.

Based on the additional information that MST-E0045 was not a precursor for OST-1857, the inspectors noted that the addition of this precursor would have been appropriate because MST-E0045 involved work on the power supply to the 'A' train of RHR while OST-1857 involves manipulation of 'B' train RHR components. These activities were recognized as conflicting activities in the previous refueling outage schedule (RFO-11) and similar maintenance activities were designated as precursors during RFO-12 (e.g. MST-E0075). The licensee acknowledges that not designating MST-E0045 and OST-1857 as conflicting was an error in scheduling.

.7 Institute of Nuclear Power Operations Report Review

The Inspectors reviewed the final Institute of Nuclear Power Operations (INPO) evaluation from their 2004 assessment.

4OA6 Meetings, Including Exit

On January 6, 2005, the resident inspectors presented the inspection results to Mr. Scarola and other members of his staff. The inspectors confirmed that the inspection report did not include proprietary information and no proprietary information was provided or examined during the inspection.

4OA7 Licensee-Identified Violations

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

On October 18, 2004, contrary to the requirements of TS 6.8.1, which requires in part that the licensee establish and implement general operating procedures; control room operators failed to adhere to Procedures GP-007, "Normal Plant Cooldown Mode 3 to Mode 5" and OST-1033, "Daily Surveillance Requirements Daily Interval Mode 5, 6, and Defueled" when they reduced reactor coolant system cold leg temperature below 90F while relying upon the low temperature overpressure protection (LTOP) system for overpressure protection. Credit for LTOP can only be taken when the RCS cold leg temperature is 90F or greater. LTOP was inoperable for approximately 3 hours before the error was identified and temperature restored. The lowest noted temperature was 88.3F.

The finding, assessed using Appendix G of MC-0609, was determined to be of very low safety significance (Green). The finding screened Green using Figure 1. A quantitative assessment was not required because the finding did not cause a loss of thermal margin, a loss of inventory, or result in a non-compliance with the LTOP TS, TS 3.4.9.4. Further mitigating the risk significance was the fact the RCS was vented via the power operated relief valves (PORVs) at the time LTOP was inoperable. Credit for the vent path could not be taken because the PORVs were not gagged to prevent inadvertent closure.

ATTACHMENT: SUPPLEMENTAL INFORMATION

<u>SUPPLEMENTAL INFORMATION</u>

KEY POINTS OF CONTACT

<u>Licensee personnel</u>

M. Blew, ISI Coordinator
D. Braund, Superintendent, Security
J. Briggs, HNP, Superintendent, Environmental and Chemical
D. Corlett, Superintendent, Licensing
F. Diya, Manager - Engineering
R. Duncan, Director - Site Operations
B. Gause, Health Physics Supervisor, Support/Dosimetry
G. Gazda, Health Physics Supervisor, Operations/ALARA
W. Gurganious, Manager - Nuclear Assessment
J. Jankens, Radiation Control Specialist, Spent Fuel
E. McCartney, Training Manager
G. Miller, Maintenance Manager
T. Morton, Manager - Support Services
T. Natale, Manager -Outage and Scheduling
B. Panela, BACC Program Engineer
T. Pilo, Supervisor - Emergency Preparedness
K. Rogers, Radiation Control Specialist, ALARA
J. Scarola, Vice President Harris Plant
G. Simmons, Superintendent - Radiation Control
M. Wallace, Licensing Specialist
B. Waldrep, General Manager Harris Plant
E. Wills, Operations Manager

<u>NRC personnel</u>

P. Fredrickson, Chief, Reactor Projects Branch 4
C. Welch, Acting Senior Resident Inspector
R. Musser, Senior Resident Inspector
P. O'Bryan, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000400/2004006-04	FIN	Unnecessary Increase in Risk of Losing the Decay Heat Removal Key Safety Function (Section 4OA5.4)

Open and Closed

05000400/2004006-01	NCV	Failure to Provide an Adequate Isolation for the Disassembly of CVCS Valve 1CS-243 (Section 1R20.3)
05000400/2004006-02	NCV	Failure to Promptly Identify and Correct Flow Paths Which Bypassed the Containment Sumps' Screens (Section 4OA5.2)
05000400/2004006-03	NCV	Failure to Follow the Procedure for Taping Leads Lifted From Time Delay Relay 2-1/1711 (Section 4OA5.3)

Closed

05000400/2004-003-00	LER	Automatic Reactor Trip -Turbine Trip Caused by Failure of a Circuit Card in the Rod Control System (Section 4OA3.1)
05000400/2004-002-00	LER	Inoperability of Containment Fan Cooler (AH-1) (Section 4OA3.2)
2515/152	TI	Reactor Pressure Vessel Lower Head Penetration Nozzles (Section 4OA5.1)
2515/153	TI	Reactor Containment Sump Blockage (Section 4OA5.2)
2515/160	TI	Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (Section 4OA5.5)

Closed

05000400/2004009-01	URI	Failure to Follow the Procedure for Taping Leads Lifted From Time Delay Relay 2-1/1711 (Section 4OA5.3)
05000400/2004009-02	URI	Assessment of Increased Plant Risk (Section 4OA5.4)

LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather Protection

Procedures:

AP-301, "Adverse Weather Operations"
OP-161.01, "Operations Freeze Protection and Temperature Maintenance Systems,"

1R04 Equipment Alignment

Partial System Walkdown

Fuel Pool Cooling system:
- Procedure OP-116, "Fuel Pool Cooling System," Rev. 29
- Drawing 2165-S-0805, "Simplified Flow Diagram Fuel Pools Cooling Unit 1, Rev. 10.

Turbine Driven Auxiliary Feedwater system:
- Procedure OP-137, "Auxiliary Feedwater System," Rev. 24.
- Drawing 2165-S-0544, "Simplified Flow Diagram Feedwater System Unit 1, Rev. 40.

Safety Injection system:
- Procedure OP-110, "Safety Injection System," Revision 24
- Drawing 2165-S-1308, "Simplified Flow Diagram Safety Injection System, Rev. 12.
- Drawing 2165-S-1309, "Simplified Flow Diagram Safety Injection System, Rev. 17.
- Drawing 2165-S-1310, "Simplified Flow Diagram Safety Injection System, Rev. 10.

Containment Isolation Valves:
- Procedure OST-1069, "Containment Building Penetration Inside Manual Isolation Valve Verification, Rev. 10.

1R05 Fire Protection

Procedures:

- results from FPT-3213, “Fire Detector Functional Test Local Fire Detector Panel 13 12 Month interval”
- results from FPT-3201, “Fire Detector Functional Test Local Fire Detector Panel 1 12 Month Interval”
- results from FPT-3157, “Fire Extinguisher Inspection: Containment Building”
- results from FPT-3154, “Fire Extinguisher Inspection: Fuel Handling Building”
- results from OPT-3107, “Hose Rack Inspection, Containment”

1R06 <u>Flood Protection Measures</u>

FSAR Sections

- 2.4.10, “Flooding Protection Requirements”.
- 3.6A.6, “Flooding Analysis”.

Calculations:

- Ebasco Services Incorporated #SD-4, “Turbine Building Sump Size”
- Ebasco Services Incorporated #SD-1, “Turbine Building Sump Data Sheet”

1R08 Inservice Inspection (ISI) Activities

- EGR-NGGC-0207, Boric Acid Corrosion Control, Rev. 0
- NDEP 0612, VT-2 Visual Examination of Nuclear Power Plant Components, Rev.18
- NDEP-0425, Ultrasonic Examination of Austenitic Pipe Welds (PDI), Rev. 4
- NDEP-0426, Ultrasonic Through-Wall Sizing in Pipe Welds (PDI), Rev. 2
- NDEP-0434, Ultrasonic Examination of Ferritic Welds in Ferritic Pipe, Rev. 3
- NDEP-0436, Manual Ultrasonic Examination of Similar and Dissimilar Metal Piping Welds, Rev. 1
- NGG Program Health Report, Boric Acid Corrosion Control Program, August 30, 2004
- HNP-ISI-002, HNP ISI Program Plan 2nd Interval, Rev. 0
- H-ISI/SBO-03-01, Harris ISI/SBO Assessment, 9/24/2003
- Assessment number: 56304, Weld Inspection and Repair / Replacement (RIR) Activities, 5/23/2002
- Assessment number: 43251, HNP Welding Program, 7/13/2001
- Assessment number: 113075, BACC Program, 3/ 8-11 /04
- Welding Procedure Specification number: 01 2 01, Rev. 1
- Welding Procedure Specification number: 08 2 01, Rev. 1
- Procedure Qualification Record number: 193B, Rev. 2
- Action Request Number: 0012733, Corrective action for active boric acid leaks found during boric acid corrosion control program walkdown
- Action Request Number: 90589, MT in Preparation of Freeze Seal Revealed 6 linear indications
- Action Request (AR) 00127233, Results of BACC walkdown during SG-C forced outage.
- Work Order 00369177, Perform Freeze Seal of Line 3SW14-46SB-1
- HNP-02-063, 60-Day Response to NRC BL 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, 5/15/2002
- HNP-04-097, 60-day response to NRC BL2004-01 For the Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized Water Reactors, 7/2/2004

1R12 Maintenance Effectiveness

Action Requests

044487 Breaker 1&4 A33-SA-2C Thermal Overload Trip After Long Run
136104 MR Repetitive Functional Failure
133645 2&3A SFP Tripped On Overload
132679 2&3B Spent Fuel Pump Tripped On Overload
135334 RVLIS Train A Failed
054982 Safety Battery Charger High Voltage Trip
134180 Battery Charger 1B-SB Tripped While In Operation
134196 Clarify Technical Specification Applicability
130512 1A-SA Battery Charger Failed To Load When Placed In Service
105539 System 5230 Functional Failures
145008 The Thermal Overload Trip Of The 2&3A Fuel Pool Cooling Pump On 8/1/04 Was Not Identified As A Maintenance Rule Functional Failure

Miscellaneous Documents

System Scoping Review for System 5232 125 VDC Electrical Distribution
System Scoping Review for System 1050 Incore Instrumentation
System Scoping Review for System 7110 Spent Fuel Pool Cooling
Maintenance Rule Event Log Report 12/01/2004 (System 5232)
Maintenance Rule Event Log Report 12/01/2004 (System 1050)
Maintenance Rule Event Log Report 12/01/2004 (System 7110)
Engineering Change (EC) Request 4949 Battery Charger High Voltage Trip Annunciation
EC 58480 Spent Fuel Pool Thermal Overloads

Work Orders

598552 1B-SB Battery Charger Tripped While In Service
603859 A Train Of RVLIS And Incore Thermocouples Inoperable
60045903 EC 58480 Replace 2&3B SFP Cooling Pump O/L Heaters
60045901 Replace 2&3A SFP Cooling Pump O/L Htrs per EC 58480

Procedures

MST-C0001 Core Subcooling Margin Monitor Computation Check
MST-I0321 Reactor Vessel Level Monitoring System Microprocessor Calibration Check
MST-E0014 1E Battery Charger Capacity Test
PIC-E050 C&D Battery Charger Relay Card Calibrations
CM-E0001 Station Battery Equalizing Charge
OP-156.01 DC Electrical Distribution

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

- OST 1813, “Remote Shutdown System Operability 18-month Interval Modes 5, 6, Defueled.”
- “Key Safety Function Availability Checklists” for October 17 and 19.
- OMP-003, “Outage Shutdown Risk Management.”
- WCM-001, “On-line Maintenance.”

1R15 Operability Evaluations

- AP-618, “Operability Determinations”

1R17 Permanent Plant Modifications

- Engineering Change Request (EC) #56215
- System Description SD-173, “Control Room HVAC System”
- Design Basis Document DBD-138, “Control Room Heating, Ventilating, and Air Conditioning System.”
- Final Safety Analysis Report Section 9.4, Air-Conditioning, Heating, Cooling, and Ventilation”
- OP-173, “Control Room Area HVAC System.”
- OST-1847, “Safety Injection Actuation Control Room Ventilation Isolation Train A, 18-Month Interval Modes 1-6.”
- OST-1848, “Safety Injection Actuation Control Room Ventilation Isolation Train B, 18-Month Interval Modes 1-6.”
- Drawing 2165-S-1017, “Simplified Flow Diagram HVAC-Containment Building, RAB, and Control Room .”

1R19 Post Maintenance Testing

- OPT-1511, “Emergency Diesel Generator Overspeed Trip Test Modes 1-6.”
- MPT-M0035, “EDG Overspeed Trip Pneumatic Response Time.”
- CM-M0168, “Emergency Diesel Generator Mechanical Overspeed Trip Adjustments/Replacement.
- OST-1824, “1B-SB Emergency Diesel Generator Operability Test 18 Month Interval Modes 1 through 6 and Defueled.”
- EST-212, “Type C Local Leak Rate Tests,” for valve 1CS-9.
- EST-221, “Type C LLRT of Containment Purge Make-up Penetration M-57.”
- EST-220, “Type C LLRT of Containment Purge Exhaust Penetration M-58.”
- OST-1108, “RHR Pump Operability Quarterly Interval Mode 4, 5, and 6.

1R20 Refueling and Outage Activities

- FHP-020, “Refueling Operations”
- FHP-014, “Fuel and Insert Shuffle Sequence.”

Generic Letter 88-17 Documents:

- AOP-020, “Loss of [Reactor Coolant System] Inventory or Residual Heat Removal While Shutdown,” Revision 19
- AP-013, “Plant Nuclear Safety Committee,” Revision 22
- ESR 9500808, “Removable Equipment Hatch Cover Bolting Requirements,” Revision 0
- ESR 9800297, “Containment Closure Procedure,” Revision 0
- OMP-003, “Outage Shutdown Risk Management,” Revision 11
- OST-1034, “Containment Penetrations Test Weekly Interval During Core Alterations and Movement of Irradiated Fuel Inside Containment,” Revision 10 and

- OST-1091, “Containment Closure Test Weekly Interval During Core Alterations and Movement of Irradiated Fuel Inside Containment,” Revision 10

1R22 Surveillance Testing

- OST 1813, “Remote Shutdown System Operability 18-month Interval Modes 5, 6, Defueled.”
- EPT-328, “Safety Injection to Cold Leg/Hot Leg Valves 1SI-3, 1SI-4, 1SI-52, 1SI-86, and 1SI-107 Generic Letter 89-10 MOV Test.”
- OST-1824, “1B-SB Emergency Diesel Generator Operability Test 18 Month Interval Modes 1 through 6 and Defueled.”
- OST-1076, “Auxiliary Feedwater Pump 1B-SB Operability Test Quarterly Interval Modes 1-4"
- EST-221, “Type B Local Leak Rate Test,” performed on the containment equipment hatch following closure for RFO-12.
- EST-923, “Initial Criticality and Low Power Physics Testing”
- EST-724, “Shutdown and Control Rod Drop Test Using Computer”
- EST-813, “Control Rod Drive Mechanism Timing Test Using Computer”
- OST-1112, “Rod Position Indication Test 18 Month Interval Modes 3-5"

1R23 Temporary Plant Modifications

- EM-005, “Temporary Power for Bus Outages.”
- EM-211, “Temporary Power Feed to Class 1E 125V Battery Charger 1A-SB.”
- Work Order #0043060501, temporary power preparation, installation, and removal.
- DBD-128, “Service Water System Traveling Screens and Screen Wash System.”
- FSAR 9.2.1, “Service Water System.”
- CPL-2165-S-0808, “Simplified Flow Diagram, Cooling Tower Blowdown, Make-up, & Intake Structures Screenwash Systems, Unit 1.”
- EC #54754, “Temporarily Defeat MFP, CBP, & HDP Low Flow Trips and CBP High Oil Temperature Trips to Support On-Line Plant Trip Reduction Initiative.”
- FSAR 14.2.12, “Condensate and Feed Systems.”
- DBD-112, “Condensate, Main Feedwater, Condensate Polishers, Feedwater Drains and Vent Systems.”
- SD-134, “Condensate and Feewater.”
- AOP-010, “Feedwater Malfunctions.”
- APP-ALB-016, “Main Control Board” Annunciator Response.
- APP-ALB-019, “Main Control Board” Annunciator Response.

2OS1 Access Controls to Radiologically Significant Areas

Procedures, Manuals, and Guidance Documents

- Administrative Procedure (AP)-110, Pre-Job Briefings, (Revision) Rev. 13
- AP-504, Administrative Controls for Locked and Very High Radiation Areas, Rev. 20
- AP-535, Performing Work in the RCA, Rev. 17
- AP-545, Containment Entries, Rev. 27

- AP-555, Radiography, Rev. 1
- Corrective Action Program (CAP)-Nuclear Generation Group Corporate (NGGC)-0200, Corrective Action Program, Rev. 7
- Dosimetry (DOS)-NGGC-0002, Dosimetry Issuance, Rev. 21
- Environmental and Radiation Control (ERC)-003, Radiation Protection - Conduct of Operations, Rev. 19
- Health Physics Procedure (HPP)-600, Radiation Work Permits, Rev. 19
  HPP-625, Performance of Radiological Surveys, Rev. 19
- HPP-627, Radiological Controls for Diving Operations, Rev. 7
- HPP-800, Handling Radioactive Material, Rev. 41
- Health Physics (HPS)-NGGC-0003, Radiological Posting, Labeling & Contamination, Rev. 8
- HPS-NGGC-0008, Performing Work in the RCA, Rev. 2
- HPS-NGGC-0014, Total Exposure, Managing RWP's, Rev. 2
- Nuclear Generation Group Program Manual (NGGM-PM)-0002, Radiation Control and Protection Manual, Rev. 33
- Plant Program (PLP)-511, Radiation Control and Protection Program, Rev. 17

<u>Radiation Work Permit (RWP) Documents</u>

- RWP 2009, Cavity Decon
- RWP 2011, Reactor Head Activities RFO-12
- RWP 2012, Refueling Activities RFO-12 (Task 05: Remove/Replace Upper Internals)
- RWP 2012, Refueling Activities RFO-12 (Task 10: Core Offload/Reload)
- RWP 2012, Refueling Activities RFO-12 (Task 14: Remove/Replace Control Rod)
- RWP 2014, Seal Table Activities RFO-12
- RWP 3071, Radiography
- RWP 3141, AOV/MOV/Valve work

<u>Licensee Records and Data</u>

- Air Sample Log, October 19-20, 2004 (Air samples related to RWPs 2014, 2012, and 2011)
- Air Sample Record, Sample #AS-20041020-012, 10/20/04
- Personnel Exposure Record, RWP 2012, Remove/Reinstall Reactor Head O-rings, 10/20/04
- Radiological Survey Record, Survey #1020-034, 10/20/04

<u>Nuclear Condition Report (NCR) and Quality Assurance (QA) Documents</u>

- Action Request (AR) 97024, RC RFO-11 Post assessment (Weakness 1), 6/25/03
- AR 103413, New HRA dose rates in 1&4 purification pump area - FHB 2165, 8/31/03
- AR 105193, RWP ED Setpoint exceeded, 9/22/03
- AR 110809, Radwaste drums dose rate measurement errors, 11/13/03
- AR 117258, Dosimetry incorrectly worn in a contamination area, 2/5/04
- AR 140366, Rad posting procedure non-compliance, 10/18/04

- AR 140663, Radiological Posting Moved, 10/19/04
- AR 140849, PCO in spent fuel decon pit, 10/21/04
- AR 141409, Poor radiation worker practice, 10/23/04
- AR 141414, Posted radiological boundary violation, 10/23/04

2OS2 As Low As Reasonably Achievable

Procedures, Manuals, and Guidance Documents

- Administrative (ADM)-NGGC-0105, ALARA Planning, Rev. 6
- AP-110, Pre-Job Briefings, Rev. 13
- AP-530, ALARA, Rev. 8
- AP-535, Performing Work in Radiation Control Areas, Rev. 17
- CAP-NGGC-0200, Corrective Action Program, Rev. 7
- DOS-NGGC-0002, Dosimetry Issuance, Rev. 21
- DOS-NGGC-0004, Administrative Dose Limits, Rev. 7
- HPS-NGGC-0003, Radiological Posting, Labeling And Surveys, Rev. 8
- HPS-NGGC-0014, Total Exposure Manage Radiation Work Permits, Rev. 2
- NGGM-PM-0002, Radiation Control and Protection Manual, Rev. 33

Licensee Records and Data

- 2003 Dose Performance By Group, Graphical Data
- 2003 Harris Plant ALARA Report
- 2004 ALARA Budget Development Worksheet
- ALARA 5 year plan 2004-2008
- ALARA Committee Meeting Minutes, January 2004-August 2004
- CY 2003 Daily Net Generation Graph
- Declaration of Pregnancy (DOS-NGGC-0002, Attachment 6), dated 8/23/04
- Embryo/Fetus Exposure Monitoring Form (DOS-NGGC-002, Attachment 8), Aug. - Oct., 2004
- Memo to HNP Supervisors from Bob Duncan, submitting 2004 Dose Budget
- Power Point presentation: Forced Outage ALARA Review
- Refueling Outage (RFO)11 ALARA Report
- RFO-11 Self-Assessment Report
- RFO-12 Dose Projections, Rev. 5, 10/13/04
- Spreadsheet: 2003 CRE Rolling Averages per Reactor

Temporary Shielding Requests (TSR)

- TSR 04-005, Seal Table, 2/2/04
- TSR 04-008, Shield curtain for storing high rad trash, 2/2/04
- TSR 04-011, Incore Sump, 2/11/04
- TSR 04-012, Reactor Head, 2/11/04
- TSR 04-028, Shield fuel transfer gap for offloading and reloading the core, 7/20/04

ALARA Work Packages (AWP)

- AWP 12-005, RFO-12 Decon Activities, Rev. 0
- AWP 12-006, Shielding Activities, Rev. 0
- AWP 12-007, RFO-12 Reactor Headwork/Refueling, Rev. 0
- AWP 12-008, ASME Section XI Inspections, Rev. 0
- AWP 12-010, Seal Table Maintenance Activities, Rev. 0
- AWP 12-011, Installation and Removal of Scaffolding, Rev. 0
- AWP 12-012, Installation and Removal of Insulation, Rev. 0
- AWP 12-016, Miscellaneous Valve Work including AOV and MOV, Rev. 0
- AWP 12-019, Reactor Coolant Pump Maintenance and Seal Replacement, Rev. 0
- RFO-12 Cavity Decon (supplement to AWP 12-005)
- SI-327 Radiography Plan (supplement to AWP 12-016)

Radiation Protection Pre-Job Brief Packages

- AP-110, Rev. 14, 10-19-04 (1630), Lift Rx Head/transfer to head stand, RWP 2011, AWP 12-007
- AP-110, Rev. 14, 10-24-04 (0700), Cutting incore thimbles underwater, RWP 2012, AWP 12-007
- AP-110, Rev. 14, 10-26-04 (0000), Radiography on 1SI-327, RWP 3071, AWP N/A
- AP-110, Rev. 14, 10-26-04 (1030), Replace Upper Internals, RWP 2012, AWP 12-007

NCR and QA Documents

- AR 94112, Annual goal for PCEs exceeded, 5/21/03
- AR 95288, Post RFO11 higher dose rates, 6/4/03
- AR 103954, Valve torques not properly planned for resin sluice, 9/8/03
- AR 105763, Water shields leaking, 9/26/03
- AR 113882, Exposure estimate exceeded for valve work, 12/18/03
- AR 130809, Dose estimate for F50/60 changeout, 6/30/04

2PS2 Radioactive Material Processing and Transportation

Procedures, Manuals, and Guidance Documents

- CAP-NGGC-0200, Corrective Action Program, Rev. 7
- HPP-880, Spent Nuclear Fuel Shipping and Receipt, Rev. 25
- HPS- NGGC-0001, Radioactive Material Receipt and Shipping Procedure, Rev. 18

Records and Data

- 2003 Annual Radioactive Effluent Release Report
- Designation of Qualified Reviewers, Memorandum to File dated June 17, 2004
- Radioactive Materials Receipt Log 2004 (Year-To-Date)

- Radioactive Materials Shipment Logbook 2004 (Year-To-Date)
- Radman Database Report for Harris Nuclear Plant, Change 39, dated 04/01/04
- Radwaste Shipment No. 03-048 shipped to UniTech Services Group, Inc.
- Radwaste Shipment No. 03-057 for IF-300 Fuel Cask shipped to Brunswick Nuclear Plant
- Radwaste Shipment Nos. 03-061 and 04-021 shipped to Studsvik, Inc.
- Radwaste Shipment No. 04-012 for IF-300 Fuel Cask received from Brunswick Nuclear Plant
- Radwaste Shipment Nos. 04-020 and 04-076 of DAW shipped to GTS Duratek
- Radwaste Shipment No. 04-058 for snubbers received from Wyle Laboratory
- Radwaste Shipment No. 04-062 shipped to Eastern Technologies, Inc.
- Radwaste Shipment No. 04-072 for Pressurizer Safety Relief Valves shipped to Wyle Laboratory
- RWP 1986, Health Physics Routines RFO-12, (Task 10: HP Shipping Activities)

NCR and QA Documents

- AR 00100320, Acceptance criteria exceeded on a RAM package, 07/29/03
- AR 00104605, Radwaste shipping cask non-compliance, 08/18/03
- AR 00110688, Contamination on shipping cask transport vehicle, 11/12/03
- AR 00110809, Radwaste drums dose rate measurement errors, 11/12/03
- AR 00110845, Radioactive Shipment Improvement, 11/13/03
- AR 00110847, Administrative errors identified on RAD shipment paperwork, 11/11/03
- AR 00112234, Spent fuel shipping assessment, 12/02/03

2PS3 Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program

Procedures, Manuals, and Guidance Documents

- DOS-NGGC-0005, Skin Dose from Contamination, Rev. 7
- DOS-NGGC-0021, Whole Body Counter (WBC) System Operation, Rev. 12
- Radiation Control Technical Report, Release of Personnel and Materials from Radiation Control Areas, Rev. 0 (2/25/00)

NCR and QA Documents

- AR 095341, HNP RC supervisor contacted by radiation protection personnel at another facility regarding radioactive particle on shoe of individual during in-processing, 5/27/03
- Radiochemistry Laboratory Analysis, sample #031406 (Hot particle - shoe), 6/10/03
- SPM-904C Calibration Record, s/n 90435, 7/26/02
- SPM-904C Calibration Record, s/n 90436, 7/26/02
- SPM-904C Calibration Record, s/n 90437, 7/26/02
- SPM-906 Calibration Record, s/n 906082, 4/25/03

4OA1 Performance Indicator Verification

Records

- 2003 Annual Radioactive Effluent Release Report
- CAP-NGGC-0200, Corrective Action Program, Rev. 7
- Chemistry and Radiochemistry (CRC)-851, ODCM Software Instructions and Documentation, Rev. 21
- Shearon Harris Nuclear Power Plant, Off-Site Dose Calculation Manual (ODCM), Rev. 16
- Operators' logs
- Equipment Inoperability Reports
- Maintenance Rule Database
- Work Order Reports
- Action Request Reports

NCR and QA Documents

- AR 128570, A slow pressure decrease was observed from the "C" Waste Gas Decay Tank resulting in an accidental release of waste gas
- Progress Energy, Shearon Harris Nuclear Power Plant, Batch Gaseous Effluent Permit Nos. 40090.009.002, 40091.099.003.G, 40092.099.004.G, 40094.071.001.G, 40095.072.001.G, 40101.073.001.G, and 40114.036.001.G

4OA2 Identification and Resolution of Problems

- CAP-NGGC-0200, "Corrective Action Program."

4OA5 Other Activities

- NDEP-0612, VT-2 Visual Examination of Nuclear Power Plant Components, Rev. 18
- HNP-04-097, 60-Day Response to NRC BL 2004-01 For the Inspection of Alloy 82/182/600
  Materials Used In the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections At Pressurized Water Reactors, 7/27/2004