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

January 30, 2006

Florida Power and Light Company
ATTN: Mr. J. A. Stall, Senior Vice President
Nuclear and Chief Nuclear Officer
P. O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: ST. LUCIE NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 05000335/2005005 AND 05000389/2005005

Dear Mr. Stall:

On December 31, 2005, the US Nuclear Regulatory Commission (NRC) completed an inspection at your St. Lucie Units 1 and 2. The enclosed integrated inspection report documents the inspection findings which were discussed on January 4, 2006, with Mr. Jefferson and other members of your staff.

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

This report documents four inspector identified findings and two self-revealing findings, all of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. Additionally, two licensee identified violations, which were determined to be of very low safety significance, are listed in this report. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCVs), in accordance with Section VI.A of the NRC's Enforcement Policy. If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the St. Lucie 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) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Joel T. Munday, Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket Nos.: 50-335, 50-389
License Nos.: DPR-67, NPF-16

Enclosure: Inspection Report 05000335/2005005, 05000389/2005005
w/Attachment - Supplemental Information

cc w/encl: (See page 3)

cc w/encl:

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-335, 50-389
License Nos.:	DPR-67, NPF-16
Report Nos.:	05000335/2005005, 05000389/2005005
Licensee:	Florida Power & Light Company (FPL)
Facility:	St. Lucie Nuclear Plant, Units 1 & 2
Location:	6351 South Ocean Drive Jensen Beach, FL 34957
Dates:	October 1 - December 31, 2005
Inspectors:	T. Hoeg, Senior Resident Inspector S. Sanchez, Resident Inspector J. Fuller, Reactor Inspector (Sections 1R08 and 4OA5) B. Crowley, NRC Consultant (Section 4OA5) J. Lenahan, Senior Reactor Inspector (Section 4OA5) J. Rivera-Ortiz, Reactor Inspector (Section 4OA5) R. Chou, Reactor Inspector (Section 4OA5) G. Kuzo, Senior Health Physicist (Sections 2OS2,4OA1, and 4OA5) A. Nielsen, Health Physicist (Sections 2OS2,4OA1, and 4OA5) J. Griffis, Health Physicist (Sections 2OS2,4OA1, and 4OA5)
Approved by:	Joel Munday, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000335/2005-05, 05000389/2005-05; 10/01/2005 - 12/31/2005; St. Lucie Nuclear Plant, Units 1 & 2; Refueling and Other Outage Activities, Inservice Inspection Activities, Radiation Safety, and PI&R Annual Sample Review.

The report covered a three month period of inspection by resident inspectors and several other inspectors from Region II. Six Green non-cited violations (NCVs) were identified. The significance of most findings is identified 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. NRC- Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing NCV of Technical Specification 6.8.1.b, “Refueling Operations,” was identified when the licensee failed to properly implement system operating procedure NOP-03.05, “Placing the 1B SDC System in Operation” while attempting to restore reactor plant shutdown cooling (SDC) flow. As a result a low pressure safety injection (LPSI) pump was started with its suction valve closed which caused the pump to cavitate. This finding had human performance cross-cutting aspects in that an operator failed to comply with procedural requirements.

  This finding is greater than minor because it is associated with the configuration control and human performance attributes of the Initiating Events cornerstone and adversely impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. In addition, if left uncorrected, this finding would result in a more significant safety concern. The inspectors evaluated the finding using Inspection Manual Chapter (IMC) 0609, Appendix G, Attachment 1, Checklist 3, "Pressurized Water Reactor (PWR) Refueling Operations with RCS Open and Refueling Cavity Level < 23 feet.” The finding affected one train of decay heat removal (DHR) which was required to be operable; therefore, the finding did not screen out in Phase 1. Subsequently, the Region II SRA evaluated the finding using the IMC 609, Appendix G, Attachment 2, Phase 2 Significance Determination Process Template for PWR During Shutdown. This finding was a precursor finding that has the potential to cause a loss of the operating train of DHR. The Phase 2 SDP evaluation determined the finding to be of very low safety significance (Green) because the required operating SDC train was only briefly interrupted; the standby SDC train was promptly placed in service; and the affected train was quickly restored. The licensee took prompt action to enter the item into their corrective action program and implement interim corrective actions. The cause of the finding is related to the cross-cutting element of human performance. (Section 1R20)

- <u>Green</u>. A NCV of TS 6.8.1.f, Fire Protection Program Implementation, was identified by the inspectors during a trend review of multiple small fires in the Unit 1 containment building during outage hot work (air arc gouging) activities. The inspectors determined that the licensee failed to take timely and effective corrective actions to control hot work activities. The licensee entered the issue into their CAP as Condition Report (CR) 2005-33661. This CR requires a comprehensive common cause analysis by a cross-functional team, which will examine the various fire protection issues identified, to determine whether or not a generic fire protection programmatic weakness is present.

  This finding is greater than minor because it is associated with the protection against external factors attribute of the Initiating Events cornerstone and adversely impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. In addition, if left uncorrected, this finding would result in a more significant safety concern. The inspectors determined that the finding was associated with fire prevention and administrative controls and assigned a low degradation rating. As a result of the phase 1 screening, this finding was determined to be of very low safety significance. The cause of the finding is related to the cross-cutting element of problem identification and resolution, specifically involving incomplete corrective actions. (Section 4OA2)

- <u>Green</u>. The inspectors identified a NCV of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures and Drawings. Licensee activities affecting quality were not accomplished in accordance with site procedures GMP-05, ADM 02.01, and QI 13/PSL2, in that a steel file that had been previously used on carbon steel material was used during weld preparation on FT-08-1A, Unit 2 Main Steam flow transmitter, which is a stainless steel, ASME Class 2, pressure boundary component. Licensee procedures specifically prohibit the use of carbon steel contaminated tools on stainless steel. The licensee failed to meet procedural requirements by not maintaining control of files, wire brushes, and grinding wheels issued from the tool room inside the Radiation Controlled Area. The licensee immediately entered the finding into their corrective action system, and conducted stand-down meetings with site welding personnel regarding the procedure requirements. The licensee decided to not install the contaminated tubing, and made new repair welds with new stainless steel base material, new wire brushes, and other new tools.

  This finding is greater than minor because if the finding was left uncorrected, it could become a more significant safety concern in that the contamination of the stainless steel base metal and weld with carbon steel particles could lead to localized outside diameter initiated pitting corrosion that would increase the likelihood of a through-wall pressure boundary leak. Additionally, the finding affected the equipment performance attribute of the Reactor Safety / Initiating Event Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. This finding was determined to be of very low safety significance based on the IMC 0609, Appendix A, Phase 1 SDP worksheet and is associated with the initiating event cornerstone. The carbon steel contamination of FT-08-1A did not contribute to the increased likelihood of a primary or secondary loss of coolant accident (LOCA), and the finding did not contribute to either the likelihood of a reactor trip or the likelihood that mitigation equipment or functions will not be available; therefore, the finding screened as Green. A contributing cause of the finding is related to the cross-cutting element of human performance. (Section 1R08)

Cornerstone: Mitigating Systems

- Green. A NCV was identified by the inspectors for the failure to comply with 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures and Drawings. As a result of plant personnel failing to follow site procedures, a bucket containing sheets of paper, tools, and other miscellaneous items were brought inside the Unit 2 reactor containment buliding (RCB), and were not listed in the foreign material exclusion (FME) log book. Upon identification of this deficiency, the licensee immediately entered the RCB and logged the bucket and all of its contents in the FME log book. Licensee management also conducted a detailed containment close out inspection prior to the restart of Unit 2.

  This finding is more than minor because foreign material left inside containment can be transported to the containment sump and cause restriction of the ECCS pump suction during LOCA conditions. Additionally, this finding is associated with the equipment performance attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). This finding was determined to be of very low safety significance (Green) based on the IMC 0609, Appendix A, Phase 1 SDP worksheet. This loss of FME control was not a design or qualification deficiency confirmed to result in loss of ECCS function per GL 91-18, nor did it represent an actual loss of a system safety function or the loss of a single train; therefore, the finding screened as Green. The inspectors determined that the human error which resulted in failure to properly document the material in the FME log was related to the human performance cross-cutting area. (Section 1R20)

Cornerstone: Occupational Radiation Safety

- Green. A self-revealing NCV of Technical Specification (TS) 6.11 was identified for failure to implement adequate radiological controls for work near contaminated pressurizer components as required by Health Physics Procedure (HPP)-3, High Radiation Areas, Rev. 17A. On November 25, 2005, two individuals entered the Unit 1 (U1) pressurizer cubicle to perform a visual inspection of valve V1249 and subsequently became contaminated due to failure to follow Radiation Work Permit (RWP) requirements and inadequate Health Physics Technician (HPT) coverage.

This finding is greater than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of exposure/contamination control and adversely affects the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operations. The failure to implement required RWP controls for high radiation area or highly contaminated area work activities could result in unintended exposures. The finding was determined to be of very low safety significance because the individuals were monitored for exposures from external radiation fields and from internally deposited radionuclides, as appropriate; and no individuals exceeded either internal or external exposure limits. The finding involved the cross-cutting aspect of Human Performance because the contamination events were a direct result of worker and HPT failures to implement required radiological controls. The licensee entered this finding in its corrective action program (CAP) as CR 2005-32859. (Section 2OS1).

Cornerstone: Public Radiation Safety

- Green. The inspectors identified a NCV of 10 CFR 71.5 for failure to follow Department of Transportation (DOT) regulations for proper closure of Type A shipping packages. Specifically, for Type A packages containing in-core instrument cutting equipment shipped on October 3, 2003 and February 17, 2005, the licensee failed to prepare the package closures in accordance with the container vendor specifications as required by DOT 49 CFR 173.475 (e).

  The finding was more than minor because it is associated with the Public Radiation Safety Cornerstone program and process attribute involving transportation procedures. The cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive material released into the public domain was affected because the identified issue involved shipments of radioactive material that were contrary to NRC and DOT regulations. The finding is of very low safety significance (Green) because it did not involve a radiation limit being exceeded or a package being breached. This finding also involved the cross-cutting aspect of problem identification and resolution regarding implementation of Operating Experience (OE). Although the licensee had reviewed OE 19531 associated with lid bolting torque for a Type A package they did not enter this OE into their CAP process and implement actions to prevent similar occurrences. After inspector identification of the issue, the licensee entered this finding into the CAP as CR 2005-25727 (Section 2PS2).

B. Licensee-Identified Violations

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

Report Details

Summary of Plant Status

Units 1 and 2 began the report period at full Rated Thermal Power (RTP). Unit 1 was shutdown on October 17, 2005, for scheduled refueling outage (RFO) 20. Unit 1 completed RFO 20 and returned to full RTP on December 22, 2005, where it remained throughout this report period. Unit 2 was shutdown due to the approach of hurricane Wilma on October 30, 2005. Unit 2 returned to full RTP on November 2, 2005, where it remained throughout this report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

On October 21 and 23, the inspectors reviewed and verified licensee actions taken in accordance with their procedural requirements prior to the onset of hurricane Wilma. The inspectors observed plant conditions, evaluating those conditions using criteria documented in Administrative Procedure ADM-04.01, Revision 12, "Hurricane Season Preparation." The inspectors performed site walkdowns and plant tours to verify the licensee had made the required preparations. The inspectors performed reviews of plant exterior areas vulnerable to high winds and hurricane conditions including the following areas:

- Switchyard
- Turbine Building
- Intake Cooling Water (ICW) Basin

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

.1 Partial Equipment Walkdowns

The inspectors conducted three partial equipment alignment verifications of the safety-related systems listed below to review the operability of required redundant trains or backup systems while the other trains were inoperable or out of service (OOS). The inspectors attempted to identify any discrepancies that could impact the function of the system, and therefore, potentially increase risk. These inspections included reviews of applicable Technical Specifications (TS), plant lineup procedures, operating procedures, and/or piping and instrumentation drawings (P&ID), which were compared with observed equipment configurations. The inspectors also reviewed applicable reactor control operator (RCO) logs; out of service and operator work around (OWA) lists; active temporary system alterations (TSA); and any outstanding condition reports (CRs) regarding system alignment and operability.

- Unit 1 Shutdown Cooling System Train B
- 1B Emergency Diesel Generator (EDG)
- 1A Vital Electrical Switchgear

b. Findings

No findings of significance were identified.

.2 Complete Equipment Walkdown

a. Inspection Scope

During the week of December 5, the inspectors completed a detailed alignment verification of the Unit 1 Component Cooling Water (CCW) system using P&ID 8770-G-083, Component Cooling System, and applicable procedures to walkdown and verify equipment alignment. The inspectors reviewed relevant portions of the Updated Final Safety Analysis Report (UFSAR) and TS, and interviewed the responsible system engineer. The inspectors also verified electrical power requirements, component labeling, pipe hangers and support installation, and associated support systems status. The walkdown also included evaluation of system piping and supports to verify that: 1) piping and pipe supports did not show evidence of water hammer; 2) oil reservoir levels indicated normal; 3) snubbers did not indicate any observable hydraulic fluid leakage; 4) hangers were within the setpoints; and 5) component foundations were not degraded. Furthermore, the inspectors examined equipment out of service lists; active open work orders (WO); the CCW system health report; and any CRs that could affect system alignment and operability.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Protection - Tours

a. Inspection Scope

The inspectors conducted tours of the eight areas listed below to verify they conformed with procedure AP-1800022, Revision 38C, Fire Protection Plan. The inspectors specifically examined any transient combustibles in the areas and any ongoing hot work or other potential ignition sources. The inspectors also assessed whether the material condition, operational status, and operational lineup of fire protection systems, equipment and features were in accordance with the Fire Protection Plan. Furthermore, the inspectors evaluated the use of any compensatory measures being performed in accordance with the licensee's procedures and Fire Protection Plan.

- Unit 2 ICW Structure
- Unit 1 Control Room Area
- Unit 1 Containment 18' Elevation
- Unit 1 Auxiliary Feedwater (AFW) Structure
- Unit 1 Containment 7.5' Elevation
- Unit 2 Boric Acid Tank Room
- Unit 2 Emergency Core Cooling System (ECCS) Room
- Unit 2 Condensate Storage Tank Enclosure

b. Findings

No findings of significance were identified.

.2 Fire Protection - Drill Observation

a. Inspection Scope

The inspectors observed a fire drill conducted in the Unit 1 Cable Spreading Room on December 5, 2005. The drill was observed to evaluate the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies, openly discussed them in a self-critical manner at the debrief, and took appropriate corrective actions as required. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient fire fighting equipment brought to the scene; (5) effectiveness of command and control; (6) search for victims and propagation of the fire into other plant areas; (7) smoke removal operations; (8) utilization of pre-planned strategies; (9) adherence to the pre-planned drill scenario; and (10) drill objectives.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

Internal Flooding

The inspectors reviewed UFSAR Section 3.4, Water Level (Flood) Design and UFSAR Table 3.2-1, Design Classification of Structures, System and Components, and that specific equipment and components in the Unit 1 ECCS pump room (i.e., HPSI, LPSI, and CS systems) that were susceptible to damage from flooding met the stated requirements. The inspectors also reviewed procedure 1-ONP-24.01, Reactor Auxiliary

Building Flooding, and verified certain actions required to be taken could be accomplished as written. The inspectors reviewed the Unit 1 ECCS pump room sump level indication and control system preventative maintenance (PM) schedule. The inspectors also verified the corrective action program (CAP) was being used to identify equipment issues that could be impacted by potential internal flooding.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

During the recent Unit 1 RFO, the inspectors observed the re-tubing activities of the 1A and 1B CCW heat exchangers. The inspectors also reviewed applicable Eddy Current Testing (ECT) procedures, equipment calibration records, and CCW Heat Exchanger Component Specific Technique Sheets. Furthermore, the inspectors also interviewed the responsible system engineer and reviewed the records and documentation indicating that the frequency of inspection was sufficient to detect degradation to ensure TS operability prior to loss of heat removal capabilities below design basis values.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Piping Systems ISI

a. Inspection Scope

From October 31 through November 4, 2005, the inspectors observed and reviewed the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and the risk significant piping system boundaries for Unit 1. The inspectors selected a sample of American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section XI required examinations and code components in order of risk priority as identified in Section 71111.08-03 of inspection procedure 71111.08, "Inservice Inspection Activities," based upon the ISI activities available for review during the onsite inspection period. The inspectors reviewed the Owner's Activity Reports (OAR) for the last two refueling outages.

The inspectors conducted an on-site review of nondestructive examination (NDE) activities to evaluate the licensee's compliance with TS; ASME Section XI and Section V, 1989 Edition with no Addenda for Class 1, 2, and 3 systems. For Unit 1, this was the first outage of the third period of the third interval. The inspectors verified that indications and defects were appropriately evaluated and dispositioned in accordance with the applicable requirements of ASME Section XI. Specifically, the inspectors observed the following examination:

Ultrasonic Testing (UT):

- 1-SGA-W5, Steam Generator 1A Primary Side Inlet Nozzle to Shell Weld, ASME Class 1

Specifically, the inspectors reviewed the following examination records:

Ultrasonic Testing (UT):

- RC-103-FW-2004, Preservice Inspection (PSI) of Tee to Reducer on Combined Pressurizer Spray Line Tee, ASME Class 1
- SI-105-3-SW-2, Pipe to Elbow, HPSI Pump 1A to Header 1A, ASME Class 2
- SI-105-4-SW-2, Pipe to Elbow, HPSI Pump 1A to Header 1A, ASME Class 2
- SI-105-4-SW-1, Elbow to Pipe, HPSI Pump 1A to Header 1A, ASME Class 2

Radiographic Examination (RT)

- RC-103-FW-2004, PSI of Tee to Reducer (#176) on Combined Pressurizer Spray Line Tee, Stainless Steel, ASME Class 1
- RT-450-00, SC 008, Pressurizer Spray Nozzle Repair, Carbon Steel to Stainless Steel, ASME Class 1
- W-123, 4" Pressurizer Spray Line Tee, ASME Class 1

Qualification and certification records for examiners, inspection equipment, and consumables along with the applicable NDE procedures for the above ISI examination activities were reviewed and compared to requirements stated in ASME Section V and Section XI.

The inspectors reviewed the licensee's operating experience assessment for issues associated with the Mihama-3 pipe rupture event of August 9, 2004. This pipe failure was attributed to Flow Accelerating Corrosion (FAC), and the inspectors held discussions with licensee personnel regarding their FAC program. The inspectors reviewed the licensee's evaluation for a FAC location that had a degraded wall thickness measurement. The inspectors reviewed information to determine that the licensee had entered this condition into their CAP, and that they were taking appropriate actions to ensure that the subject piping would have enough wall thickness to meet structural requirements for the duration of the next operating cycle.

Specifically, the inspectors reviewed the following weld records that contained recordable indications from previous examinations:

- FAC Component 12ES1-P-7-16 (Location 13), High Pressure Extraction Steam to Feedwater Heater 5A
- RT Weld Number W-119, Pressurizer Spray Line Tee, ASME Class 1
- RT Weld Number W-175, Pressurizer Spray Line Tee, ASME Class 1

Pressure boundary welding associated with one ASME Class 2 component, Unit 2 vent assembly for flow instrument FT-08-1A, was reviewed by the inspectors to verify the welding process and preservice examinations were performed in accordance with the requirements of ASME Code Sections III, V, IX, and XI. The inspectors reviewed weld data sheets, the welding procedure specification (WPS), and supporting welding procedure qualification records (PQR) for the following weld:

- Weld Joint 02037, GTAW Fillet Weld, ASME Class 2, Main Steam Flow Element

The inspectors performed a review of ISI related problems that were identified by the licensee and entered into the CAP. The inspectors reviewed these corrective action documents to confirm that the licensee had appropriately described the scope of the problems. Additionally, the inspectors' review included confirmation that the licensee had an adequate threshold for identifying issues, and had implemented effective corrective actions. Through interviews with licensee staff and review of CRs, the inspectors evaluated the licensee's threshold for identifying lessons learned from industry issues related to ASME Section XI. The inspectors performed these reviews to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

b. Findings

Introduction: A Green inspector identified Non-Cited Violation (NCV) was identified for the failure to comply with 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures and Drawings. As a result of plant personnel failing to follow site procedures, a steel file that had been previously used on carbon steel material was used during weld preparation on a safety-related, stainless steel, ASME Class 2, pressure boundary component. Licensee procedures GMP-05, ADM 02.01, and QI 13/PSL2 all contain similar requirements that prohibit the use of carbon steel contaminated steel files, grinding wheels, and wire brushes on stainless steel components.

Description: On Tuesday, October 25, 2005, the inspectors observed licensee personnel grinding on a stainless steel component with a steel file that had previously been used on carbon steel material. This observation was made during weld preparation activities for the replacement of ½ inch tubing associated with the Unit 2 Main Steam flow transmitter, FT-08-1A. A steam leak was identified by the licensee in this ASME Class 2 weld during a Mode 3 walk-down, while the plant was shutdown for Hurricane Wilma. Plant personnel associated with this work were not aware of the procedural requirements that prohibit the use of tools that may have been previously used on carbon steel material for use on stainless steel components. The workers associated with this replacement activity stated that they received the file from the hot tool room inside the Radiation Controlled Area (RCA). If this failure to implement procedural requirements was left uncorrected, contamination of stainless steel welds and base metal would continue. This contamination of stainless steel base material with carbon steel particles could lead to the increased likelihood of outside diameter initiated pitting and subsequent pressure boundary leaks. Additionally, if a weld fit up area was contaminated with carbon steel particles, the weld joint could become diluted and not maintain the proper mechanical properties as specified in its design requirements.

Procedures QI 13/PSL-2 entitled “Housekeeping and Cleanliness Control,” ADM 02.01, “Control of Chemicals and Materials for the Maintenance of Plant Systems,” GMP-05, “Control of Welding Special Processes”, and the Florida Power and Light Weld Control Manual, all contain requirements that grinding wheels, files, and wire brushes for stainless steel and nickel-based alloys shall not have been previously used on carbon or low alloy steels.

The licensee issued CR 2005-28906 for the performance deficiency identified by the inspectors. Upon identification of this deficiency, the licensee conducted a stand-down meeting with site welders and instructed them on the requirements of the procedures. The licensee decided to not install the contaminated tubing, and made new repair welds with new stainless steel base material, new wire brushes, and other new tools.

Analysis: The performance deficiency associated with this inspector-identified NCV was that licensee activities affecting quality were not accomplished in accordance with site procedures GMP-05, ADM 02.01, and QI 13/PSL2, in that a steel file that was contaminated with carbon steel particles was used on a safety-related, stainless steel, ASME Class 2, pressure boundary component. This failure to implement procedural requirements should have been prevented and was reasonably within the licensee’s ability to foresee, identify, and correct.

In accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, this finding is greater than minor because if the finding was left uncorrected, it could become a more significant safety concern in that the contamination of the stainless steel base metal and weld with carbon steel particles could lead to localized outside diameter initiated pitting corrosion that would increase the likelihood of a through-wall pressure boundary leak. Additionally, the finding affected the equipment performance attribute of the Reactor Safety / Initiating Event Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

This finding was determined to be of very low safety significance based on the IMC 0609, Appendix A, Phase 1 Significance Determination Process (SDP) worksheet. The finding is associated with the initiating event cornerstone. The carbon steel contamination of FT-08-1A did not contribute to the increased likelihood of a primary or secondary loss of coolant accident (LOCA), and the finding did not contribute to either the likelihood of a reactor trip or the likelihood that mitigation equipment or functions will not be available; therefore, the finding screened as Green.

The inspectors determined that a contributing cause of this finding was related to the human performance cross-cutting area, in that the licensee did not maintain appropriate tools for use in the hot tool room applicable to stainless steel, and the personnel responsible for issuing tools to be used on stainless steel were not aware of the procedural requirements.

Enforcement: 10 CFR 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, and drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. The licensee implements this requirement, in part, through procedures GMP-05, ADM 02.01, and QI 13, which all contain similar statements as quoted below: "Grinding wheels and brushes for stainless steel and nickel-based alloys shall not have been previously used on carbon or low alloy steels." Contrary to the above, on October 25, 2005, it was determined that the licensee had failed to accomplish weld repair activities in accordance with procedure requirements as stated in GMP-05, ADM 02.01, and QI 13, in that grinding on stainless steel components was performed using a file that had previously been used on carbon steel material.

This violation is associated with an inspection finding that is characterized by the Significance Determination Process as having very low risk significance (Green) and is in the licensee's CAP as CR 2005-28906. This violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-389/2005-05-01, Failure to Control Tooling for Use Only on Stainless Steel

.2 Boric Acid Corrosion Control (BACC) ISI

a. Inspection Scope

From October 31 through November 4, 2005, the inspectors reviewed the licensee's Boric Acid Corrosion Control Program (BACCP) to ensure compliance with commitments made in response to NRC Generic Letter 88-05 "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary," and Bulletin 2002-01 "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity."

The inspectors conducted an on-site record review, and an independent walk-down of the Reactor Containment Building (RCB), which is not normally accessible during at-power operations, to evaluate compliance with BACCP requirements and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. In particular, the inspectors verified that the licensee's visual examinations focused on locations where boric acid leaks could cause degradation of safety significant components and that degraded or non-conforming conditions were properly identified in the licensee's CAP.

The inspectors reviewed a sample of engineering evaluations completed for boric acid found on RCS piping and other ASME code class components to verify that the minimum design code required section thickness had been maintained for affected components. The inspectors also reviewed licensee CRs initiated for evidence of boric acid leakage to confirm that they were consistent with requirements of Section XI of the ASME Code, 10 CFR 50 Appendix B Criterion XVI, and licensee BACCP procedures. The inspectors reviewed the qualifications and certification records for licensee personnel who performed the boric acid walk-down inspections and who completed the engineering evaluations for identified boric acid leakage. Specifically, the inspectors reviewed:

- CR 2005-29026, Engineering Evaluation for Valve V3464 (Root Valve for FT-3305), Safety Related ASME Class 2
- CR 2005-28206, Engineering Evaluation for Valve V1337, Safety Related ASME Class 2
- CR 2005-28208, Engineering Evaluation for Valve V1339 (Root Isolation valve for PT-1181), Safety Related ASME Class 2
- CR 2005-28945, Engineering Evaluation for results of ICI Quickloc Flange Inspections

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

Resident Inspector Quarterly Review

a. Inspection Scope

On December 15, 2005, inspectors observed and assessed licensed operator actions during a simulator evaluation. During this simulator evaluation, the inspector witnessed the operating crew perform a reactor start up following a refueling outage (RFO) per operating procedure 1-320008, Rev 26, " Unit 1 Criticality Following Refueling Outage." The inspector specifically evaluated the following attributes related to the operating crews' performance:

- Clarity and formality of communication
- Prioritization, interpretation, and verification of alarms
- Control board operation and manipulation, including high-risk operator actions
- Oversight and direction provided by operations supervision, including ability to identify and implement appropriate TS actions, regulatory reporting requirements, and emergency plan actions and notifications
- Effectiveness of the post-evaluation critique

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the reliability and deficiencies associated with the two systems listed below, including associated CRs. The inspectors verified the licensee's maintenance effectiveness efforts met the requirements of 10 CFR 50.65 and Administrative Procedure ADM-17.08, Implementation of 10 CFR 50.65, The Maintenance Rule. The inspectors focused on the licensee's system functional failure determination, a(1) and a(2) classification determinations, corrective actions, and the appropriateness of established performance goals and monitoring criteria. The inspectors also attended applicable expert panel meetings, and interviewed responsible engineers. The inspectors reviewed associated system health reports, system walkdown reports, and the licensee's goal setting and monitoring requirements.

- Unit 1 CCW System
- Unit 2 Low Pressure Safety Injection (LPSI) System

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the risk assessments for the following seven Systems, Structures, or Components (SSCs) or a combination thereof that were non-functional due to planned and/or emergent work. The inspectors also walked down and/or reviewed the scope of work to evaluate the effectiveness of licensee scheduling, configuration control, and management of online risk in accordance with 10 CFR 50.65(a)(4) and applicable program procedure ADM-17.16, Implementation of the Configuration Risk Management Program. The inspectors interviewed responsible Senior Reactor Operators on-shift, verified actual system configurations, and specifically evaluated results from the online risk monitor (OLRM) for the combinations of out of service (OOS) risk significant SSCs listed below:

- Unit 1 Shutdown Safety Assessment for Hurricane Wilma
- 2B EDG Testing
- Unit 1 Safety Assessment for Loss of Shutdown Cooling Train A
- 2A ICW Pump, 2A ECCS Pumps, and 2A Instrument Air Compressor OOS For Planned Maintenance
- Unit 2 Condensate Storage Tank OOS For Test of Cross-Tie Line
- Unit 1 Entry Into Mode 3 Following Refuel Outage
- 1A Charging Pump and 1A high pressure safety injection (HPSI) Pump OOS For Valve Replacement

b. Findings

No findings of significance were identified.

1R14 Non-Routine Events

a. Inspection Scope

For the non-routine event described below, the inspectors reviewed RCO logs, plant computer data, and strip charts to determine what occurred and how the operators responded, and to determine if the response was in accordance with plant procedures:

- On October 23, 2005, the inspectors observed the site response to the approach of Hurricane Wilma. The inspectors reviewed the licensee's actions taken to prepare the plant for hurricane force winds and heavy rainfall. The inspectors also observed the licensee performing a reactor plant shutdown and placing Unit 2 in Mode 3 Hot Standby.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following five CR interim dispositions and operability determinations to ensure that TS operability was properly supported and the affected SSC remained available to perform its safety function with no increase in risk. The inspectors reviewed the applicable UFSAR, and associated supporting documents and procedures, and interviewed plant personnel to assess the adequacy of the interim CR disposition.

- CR 2005-29682, Unit 1 CCW System Train A High Radioactivity
- CR 2005-27368, Unit 1 Reactor Water Tank Foreign Material Exclusion
- CR 2005-29301, Evaluation of 1B LPSI Pump Following Cavitation
- CR 2005-30289, 2B2 Safety Injection Leakage
- CR 2005-32271, FCV-07-1B Failed to Fully Stroke Open

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

Cumulative Effects of Operator Work Arounds (OWAs)

a. Inspection Scope

The inspectors performed a semi-annual evaluation of the potential cumulative affects of all outstanding Unit 2 OWAs to determine whether or not they could affect the reliability, availability, and potential for misoperation of a mitigating system; affect multiple mitigating systems; or affect the ability of operators to respond in a correct and timely manner to plant transients and accidents. The inspectors discussed these potential effects with control room supervision and operators. The inspectors also assessed whether OWAs were being identified and entered into the licensee's CAP at an appropriate threshold. The inspectors reviewed the minutes of the quarterly OWA team meeting, which met to systematically examine individual and cumulative OWA status and repair priority, and assess overall risk.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed licensee procedures QI-3-PSL-1, Design Control, ENG-QI-1.7, Design Input / Verification, and ADM-17.11, 10 CFR 50.59 Screening, and observed part of the licensee's activities to implement a design change that affected the Unit 1 RCB ECCS sump structure. The licensee installed a permanent floor drain cover plate over an existing floor drain located within the enclosed screened area of the ECCS sump. The inspectors reviewed the associated 10 CFR 50.59 screening against the system design basis documents to verify that the modifications had not affected system operability/availability. The inspectors reviewed selected ongoing and completed work activities to verify that installation was consistent with the design control documents listed in maintenance support package (MSP) 05236, ECCS Sump Drain Cover.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors witnessed and reviewed WO post-maintenance test (PMT) activities of the six risk significant SSCs listed below. The following aspects were inspected: (1) Effect of testing on the plant recognized and addressed by control room and/or engineering personnel; (2) Testing consistent with maintenance performed; (3) Acceptance criteria demonstrated operational readiness consistent with design and licensing basis documents such as TS, UFSAR, and others; (4) Range, accuracy and calibration of test equipment; (5) Step by step compliance with test procedures, and applicable prerequisites satisfied; (6) Control of installed jumpers or lifted leads; (7) Removal of test equipment; and, (8) Restoration of SSCs to operable status. The inspectors also reviewed problems associated with PMTs that were identified and entered into the licensee's CAP.

- WO#35028804, 2A CCW Pump Breaker Trip After Pump Start
- WO#35029026, FCV-07-1B Failed Stroke Time
- WO#35011844, 1C AFW Pump Return to Service Following Maintenance
- WO#35030934, Replace Isolation Valve (V2340) for Charging Pump Discharge to 1A HPSI Header
- 1-OSP-68.01, Unit 1 RCB Integrated Leak Rate Test (ILRT) Following Outage Hatch Replacement
- Unit 1 Replacement Pressurizer Normal Operating Pressure/Temperature Test

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

Outage Planning, Control and Risk Assessment

During pre-outage planning, the inspectors reviewed the risk reduction methodology employed by the licensee for SL1-20, in particular the Risk Assessment Team (RAT) notebook. The inspectors also examined the licensee's implementation of shutdown safety assessments during SL1-20 in accordance with Administrative Procedure 0-AP-010526, Outage Risk Assessment and Control, to verify whether a defense in depth concept was in place to ensure safe operations and avoid unnecessary risk. Furthermore, the inspectors regularly monitored outage planning and control activities in the Outage Control Center (OCC), and interviewed responsible OCC management, during the outage to ensure SSC configurations and work scope were consistent with TS requirements, site procedures, and outage risk controls.

Monitoring of Shutdown Activities

The inspectors witnessed the shutdown and cooldown of Unit 1 beginning on October 17, 2005. The inspectors also monitored plant parameters and verified that shutdown activities were conducted in accordance with TS and applicable operating procedures, such as: 1-GOP-123, Turbine Shutdown - Full Load to Zero Load; 1-GOP-203, Reactor Shutdown; 1-GOP-305, Reactor Plant Cooldown - Hot Standby To Cold Shutdown; and 1-NOP-03.05, Shutdown Cooling.

Outage Activities

The inspectors examined outage activities to verify that they were conducted in accordance with TS, licensee procedures, and the licensee's outage risk control plan. Some of the more significant inspection activities accomplished by the inspectors were as follows:

- Walked down selected safety-related equipment clearance orders
- Verified operability of RCS pressure, level, flow, and temperature instruments during various modes of operation
- Verified electrical systems availability and alignment
- Reviewed actions taken in preparation for Hurricane Wilma
- Verified shutdown cooling system and spent fuel pool cooling system operation
- Evaluated implementation of reactivity controls
- Reviewed control of containment penetrations
- Examined FME controls put in place inside containment (e.g., around the refueling cavity, near sensitive equipment and RCS breaches) and around the spent fuel pool (SFP)

Refueling Activities and Containment Closure

The inspectors witnessed selected fuel handling operations being performed according to TS and applicable operating procedures from the main control room, refueling cavity inside containment and the SFP. The inspectors also examined licensee activities to control and track the position of each fuel assembly. Furthermore, the inspectors evaluated the licensee's ability to close the containment equipment, personnel, and emergency hatches in a timely manner per procedure 1-MMP-68.02, Containment Closure.

Heatup, Mode Transition, and Reactor Startup Activities

The inspectors examined selected TS, license conditions, license commitments and verified administrative prerequisites were being met prior to mode changes. The inspectors also reviewed measured RCS leakage rates, and verified containment integrity was properly established. The inspectors performed a detailed RCB sump closeout inspection prior to plant heat up operations. The inspectors also conducted a thorough RCB walkdown on December 17 after the Unit 1 reactor plant had reached Mode 3 and was at normal operating pressure and temperature. The results of low power physics testing were discussed with Reactor Engineering and Operations personnel to ensure that the core operating limit parameters were consistent with the design. The inspectors witnessed portions of the RCS heatup, reactor startup and power ascension in accordance with the following plant procedures:

- Pre-operational Test Procedure (POP) 1-3200088
- Unit 1 Initial Criticality Following Refueling
- POP 0-3200092, Reactor Engineering Power Ascension Program
- 1-GOP-201, Reactor Plant Startup - Mode 2 to Mode 1
- 1-GOP-302, Reactor Plant Startup - Mode 3 to Mode 2
- 1-GOP-303, Reactor Plant Heatup - Mode 3 <1750 to Mode 3 >1750
- 1-GOP-403, Reactor Plant Heatup - Mode 4 to Mode 3
- 1-GOP-504, Reactor Plant Heatup - Mode 5 to Mode 4

Correction Action Program

The inspectors reviewed CRs generated during SL1-20 to evaluate the licensee's threshold for initiating CRs. The inspectors reviewed CRs to verify priorities, mode holds, and significance levels were assigned as required. Resolution and implementation of corrective actions of several CRs were also reviewed for completeness. The inspectors routinely reviewed the results of Quality Assurance (QA) daily surveillances of outage activities.

b. Findings

.1 Shutdown Cooling Operations

Introduction: A Green self-revealing NCV of Technical Specification 6.8.1.b, "Refueling Operations," was identified when the licensee failed to properly implement system operating procedure NOP-03.05, "Placing the 1B SDC System in Operation" while attempting to restore reactor plant shutdown cooling (SDC) flow. As a result a LPSI pump was started with its suction valve closed which caused the pump to cavitate. This finding had human performance cross-cutting aspects in that an operator failed to comply with procedural requirements.

Description: On October 28 at 11:30 pm, while Unit 1 was in a scheduled RFO, the licensee started the 1B LPSI pump in the SDC mode of operation with its suction valve closed causing the pump to cavitate. Prior to starting the pump the SRO checked the control room operating board and misinterpreted the actual position of the suction valve, V3651, and thought it was open when it actually indicated closed. The SRO then directed the reactor operator (RO) to start the 1B SDC pump who did so without checking his panel for a proper system valve lineup. Upon starting the pump the operator immediately noticed pump flow and motor amperage were abnormal and stopped the pump. The 1A SDC train was then placed into service. The licensee subsequently conducted pump venting, vibration measurement analysis, and a pump full flow test and concluded the pump was not damaged and was operable.

Analysis: The inspectors determined that starting the 1B LPSI pump with its suction valve closed was a performance deficiency requiring a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening." The inspectors determined that the finding was more than minor because it was associated with the configuration control and human performance attributes of the Initiating Events cornerstone and adversely impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. In addition, if left uncorrected, this finding could result in a more significant safety concern.

The inspectors evaluated the finding using IMC 0609, Appendix G, Attachment 1, Checklist 3, "Pressurized Water Reactor (PWR) Refueling Operations with RCS Open and Refueling Cavity Level < 23." The finding affected one train of decay heat removal (DHR) which was required to be operable; therefore, the finding did not screen out in Phase 1. Subsequently, the Region II SRA evaluated the finding using the IMC 609,

Appendix G, Attachment 2, Phase 2 Significance Determination Process Template for PWR During Shutdown. This finding was a precursor finding that has the potential to cause a loss of the operating train of DHR. The Phase 2 SDP evaluation was performed for the LORHR initiator in POS 2 with at least 3 hours to core damage upon a complete loss of SDC. The secured SDC train was recoverable and other decay heat removal trains were available. The Phase 2 SDP evaluations determined the finding to be of very low safety significance (Green).

The inspectors determined that the operability of the pump had not been adversely impacted since the pump was operated in this condition for a minute or less, subsequent engineering evaluations and inspections determined the pump was not damaged, and a full flow test was performed satisfactorily.

Enforcement: Technical Specification 6.8.1.b, "Refueling Operations," requires, in part, that written procedures covering refueling operations shall be implemented. Contrary to the above, on October 28, 2005, the operators failed to implement operating procedure NOP-03.05, "Placing the 1B SDC System in Operation" step 6.4.5, which instructed the operators to ensure valve V3651 was open prior to starting the 1B LPSI pump. The licensee provided remedial training to those operators involved and entered the event into their CAP as CR 2005-29297. Because the failure to implement prescribed procedure steps was of very low safety significance and had been entered into the licensee's CAP, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000335/2005005-02: Failure to Implement Prescribed Procedure Steps Resulting in Starting the 1B LPSI Pump With its Suction Valve Closed.

.2 Foreign Material Exclusion Program Review

Introduction: A Green NCV was identified by the inspectors for the failure to comply with 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures and Drawings. As a result of plant personnel failing to follow site procedures, a bucket containing sheets of paper, tools, and other miscellaneous items were brought inside the Unit 2 RCB, and were not listed in the FME log book.

Description: On October 26, 2005, the inspectors conducted an independent walk-down of the Unit 2 RCB while the plant was in Mode 3. The inspectors observed that the licensee had failed to log a five gallon bucket and its contents in the required FME log book, which was located outside of the RCB personnel airlock. Licensee procedure ADM-27.13, "Foreign Material Exclusion," requires that all materials that are taken into an FME-1 Area shall be logged in and out if tool control logging is in effect. This procedure stipulates that tool control logging shall be in effect for FME-1 Areas, and the entire RCB is considered to be an FME-1 Area during Modes 1, 2, 3, and 4.

Because these items were not listed on the FME log, and the licensee did not intend to do a final inspection of the RCB prior to Unit 2 restart, this bucket and all of its contents could have been left inside the RCB. The licensee issued CR 2005-28985 and CR 2005-29019 for the performance deficiency identified by the inspectors. Upon identification of this deficiency, the licensee immediately entered the RCB and logged the bucket and all of its contents in the FME log book. Licensee management also conducted a detailed containment close out inspection prior to the restart of Unit 2.

Analysis: The performance deficiency associated with this inspector-identified NCV was that licensee activities affecting quality were not accomplished in accordance with site procedure ADM-27.13, in that a bucket containing sheets of paper, tools, and other miscellaneous items were left unattended inside the Unit 2 RCB, and were not listed in the FME log book. This failure to implement procedural requirements could have been prevented and was reasonably within the licensee's ability to foresee, identify, and correct.

In accordance with IMC 0612, Appendix B, this finding is more than minor because foreign material left inside containment could potentially be transported to the containment sump and cause restriction of the ECCS pump suction during LOCA conditions. Additionally, this finding is associated with the equipment performance attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage).

This finding was determined to be of very low safety significance (Green) based on the IMC 0609, Appendix A, Phase 1 SDP worksheet. The finding affected components associated with the mitigating systems cornerstone. This loss of FME control was not a design or qualification deficiency confirmed to result in loss of ECCS function per GL 91-18, nor did it represent an actual loss of a system safety function or the loss of a single train; therefore, the finding screened as Green.

The inspectors determined that the human error, which resulted in failure to properly document the material in the FME log, was related to the human performance cross-cutting area.

Enforcement: 10 CFR 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, and drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. The licensee implements this requirement, in part, through procedure ADM-27.13, which states in part, that tools or materials to remain inside the FME area upon establishing access controls are to be entered on the Access Control Log (Appendix A of ADM-27.13). The procedure also states that when materials and tools are designated as "long term," the access control log sheet(s) identifying such materials and tools shall remain an active part of the log. Contrary to the above, on October 26, 2005, it was determined that the licensee had failed to follow site procedure requirements as stated in licensee procedure ADM-27.13, in that numerous items were left inside the FME area, and were not entered on the access control log sheets.

This violation is associated with an inspection finding that is characterized by the SDP as having very low risk significance (Green) and is in the licensee's CAP as CR 2005-28985 and CR 2005-29019. This violation is being treated as a NCV, consistent with

Section VI.A of the NRC Enforcement Policy: NCV 50-389/2005-05-03, Loss of FME Integrity When Material Was Found in RCB Which Was Not on the FME Log.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed portions of the following six surveillance tests and monitored test personnel conduct and equipment performance, to verify that testing was being accomplished in accordance with applicable operating procedures. The test data was reviewed to verify it met TS, UFSAR, and/or licensee procedure requirements. The inspectors also verified that the testing effectively demonstrated the systems were operationally ready, capable of performing their intended safety functions, and that identified problems were entered into the licensee's CAP for resolution. The tests included one inservice test (IST) as follows:

- OP 1-220050A, 1A EDG Periodic Test
- OP 2-0410050, 2A High Pressure and Low Pressure Safety Injection System Functional Tests
- OSP 1-03.02A/B, 1A/1B LPSI Flow Test
- OP 1-0410025, Stroke Testing of the Safety Injection Tank Discharge and SI Loop Check Valves
- OP 2-220050A, 2A EDG Periodic Test
- OSP 1-68.01, Containment integrated leak rate test (ILRT)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors continued to periodically screen active TSAs for risk significant systems. The inspectors examined the TSA listed below, which included a review of the technical evaluation and its associated 10CFR50.59 screening. The TSA was compared against the system design basis documentation to ensure that (1) the modification did not adversely affect operability or availability of other systems; (2) the installation was consistent with applicable modification documents; and (3) did not affect TS or require prior NRC approval. The inspectors also observed accessible equipment related to the TSA to verify configuration control was maintained.

- TSA 2-05-10, Temporary Gage Installation at FT-3341 on Reactor Coolant Pump 2B2

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Occupational Radiation Safety Cornerstone

2OS1 Access Controls To Radiologically Significant Areas

Access Controls The inspectors reviewed and evaluated licensee guidance and its implementation for controlling and monitoring worker access to radiologically significant areas and tasks associated with the Unit 1 Refueling Outage Cycle 20 (1 RFO 20). The inspectors evaluated changes to, and adequacy of procedural guidance; directly observed implementation of established administrative and physical radiological controls; appraised radiation worker and technician knowledge of, and proficiency in implementing radiation protection activities; and assessed radiation worker (radworker) exposures to radiation and radioactive material.

The inspectors directly observed controls established for workers and Health Physics Technician (HPT) staff in potential airborne radioactivity area, radiation area, high radiation area (HRA), locked-high radiation area (LHRA), and very high radiation area (VHRA) locations. Controls and their implementation for LHRA keys and for storage of irradiated material within the U1 spent fuel pool (SFP) were reviewed and discussed in detail. Established radiological controls were evaluated for selected 1 RFO 20 tasks including original reactor vessel closure head (ORVCH) and original pressurizer (OPZR) replacement and temporary onsite storage; Resistance Temperature Detector (RTD) modifications; refueling operations; valve maintenance; radioactive waste (radwaste) processing and storage; and radioactive material/waste shipping activities. In addition, licensee controls for areas where dose rates could change significantly as a result of plant shutdown and refueling operations were reviewed and discussed.

For selected tasks, the inspectors attended pre-job briefings and reviewed RWP details to assess communication of radiological control requirements to workers. Occupational worker adherence to selected RWPs and HPT proficiency in providing job coverage were evaluated through direct observations and interviews with licensee staff. Electronic dosimeter (ED) alarm set-points and worker stay times were evaluated against applicable radiation survey results. Worker exposure as measured by ED and by licensee evaluations of skin doses resulting from discrete radioactive particle or dispersed skin contamination events during current 1 RFO 20 activities were reviewed and assessed independently. For HRA tasks involving significant dose gradients, e.g., RTD modifications and ORVHR activities, the inspectors evaluated the use and placement of whole body and extremity dosimetry to monitor worker exposure.

Postings and physical controls established within the radiologically controlled area (RCA) for access to the U1 reactor containment building (RCB) and the U1 and Unit 2 (U2) reactor auxiliary building (RAB) locations; radioactive waste processing, storage, shipping equipment and locations; and the low level waste sorting and radioactive material storage facilities were evaluated directly during facility tours. The inspectors independently measured radiation dose rates or directly observed conduct of licensee radiation surveys and results for OPRZ and ORVCH equipment and temporary storage general area locations, U1 RTD hot-leg work locations, and radioactive material/waste shipping tasks. Results were compared to current licensee surveys and assessed against established postings and radiation controls. Licensee controls were observed for selected U1 containment, U1 and U2 RAB LHRA and VHRA locations.

The inspectors evaluated implementation and effectiveness of licensee controls for both airborne and external radiation exposure. The inspectors reviewed and discussed selected whole-body count (WBC) analyses conducted between October 1, 2004, and December 1, 2005, to evaluate implementation and effectiveness of personnel monitoring, and administrative and physical controls including air sampling, barrier integrity, engineering controls, and postings for tasks having the potential for individual worker internal exposures to exceed 30 millirem (mrem) Committed Effective Dose Equivalent (CEDE). Effectiveness of external radiation exposure controls were evaluated through review and discussions of individual worker dose as measured by Thermoluminescent Dosimeter (TLD) or ED for calendar year (CY) 2004 and year-to-date 2005, focusing on selected 1 RFO 20 tasks.

Radiation protection activities were evaluated against Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), and 10 Code of Federal Regulations (CFR) Parts 19 and 20 requirements. Specific assessment criteria included UFSAR Section 11, Radioactive Waste Management, and Section 12, Radiation Protection; 10 CFR 19.12; 10 CFR 20, Subpart B, Subpart C, Subpart F, Subpart G, Subpart H, and Subpart J; TS Sections 6.8, Procedures, 6.11, Radiation Protection Program, and 6.12, High Radiation Area. Detailed procedural guidance and records reviewed for this inspection area are listed in Sections 2OS1, 2OS2, 2PS2, 4OA1 and 4OA5 of the report Attachment.

Problem Identification and Resolution Licensee Corrective Action Program (CAP) documents associated with access control to radiologically significant areas were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with Nuclear Administrative Procedure (NAP) - 204, Condition Reporting, Revision (Rev.) 6. Licensee CAP documents associated with access control issues, personnel radiation monitoring, and personnel exposure events which were reviewed and evaluated in detail during inspection of this program area are identified in Sections 2OS1, 2OS2, 2PS2, and 4OA1 of the report Attachment.

The inspectors completed 21 of the specified line-item samples detailed in Inspection Procedure (IP) 71121.01.

b. Findings

Introduction: A self-revealing, Green NCV of TS 6.11 was identified for failure to implement adequate HP controls for work conducted near contaminated pressurizer components as required by HPP-3, High Radiation Areas, Rev. 17A.

Description: On November 25, 2005, two individuals entered the U1 pressurizer cubicle to perform a visual inspection of valve V1249 internal components. The work location was controlled as a HRA since general area dose rates were greater than (>)100 millirem per hour (mrem/hr) and the interior of V1249 was highly contaminated with levels of removable contamination on the order of 1 rad beta per hour (rad β-/hr) per 100 square centimeters (100 cm^2). The workers were logged in on RWP No. 05-710, which required the use of power visors and paper coveralls, in addition to normal Protective Clothing (PC), to reduce the risk of contamination. Contrary to the established RWP requirements, the workers entered the work area in single PCs and without power visors. Although constant HP coverage was required and HPTs have the responsibility to stop work when RWP requirements are not adhered to, the technician providing job support allowed the workers to breach the valve cover and begin their inspection without the proper PCs. The inspection was stopped by the workers a short time later due to Foreign Material Exclusion (FME) concerns regarding sparks from nearby welding activities. Both workers subsequently alarmed personnel contamination monitors at the RCA exit point and were sent for investigational whole body counts. It was determined that both workers had facial contamination along with positive intakes of radioactive material.

Analysis: The inspectors determined that the licensee's failure to implement adequate HP controls for a HRA is a performance deficiency because the licensee is required by TS 6.11 to maintain and implement procedures for radiation protection and the incident was reasonably within the licensee's ability to foresee and correct. This finding is greater than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of exposure/contamination control and adversely affects the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The failure to follow established RWP requirements and to provide adequate HPT coverage for the workers in HRAs or highly contaminated areas could result in unintended occupational exposures. The finding was evaluated using the Occupational Radiation Safety SDP. The finding was not related to ALARA planning, nor did it involve an overexposure or substantial potential for overexposure, and the ability to assess dose was not compromised. For these reasons, and the fact that intakes of radioactive material were low, the SDP evaluation concluded that the finding was of very low safety significance (Green). The finding involved the cross-cutting aspect of Human Performance because the contamination events were a direct result of worker and HP technician failure to follow RWP and procedural requirements.

Enforcement: TS 6.11 requires the licensee to establish, implement, and maintain the procedures for personnel radiation protection consistent with the requirements of 10 CFR 20. Licensee procedure HPP-3, High Radiation Areas, states that individuals working in HRAs shall adhere to RWP requirements and that HP technicians providing job coverage shall be knowledgeable of those requirements and will maintain control over workers in the area. Contrary to the above, on November 25, 2005, two workers entered a HRA without RWP required protective gear and a HP technician failed to exercise control over the work activities. Because the failure to comply with TS 6.11 is of very low safety significance and has been entered into the licensee's CAP (CR 2005-32859), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000335/389, 2005005-04, Failure to Implement Required HP Controls for U1 Pressurizer Valve Work.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA) The inspectors reviewed ALARA program guidance and its implementation for ongoing 1 RFO 20 tasks. The inspectors evaluated the accuracy of ALARA work planning and dose budgeting, observed implementation of ALARA initiatives and radiation controls for selected jobs in-progress, assessed the effectiveness of source-term reduction efforts, and reviewed historical dose information.

ALARA planning documents and procedural guidance were reviewed and projected dose estimates were compared to actual dose expenditures for OPZR and ORVCH replacement, RTD maintenance activities, and refueling operations. Differences between budgeted dose and actual exposure received were discussed with cognizant ALARA staff. Changes to dose budgets relative to changes in radiation source term and/or job scope were also discussed. The inspectors attended pre-job briefings and evaluated the communication of ALARA goals, RWP requirements, and industry lessons-learned to job crew personnel. The inspectors also attended a Plant ALARA Review Committee meeting and observed the interface between plant management and ALARA planning staff.

The inspectors made direct field or closed-circuit-video observations of outage job tasks including OPZR and ORVCH replacement activities. For the selected tasks, the inspectors evaluated radworker and HPT job performance; individual and collective dose expenditure versus percentage of job completion; surveys of the work areas, appropriateness of RWP requirements; and adequacy of implemented engineering controls. For selected tasks, the inspectors interviewed radworkers and job sponsors regarding understanding of dose reduction initiatives and their current and expected accumulated doses at completion of the job tasks.

Implementation and effectiveness of selected program initiatives with respect to source-term reduction were evaluated. Chemistry program ALARA initiatives and their effect on U1 RCB and U1/U2 RAB dose rate trends were reviewed. The effectiveness of temporary shielding installed for the current outage was assessed through review of shielding request packages and pre-shielding versus post-shielding dose rate data. The inspectors also reviewed results of pipe flushing to reduce dose rates associated with OPZR activities. In addition, implementation of cobalt reduction initiatives for selected valve maintenance activities were discussed and reviewed.

Plant exposure histories for calendar year (CY) 2003, CY 2004, and YTD 2005 were reviewed, as were established goals for reducing collective exposure during the current 1 RFO 20. The inspectors reviewed and discussed dosimetry issuance and exposure tracking and goals for individuals, departments, and for selected outage tasks. The inspectors also examined and discussed dose records of declared pregnant workers to evaluate assignment of gestation dose. In addition, selected individual access records were reviewed for dose received during work in areas with high dose rate gradients and for issuance of administrative dose extensions .

ALARA program activities and their implementation were reviewed against 10 CFR Part 20, and approved licensee procedures. In addition, licensee performance was evaluated against guidance contained in Regulatory Guide (RG) 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable and RG 8.13, Instruction Concerning Prenatal Radiation Exposure. Procedures and records reviewed within this inspection area are listed in Section 2OS2 of the report Attachment.

Problem Identification and Resolution The inspectors reviewed selected CRs, self-assessments, and audits in the area of exposure control and ALARA activities. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with NAP - 204, Condition Reporting, Rev. 6. Specific CAP documents reviewed for this inspection area are identified in Section 2OS2 of the report Attachment.

The inspectors completed 28 of the specified line-item samples detailed in IP 71121.02.

b. Findings

No findings of significance were identified.

Public Radiation Safety Cornerstone

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization The inspectors reviewed and discussed the currently installed radioactive waste (radwaste) processing systems as described in the UFSAR, Section 11. In addition, stored and disposed radwaste types and quantities as documented in Effluent Release Report for CYs 2003 and 2004 were discussed with responsible licensee representatives.

The operability and configuration of selected liquid and solid radwaste processing systems and equipment were evaluated. Inspection activities included document review, interviews with plant personnel, and direct inspection of processing equipment and piping. The inspectors directly observed equipment material condition and configuration for liquid and solid radwaste processing systems and licensee staff were interviewed regarding equipment function and operability. The licensee's policy regarding abandoned radwaste equipment was reviewed and discussed with cognizant licensee representatives. Engineering staff were interviewed to assess knowledge of radwaste system processing operations. Procedural guidance involving resin dewatering activities and filling of waste packages was reviewed for consistency with the licensee's Process Control Program (PCP) and internal procedures.

Licensee radionuclide characterizations of each major waste stream were evaluated. For dry active waste (DAW), primary resin, secondary resin, and filters, the inspectors evaluated PCP and licensee procedural guidance against 10 CFR 61.55 and the Branch Technical Position (BTP) on Radioactive Waste Classification. Part 61 data and scaling factors were reviewed and discussed with licensee representatives for radwaste processed or transferred to licensed burial facilities from January 2001 through September 2005. The licensee's analyses and current scaling factors for quantifying hard-to-detect nuclides were assessed. The inspectors discussed potential for changes in plant operating conditions and reviewed selected DAW and primary resin waste stream radionuclide data to determine if known plant changes were assessed and radionuclide composition remained consistent for the period reviewed. Procedures and records reviewed within this inspection area are listed in Section 2PS2 of the report Attachment.

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

The inspectors assessed shipping-related procedures for compliance to applicable regulatory requirements. Selected shipping records were reviewed for completeness and accuracy, and for consistency with licensee procedures. Training records for individuals qualified to ship radioactive material were checked for completeness. In addition, inspectors assessed the specific training curricula provided to workers involved with packaging and preparing the ORVCH for temporary storage and subsequent shipment. Inspectors directly observed package preparation for a shipment of radioactive laundry to a processing vendor; independently verified results of contamination and direct radiation surveys; evaluated shipping paperwork for completeness; and assessed initial loading, bracing, and placarding of the transport vehicles. Responsible staff were interviewed to assess their knowledge of package preparation specifications, and applicable radiation and contamination control limits.

Transportation program guidance and implementation were reviewed against regulations detailed in 10 CFR 71, 49 CFR 170-189, and applicable licensee procedures. In addition, training activities were assessed against 49 CFR 172 Subpart H, and the guidance documented in NRC Bulletin 79-19. Procedures and records reviewed within this inspection area are listed in Section 2PS2 of the report Attachment.

Problem Identification and Resolution Licensee CAP documents associated with radwaste processing and transportation activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with NAP-204, Condition Reporting, Rev. 6.

Program audits and selected CAP documents reviewed for this inspection area are identified in Section 2PS2 of the report Attachment.

The inspectors completed the six specified line-item samples detailed in IP 71122.02.

b. Findings:

Introduction: A Green NRC-identified NCV of 10 CFR 71.5 was identified for failure to implement current package design specifications for proper closing of Type A shipping packages as required by DOT regulations. Specifically, for Type A packages containing in-core instrument cutting equipment shipped on October 3, 2003, and February 17, 2005, the licensee failed to close the package in accordance with vendor specifications as required by 49 CFR 173.475(e).

Description: The inspectors determined that the licensee did not possess all of the package preparation instructions referenced within the container certification for use in preparation of a DOT Type A package (Model SSB-18-1-7A-TRF). The Type A package was used for shipment of in-core instrument cutting equipment to other licensed facilities offsite. Subsequent calls to the vendor supplying the Type A packages resulted in receipt of current and accurate package certification and engineering evaluation documentation. The inspectors noted that the documents provided specifications for lid closure device (T-bolt) torque values, bolt closure sequence, and seal gasket inspection procedures. From review and discussion of shipping records associated with previous Type A package shipments made on October 3, 2003 and February 17, 2005, the inspectors noted that required lid closure device torque values and assembly configurations were not specified. Licensee representatives stated that previous package preparation guidance did not specify a required closure torque value nor a closure device configuration, but only required the verification that the 'T-bolts' and lid were secured tightly.

Analysis: The licensee's failure to satisfy 10 CFR 71.5 which requires compliance with 49 CFR Part 173 for DOT Type A package vendor engineering analysis specifications is a performance deficiency. Specifically, users of Type A packages are expected to maintain and implement current package engineering analysis closure specifications. The finding was more than minor because it is associated with the public radiation cornerstone program and process attribute and it affected the cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive material released into the public domain. This finding resulted in the transportation of Type A packages that had not been properly secured for shipment as required by DOT regulations. The issue was reviewed using the Public Radiation Safety Significance Determination Process and was determined to be of very low safety significance (Green) because it did not involve a radiation limit being exceeded nor packaging being breached. The licensee initiated immediate corrective actions to assure that future Type A packages are prepared properly prior to shipment. This finding involved the cross-cutting aspect of problem identification and resolution because missed lid bolting torque for a Type A package was discussed in OE 19531, however, the licensee failed to enter this OE into their CAP and ensure its implementation into their transportation program activities.

Enforcement: 10 CFR 71.5 requires licensees to conform with the regulations in DOT 49 CFR Parts 170 through 189. For Type A package shipments, 49 CFR 173.415(a) requires each offeror of a Specification 7A Type A Package to maintain on file for at least one year after the latest shipment, complete documentation of tests and engineering evaluation or comparative data showing the construction methods, packaging design and materials of construction to comply with that specification. Further, 49 CFR 173.475(e) requires that each special instruction for closing and preparation of a package be followed. For shipments of Type A packages made on October 3, 2003, and February 17, 2005, the licensee failed to implement current design document specifications for closure of the DOT Type 7A packages, in that, T-bolt torque values and closure assembly specifications for package closure were not met. The licensee documented this issue in its CAP as CR No. 2005-25727. Since this violation is of very low safety significance and the licensee entered the finding into its CAP, this violation is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000335, 389/2005005-05, Failure to Implement Appropriate DOT Type A Package Closure Requirements.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee data for the performance indicators (PIs) listed below. To verify the accuracy of the reported PI data, PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Rev. 3, were used to screen each data element.

Public Radiation Safety Cornerstone The inspectors reviewed the Radiological Control Effluent Release Occurrences PI results from October 2004, through September 2005. For the assessment period, the inspectors reviewed cumulative and projected doses to the public, out-of-service (OOS) effluent radiation monitors and selected compensatory sampling data, and CRs related to RETS/ODCM issues. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. Documents reviewed are listed in Section 4OA1 of the report Attachment.

Occupational Radiation Safety Cornerstone The inspectors reviewed the Occupational Exposure Control Effectiveness PI results from October 2004 through September 2005. For the assessment period, the inspectors reviewed electronic dosimeter alarm logs and CRs related to controls for exposure significant areas. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. Report Section 2OS1 contains additional details regarding the inspection of controls for exposure significant areas. Documents reviewed are listed in Sections 2OS1 and 4OA1 of the report Attachment.

The inspectors completed the two specified radiation protection line-item samples detailed in IP 71151.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

1. Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for followup, the inspectors performed screening of all items entered into the licensee's CAP. This was accomplished by reviewing the description of each new CR and periodically attending CR oversight group meetings.

2. Semi Annual Review to Identify Trends

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors review included daily screening of individual CRs, licensee trending efforts, and licensee human performance self-assessments. The inspectors review nominally considered the six month period of July 2005 through December 2005. Furthermore, the inspectors verified whether adverse or negative trends and issues identified in the licensee's reports were entered into the CAP.

b. Assessment and Observations

During the past six months, the inspectors identified two issues and the licensee identified one issue related to plant configuration control through the use of the licensee's TSA process. The licensee's guidance for implementing a temporary design change to the plant is through administrative procedure ADM-17.18, "Temporary System Alteration," which states that the purpose of a TSA is to maintain configuration control for non-permanent changes to plant SSCs while ensuring the applicable technical and administrative reviews and approvals are obtained. The inspectors had previously identified two plant modifications that the licensee did not control by the use of their TSA procedure including both a structural floor drain modification and an electrical control rod assembly position indication modification as documented in St. Lucie Plant Inspection Reports 2005-03 and 2005-04 respectively. In addition, the licensee identified a refueling water tank structural modification which was also implemented outside the TSA process during SL1-20 and entered it into their CAP as CR 2005-27368. The licensee considered these findings as a possible trend and tasked their system engineering department to perform reviews of their assigned SSCs to ensure current as-is configuration was in accordance with their docketed design bases. The results of the system engineering review did not identify any additional significant configuration control deficiencies.

The licensee subsequently trained their engineering staff on the usage of the TSA process and discussed recent findings related to not using the required TSA procedure. The licensee is evaluating a possible revision to their TSA procedure to clearly define what a TSA consists of and expectations on when and how to implement the TSA process. The inspectors considered the licensee's actions to be adequate in minimizing the recurrence of TSA misuse in the future.

.3 Annual Sample: Review of Hot Work Fire Prevention Controls

a. Inspection Scope

The inspectors reviewed licensee actions to resolve a trend in small fires resulting from hot work activities inside the Unit 1 RCB. The inspectors reviewed how the licensee performed hot work activities including planning, scheduling, and controlling evolutions that could produce sparks, slag, or other fire ignition sources. During the recent Unit 1 RFO, the licensee performed numerous metal cutting and arc-gouging operations associated with the planned pressurizer and reactor vessel head replacement modifications.

b. Findings and Observations

Introduction: The inspectors identified a Green NCV during a trend review of several small fire events which occurred over several weeks in the Unit 1 RCB during outage hot work (air arc gouging) activities. The inspectors determined that the licensee failed to take timely and effective corrective actions to prevent recurrence of fires started from hot work activities.

Description: On October 28, 2005, during arc-gouging metal removal activities on the reactor coolant loop A hot leg whip restraint, a small fire occurred on the fire blanket used to deflect and or capture hot slag. The fire watch immediately responded to the source of the fire and extinguished it with the use of a dry chemical type extinguisher. The licensee recommended actions to prevent recurrence included the availability of a garden sprayer to maintain the fire blanket wet at all times. On November 1, 2005, another fire occurred during hot work to remove a whip restraint on the B steam generator hot leg nozzle. The fire ignited a 4-inch diameter flexible duct used as an exhaust path to a HEPA filter intended to remove smoke and fumes from the work area. Although personnel in the work area extinguished the fire within two minutes, the containment atmosphere became slightly smoke filled and the containment was evacuated.

The licensee performed a root cause analysis following the flexible duct fire. Interim corrective actions included: directions for ensuring the placement of flexible ducts such that they are above the hot work or between the hot work and the worker, and far enough away that slag does not reach the duct opening; protecting HEPA flexible ducts with fire retardant material when run near or below hot work; and, including an information package with all hot work pre-job briefs regarding preliminary available information regarding this event. The corrective actions resulting from the licensee's root cause investigation was to revise procedure AP 0010434, "Plant Fire Protection Guidelines," to include information on the appropriate use of flexible ducts or ventilation equipment; revise hot work permit forms to include a check box to review the ventilation equipment precautions prior to permit approval; and revise the hot work permit form to include criteria that will prompt a review/approval section by the fire protection engineer or outage/work control center.

On November 8, 2005, while performing air arc gouging on the containment hatch transfer system (HTS), a crane signalman's vest lying in the area, ignited from slag generated from this work. The deck of the HTS had been covered by fire blankets in preparation for this work but in the process of personnel and equipment movement on the deck, the vest became uncovered by the fire blankets and ignited. The fire watch immediately spotted the fire and put it out with a fire extinguisher. This event was evaluated as part of the root cause analysis for the fire that previously occurred on November 1.

On November 18, 2005, air arc gouging was being performed on the annulus side of the containment construction hatch when hot slag came into contact with and ignited an air line that was being used for the arc-gouging process. Additionally, on November 20, 2005, the smell of smoke was reported from individuals working above the 62-foot elevation inside the pressurizer cubicle. Subsequent investigation by the licensee found an empty smoldering bolt bag near the base of the pressurizer skirt that was promptly extinguished and removed from the area. Apparently the bag had been inadvertently placed in an area being controlled as a hot work area. Apparent cause evaluations were performed by the licensee for these two fires with the results being a combination of inadequate situational assessment, unusual or hazardous conditions not being defined, and poor housekeeping. Resultant licensee corrective actions were to ensure site wide dissemination of these recent fire events.

<u>Analysis</u>: The inspectors determined that multiple fires inside containment caused by hot work activities was a performance deficiency requiring a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening." The inspectors determined that the finding was more than minor because it was associated with the protection against external factors attribute of the Initiating Events cornerstone and adversely impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. In addition, if left uncorrected, this finding could result in a more significant safety concern. Specifically, one of the key attributes associated with this cornerstone objective is protection against fires, and the inspectors determined that the licensee's failure to take timely and effective corrective actions with respect to hot work ignition control deficiencies was a performance deficiency. The cause of the finding is related to the cross-cutting element of problem identification and resolution, specifically involving incomplete corrective actions.

The inspectors conducted a Phase 1 initial screening in accordance with IMC 0609, "Significance Determination Process," and because the finding was associated with fire protection, this was accomplished using IMC 0609, Appendix F, Attachment 1, "Fire Protection SDP Phase 1 Worksheet." The inspectors determined that the finding was associated with fire prevention and administrative controls and assigned a low degradation rating. As a result of the Phase 1 screening, this finding was determined to be of very low safety significance (Green).

Enforcement: TS 6.8.1.f, states, in part, that procedures shall be established, implemented, and maintained covering Fire Protection Program implementation. The St. Lucie Fire Protection Plan, as delineated in Administrative Procedure 1800022, refers in Section 8.10 to FPL's Topical Quality Assurance Report (TQAR) for requirements on corrective actions related to Fire Protection. Section 16.1 of the TQAR, states, in part, that for conditions adverse to quality, the cause of the condition shall be determined and action taken to preclude repetition. Contrary to this requirement, the licensee failed to take timely and adequate corrective actions for deficiencies relating to hot work ignition source controls in the Unit 1 RCB. Specifically, multiple fires occurred in the Unit 1 containment caused, in part, by deficiencies related to hot work ignition source control during arc-gouging activities.

The licensee entered the issue into their CAP as CR 2005-33661. This CR requires a comprehensive common cause analysis by a cross-functional team, which will examine the various fire protection issues identified to determine whether or not a generic fire protection programmatic weakness is present. Because the licensee has entered the issue into their CAP and the finding is of very low safety significance (Green), this violation of TS 6.8.1, is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000335/2005005-06: Failure to Take Adequate Corrective Actions for Deficiencies Relating to Hot Work Ignition Source Controls in the Unit 1 RCB.

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 05000335/2005001, Condition Prohibited by TS Due to Inoperable Reactor Protection Instrumentation Channel and Engineered Safety Feature Actuation System (ESFAS) Instrumentation Channel

On February 19, 2005, while Unit 1 was operating at full power, the licensee identified a 3 to 4 percent high deviation between 1B steam generator narrow range water level indicator, LIC-9023A, and its three redundant level channels. The cause of this high deviation was determined to be a packing leak on an instrument isolation valve in the upper sensing line attached to the condensing pot. The instrument valve packing leakage exceeded the condensing pot's makeup capability and voided a portion of the reference leg. The lowered reference leg water level caused LIC-9023A to indicate a higher than actual steam generator water level. A subsequent engineering evaluation concluded that steam generator water level channel indication deviations greater than 2 percent render the channel inoperable. Consequently, the channel was declared OOS on March 22, 2005. Per TS 3.3.1 and 3.3.2 the inoperable Reactor Protective System Instrumentation and ESFAS Instrumentation channels were placed in the tripped condition within one hour. No new findings were identified in the inspector's review. This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee documented the problem in CR 2005-8049. This LER is closed.

.2 (Closed) Licensee Event Report (LER) 05000335/2005003, 1A EDG Actuation

On May 18, 2005, the 1A EDG auto started during scheduled emergency electrical bus undervoltage testing due to a test circuit control logic relay design condition which could allow the operation of a test switch to cause an actual 4160 Volt feeder breaker to open unexpectedly. Turning the test switch to the test position prevents an actual load shed by bypassing the undervoltage trip actuating relay contacts and connects them to the test lamp circuit. The most probable cause was determined to be a design weakness allowing an electrical race between test switch CS-9/949 and relay 27X-9/1A3 when the test switch is operated from the RELAY TEST to the OFF position in a rapid manner. A key contributing cause was that the procedure did not require slow operation of the switch. The licensee documented the condition in CR 2005-14777. The LER was reviewed and no findings of significance were identified. No violation of NRC requirements occurred. This LER is closed.

.3 (Closed) Licensee Event Report (LER) 05000389/2004004, Manual Reactor Trip Due to Condensate Pump Motor Malfunction

On December 25, 2004, the operators manually tripped the Unit 2 reactor after high temperatures were observed locally at the 2B condensate pump motor. The cause of the high temperature was subsequently identified as a motor overcurrent condition caused by an incorrect motor lug assembly process performed by the manufacturer. The licensee corrected the condition and planned inspections of other large risk significant motors for potential similar assembly deficiencies in order to evaluate the extent of the condition. The LER was reviewed and no findings of significance were identified. No violation of NRC requirements occurred. This LER is closed.

4OA5 Other Activities

.1 Institute of Nuclear Power Operations (INPO) Plant Assessment Report Review

a. Inspection Scope

The inspectors reviewed the final report for the INPO plant assessment of St. Lucie station conducted in April 2005. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings of significance were identified.

.2 Review of Reactor Vessel Closure Head (RVCH) and Pressurizer (PZR) Replacement Lifting and Transportation Program Activities

a. Inspection Scope

The inspectors reviewed the RVCH and PZR lifting programs as described in Plant Change & Modification (PC/M) Packages PC/M No. 04136M, Rev. 0, “Outside Service Platform and Outside Lifting System for Unit 1 RVCH/PZR” and PC/M No. 04137M, Rev. 0, “Hatch Transfer System and Heavy Rigging for Unit 1 RVCH/PZR”, to assure that they were prepared in accordance with regulatory requirements, appropriate industrial codes and standards, and to verify that the maximum anticipated loads to be lifted would not exceed the capacity of the lifting equipment and supporting structures.

The inspectors examined the RVCH and PZR Replacement Project lifting and transportation equipment including the polar crane, hatch transfer and skid system, the down/up-ender device, outside lifting system and the Self Propelled Modular Transport. The inspectors observed portions of transportation, lifting, and setting in position of the new pressurizer.

The inspectors reviewed the transport programs as described in PC/M No. 04138M, Rev. 0, “RVCH Offload & Transport for Unit 1 RVCH Replacement Project” and PC/M No. 04152M, Rev. 0, Pressurizer Offload and Transport for Unit 1 Pressurizer Replacement Project”, including procedures, work packages and load test records, to assure that they had been prepared and tested in accordance with regulatory requirements, appropriate industrial codes, and standards. The inspectors also reviewed polar crane inspection and maintenance records to assure it was in good condition.

The inspectors reviewed the licensee's analysis for buried piping located beneath the transport path as documented in SGT Calculation 0010003769-NL02-D-C02, Rev. 0, “Evaluation of Buried Utilities for RVCH/PZR Transport”.

The inspectors also reviewed the 10 CFR 50.59 Screening/Evaluation contained in the PC/M packages associated with the lifting and transportation program for the RVCH and PZR Replacement.

b. Findings

No findings of significance were identified.

.3 Pressurizer Replacement Inspection (IP 50003)

a. Inspection Scope

Design and Planning

The inspectors reviewed the following related to the licensee's pressurizer replacement project design and planning: the scope and schedule to identify special inspection needs; the Plant Change Modification (PCM) package; the 50.59 evaluation and Quality Assurance Program; the design and analysis for the creation of a temporary containment opening: the applicable engineering design, modification, and analysis associated with the pressurizer lifting and rigging; the radiation protection program controls, planning, and preparation associated with this PCM; the security considerations associated with vital and protected area barriers that were affected during the pressurizer replacement activities; and the controls and plans to minimize any adverse impact the activities may have had on the operating unit and common systems.

The inspectors reviewed the records associated with the fabrication of the replacement pressurizer and heater tubes manufactured in Chalon, France to verify compliance with Sections II, III, V, IX, and XI of the ASME Code.

The inspectors selected the upper shell and head of the pressurizer records contained in the Quality Assurance Data Packages for review, which included certificate of conformance to ASME code, N-stamped, certified material test reports (CMTR), chemical analysis, impact test, tensile strength test, drop weight test, mechanical test, fit-up, production weld data sheets, welding procedure specification, examination specification, preheat and post weld heat treatment, visual examination reports, liquid penetrant examination reports, magnetic examination reports, ultrasonic examination reports, hydrotest report, design drawings, and nonconformance report and repairs.

Removal and Replacement

During the pressurizer removal and replacement activities, the inspectors reviewed and evaluated the associated welding and nondestructive examination (NDE) activities (as listed below), the lifting and rigging of the pressurizer and associated equipment, and the radiological safety plans for temporary storage and disposal of the retired pressurizer and associated components. Major structural modifications, including activities associated with restoration of the temporary containment opening and leakage testing, controls for excluding foreign material, and the establishment of operating conditions including defueling, RCS draindown, and system isolation, were all areas inspected during the pressurizer replacement project.

The inspectors observed or reviewed records for the following welding, NDE examinations, preservice, and corrective action activities for the Class 1 piping and components of the pressurizer.

Welding

(1) Surge Line 12" Diameter

Weld 101 - Butt Weld, Elbow to Pressurizer Nozzle
Weld 102 - Butt Weld, Elbow to Pipe

(2) Spray Line 4" Diameter

Weld 119 - Butt Weld, Bottom Tee to Pressurizer Nozzle
Weld 175 - Butt Weld, Top Tee to Pipe

NDE

(1) Visual Examination (VT) & Liquid Penetrant Examination (PT)

Welds 107 and 108 - Socket Weld, Elbow or Orifice to Pipe, 3/4"-RC-169 Level Line
Welds 105 and 106 - Socket Weld, Elbow or Orifice to Pipe, 3/4"-RC-171 Level Line
Welds 103 and 104 - Socket Weld, Elbow or Nozzle to Pipe, 3/4"-RC-146 Sampling Line

(2) Radiographic Examination (RT)

Weld 101 - Butt Weld, Elbow to Pressurizer Nozzle, 12"-RC-108 Surge Line
Weld 102 - Butt Weld, Elbow to Pipe, 12"-RC-108 Surge Line
Weld 118 - Butt Weld, Reducer to Pipe, 2"-RC-146 Spray Line
Weld 119 - Butt Weld, Bottom Tee to Pressurizer Nozzle, 4"-RC-103 Spray Line
Weld 122 - Butt Weld, Tee to Pipe, 3"-RC-109 Spray Line
Weld 123 - Butt Weld, Valve to Pipe, 3"-RC-109 Spray Line
Weld 175 - Butt Weld, Top Tee to Pipe, 4"-RC-103 Spray Line

Preservice Examinations and Baseline Inspections

(1) PT

Weld 119 - Butt Weld, Bottom Tee to Nozzle, Spray Line 4"-RC-103
Weld 124 - Butt Weld, Elbow to Pipe, Spray Line 4"-RC-103
Weld 175 - Butt Weld, Pipe to Top Tee, Spray Line 4"-RC-103

(2) Ultrasonic Examination (UT)

Weld 101 - Butt Weld, Elbow to Pressurizer Nozzle, 12"-RC-108 Surge Line
Weld 102 - Butt Weld, Elbow to Pipe, 12"-RC-108 Surge Line

Weld 134 - Butt Weld, Pipe to PORC Nozzle, 4"-RC-101 PORC Line
Weld 176 - Butt Weld, Tee to Reducer, 4"-RC-103 Spray Line

Corrective Actions

- CR 2005-4205 Pressurizer Spray Line Repair
- CR 2005-5192 Weld Repair on the Tee of the Spray Line
- CR 2005-5271 Evaluation on Groove Mismatch of Safety Relief Valve

Post Installation Verification and Testing

The inspectors reviewed the post-installation verification and testing program to verify that the modifications were completed in accordance with the design, reviewed RCS leakage testing, evaluated containment testing, and verified pressurizer thermal and hydraulic performance.

The inspectors specifically reviewed licensee activities associated with the restoration of the construction hatch opening in the Steel Containment Vessel (SCV), as detailed in the licensee's Plant Change/Modification PC/M 04135 M, Removal and Reinstallation of Containment Building Steel Construction Hatch for Unit 1 RVCH/PZR Replacement Projects.

Activities associated with SCV welding were observed/reviewed and compared to the applicable codes (ASME Boiler and Pressure Vessel Code (B&PV), Section XI, 1989 Edition with no Addenda with exception of Subsection IWE for which the 1992 Edition with 1992 Addenda is applicable; Section VIII, 1968 Edition through winter 1968 Addenda), and FPL Specification SPEC-C-050.

The inspectors observed in-process welding activities for the new construction hatch weld, FW-007, including the control of welding materials. The inspectors reviewed the weld procedure, supporting procedure qualification record, and welder qualification records to confirm that the Code required essential and supplemental essential welding variables for Manual Shielded Metal Arc Welding were met. The inspectors reviewed the in-process work package, welding electrode receipt inspection and material certification records, qualification and certification records for nondestructive examination (NDE) personnel and NDE equipment and consumables. The welding electrode material certifications were compared to their appropriate specifications in ASME SFA 5.1 and SFA 5.01. The inspectors also observed the magnetic particle (MT) examination of the back gouge for weld FW-007. The inspectors reviewed examination records as specified below:

MT
FW-2007, Inside Containment Final MT
Pre and post MT exams for welded attachments
FW-2007. SCV Construction Hatch Barrel Weld Joint Prep, Containment Side and Annulus Side

RT
FW-2007, Final RT and Follow-up Repair

UT
Ultrasonic Thickness Report for SCV Construction Hatch

The inspectors also reviewed PC/M 04135M, Attachment 1 (10 CFR 50.59 Applicability Determination), and Attachment 7 (Report of Reconciliation for St. Lucie Unit 1 Steel containment Vessel Construction Hatch Repair) to verify that the modification was properly evaluated in accordance with 10 CFR 50.59.

b. Findings

No findings of significance were identified.

.4 Reactor Pressure Vessel Head (RPVH) Replacement (IP 71007)

a. Inspection Scope

The inspectors reviewed records related to the fabrication, testing, and inspection of the Unit 1 replacement RPVH and Control Element Drive Mechanisms (CEDMs) to verify compliance with applicable construction and inspection Codes (ASME Boiler and Pressure Vessel Code, Section III, 1989 Edition with no Addenda, and Section XI, 1998 Edition through 2000 Addenda) as defined in the Plant Change/Modification (PCM) Document PCM-04129, “RVCH Replacement Modification for Unit 1 RVCH Replacement Project.”

RPVH and CEDMs Housing Fabrication Records

The inspectors reviewed the following records associated with the fabrication of the replacement RPVH and CEDMs to verify compliance with Sections II, III, V, IX, and XI of the ASME Code:

Certified Material Test Report (CMTR) for the RPVH forging (Heat No. 03W77-1-1), including magnetic particle (MT) and ultrasonic (UT) examination reports after clad welding.

CMTR for CEDM adaptors BR/001 through BR/020 (Heat/Lot No. WP149).

CMTR for CEDM nozzles TD/001 through TD/025 (Heat No. WP140/Lot No. 01A), including liquid penetrant (PT) and UT examination reports after hydrostatic test, and heat treatment records according with procurement specifications.

CMTR for incore instrumentation (ICI) nozzles TD/100 through TD/109 (Heat No. RE529/Lot No.300230-02A), including PT and UT examination reports after hydrostatic test, and heat treatment records according with procurement specifications.

CMTR for ICI adaptors (Heat No. R1922-COA), including UT and PT examination reports.

Nonconformance Reports (NCRs) and Follow-up Documents.

- NCR-04/00412, Rev. 0, with regard to PT indications on the J-groove butter welds
- Followup Document CC/SL001-9010 associated with NCR-04/00412, Rev. 0, which covered the repair and PT reexamination of J-groove butter welds
- NCR-04/00410, Rev. 0, with regard to differences in the identification marks of CEDM adaptors and nozzles
- NCR-04/00401, Rev. 1, with regard to movement of thermocouples during heat treatment of material for CEDM adaptors
- CVAR 87-9001027-00, with regard to a deviation from the technical requirements of sulfur content in the material for ICI adaptors
- NCR-05/00454, Rev. 0, with regard to nonconformance of CEDM nozzles dimensions
- Followup Document CC/SL001-9120 associated with NCR-05/00454, Rev. 0, which covered the repair of dimensional deviations for CEDM nozzles in RPVH penetrations Nos. 38, 39, 42, 47-52, 54-64, 68, and 69
- Followup Document CC/SL001-9020 associated with NCR-04/00417, Rev. 0, which covered the repair of CEDM nozzle to CEDM adaptor weld in RPVH penetration No. 64

Hydrostatic Test Report CC/SL001 for the RPVH, including the chemical analysis report of the test water.

Final post weld heat treatment (PWHT) records for the RPVH, including time-temperature strip chart.

Framatome/Areva document (Doc. No. 5047583-00) for reconciliation of the replacement RPVH with the original construction Code.

In addition, for the following welds, the inspectors reviewed a sample of records for welding materials certification, NDE reports, welders performance qualifications, NDE material certifications, and NDE personnel qualifications as described below.

RPVH Clad weld W1002

- Production Weld Data Sheets
- CMTR for welding material, Heat Nos. 6A901, and 6V188
- Performance Qualification Records for one welder and two Level II UT/PT examiners
- Certification records of PT materials
- PT examination reports

RPVH J-groove butter welds

- Production Weld Data Sheets
- CMTR for welding material, Heat No. WC96F1
- Performance Qualification Records for one welder
- PT examination reports

RPVH J-groove welds

- Production Weld Data Sheets
- CMTR for welding material, Heat No. WC72F1

- Performance Qualification Records for one welder and four Level II PT examiners
- PT examination reports

CEDM nozzle to CEDM adaptor full penetration weld for RPVH penetration Nos. 1- 69

- Production Weld Data Sheets
- CMTR for welding material, Heat Nos. NX3167JK, and NX3900JK
- Performance Qualification Records for one welder
- Radiographic examination (RT) films and RT reader sheets for control rod drive housing (CRDH) weld Nos. CRDH-1, 5, 6, 10, 21, 38, 48, 49, 57, 58, and 64

ICI nozzle to quicklock adaptor full penetration weld for RPVH penetrations Nos. 70-77

- Production Weld Data Sheets
- CMTR for welding material, Heat No. NX9090JK
- Performance Qualification Records for one welder

Full penetration welds in CEDM upper pressure housing (upper weld designated as weld 1 and lower weld as weld 2), for housing Serial Nos. 5099, 5102, and 5106 (RPVH penetrations Nos. 43, 51, and 59, respectively)

- Welding Joint Log Reports
- CMTR for welding materials, Heat Nos. AT6289 and DT6396-Class 1
- Performance Qualification Records for two welders
- RT films and RT reader sheets

Full penetration welds in CEDM motor housing (upper weld designated as weld 3 and lower weld as weld 4), for housing Serial Nos. 5201, 5206, and 5233 (RPVH penetrations Nos. 43, 51, and 59, respectively)

- Welding Joint Log Reports
- CMTR for welding material, Heat No. NX0A80TS
- Performance Qualification Records for three welders
- RT films and RT reader sheets

CEDM adaptor to CEDM motor housing Omega seal weld for motor housing Serial Nos 5201, and 5206 (RPVH penetration No. 43, and 51, respectively)

- Weld Control Record Sheets
- CMTR for welding materials, Heat Nos. NX2424JK and T7635
- Performance Qualification Records for two welders and two Level II PT examiners

- Certification records of PT materials
- PT examination reports

Preservice Inspection (PSI) and Baseline Inspections

The inspectors reviewed selected NDE records, which documented the ASME Section XI PSI and baseline inspections performed to provide baseline conditions for future inspections in accordance with NRC Order EA-03-09.

Relative to ASME Section XI PSI of the replacement RPVH, the inspectors reviewed the following records:

- PT examination reports for the 32 peripheral Category B-O CEDM upper pressure housing welds (Welds 1 and 2) and motor housing welds (Welds 3 and 4)
- PT examination reports for the 8 peripheral Category B-O ICI housing welds
- A sample of UT examination reports consisting of CEDM upper pressure housing welds 1 and 2 for housing Serial Nos. 5104 and 5098
- A sample of UT examination reports consisting of motor housing welds 3 and 4 for housing serial Nos. 5341, 5343, and 5346
- Personnel certification records for three Level II (PT and UT) NDE examiners
- Certification records for PT materials used to examine the Category B-O welds listed above
- Certification records for UT equipment used to examine the Category B-O welds listed above

The inspectors reviewed the scope of the baseline inspections done in order to support future inspections required by NRC Order EA-03-09. Specifically the inspectors conducted the following activities:

- Review of UT examination reports for RPVH penetrations Nos. 2, 4, 10, 62, 65, 24, 30 (CEDM nozzles), and 70-77 (ICI nozzles)
- Review of J-groove surface ET examination report for RPVH penetrations Nos. 5, 8, 29, 47, 48, 51, 55, and 68
- Review of automated UT and manual ET procedures, including equipment specifications
- Review of personnel qualifications for five NDE examiners
- Review of under head PT examination reports of all nozzle to head J-groove welds using “PT white” acceptance criteria, including personnel qualifications
- Visual inspection of CEDM adaptor to CEDM motor housing Omega seal weld in RPVH penetrations 50, 43, and 51 (CEDM penetrations); and RPVH penetrations 70, and 71 (ICI penetrations)
- Visual inspection of J-groove weld in RPVH penetrations 59, 26, 48, 47, and 10 (CEDM penetrations); and penetrations 75, 71, 70, and 77 (ICI penetrations)
- Visual inspection of ICI nozzle to quicklock adaptor weld in RPVH penetrations 70-77

b. Findings

No findings of significance were identified.

.5 Concrete Shield Building Restoration Activities (IP 50003)

a. Inspection Scope

The inspectors examined restoration activities associated with the temporary construction opening (approximately 29 feet in diameter) in the Unit 1 concrete containment shield building wall, as detailed in the licensee's Plant Change/Modification (PC/M) Package 04-134M, Rev. 1, Removal and Restoration of Shield Building Concrete Construction Hatch for Unit 1 RVCH/PZR Replacement Project.

Relative to installation of concrete for the Unit 1 containment repair, the inspectors reviewed the commercial grade dedication plan for the concrete mix. The inspectors also reviewed vendor surveillance reports covering the licensee's inspection of the concrete batch plant and the concrete truck mixers to verify the trucks and batch plant complied with the guidance of the National Ready Mixed Concrete Association (NRMCA); that the batch plant scales were calibrated in accordance with NRMCA recommendations; and mixer efficiency tests were performed on the truck mixers in accordance with ASTM C-94. In addition, the inspectors reviewed the results of acceptance testing performed on materials (cement, fine and coarse aggregate, water, and admixtures) used for batching the concrete and records documenting post-placement inspection of concrete curing. The inspectors reviewed the concrete mix data to ensure that mix proportions for delivered concrete were selected based on trial concrete mix results, that QC acceptance criteria for the plastic concrete were based on the trail mixes, and that the trail mix met concrete strength requirements. Other records reviewed documented the results of in process tests performed on the plastic concrete (slump, entrained air, temperature, and unit weight), and results of unconfined compression tests performed on concrete test cylinders to verify concrete met design strength requirements.

b. Findings

No findings of significance were identified.

.6 Review of 10 CFR 50.59 Screening/Evaluation for the Replacement RPVH

a. Inspection Scope

The inspectors reviewed PCM-04129, "RVCH Replacement Modification for Unit 1 RVCH Replacement Project," including the associated 10 CFR 50.59 screening to verify that changes between the original RPVH and the replacement RPVH, and modifications resulting from installation of the replacement RPVH were properly evaluated in accordance with 10 CFR 50.59. The inspector's review included verification that the weight of the replacement RPVH, and its effect on reactor vessel supports and polar crane load capacity was taken in consideration as part of the design change.

b. Findings

No findings of significance were identified.

.7 Review of Quality Assurance (QA) Activities for the fabrication of the RPVH

a. Inspection Scope

The inspectors reviewed documentation to verify that the licensee implemented an adequate QA oversight of the manufacturer activities (Areva, France). The inspectors reviewed a sample of surveillance reports prepared by licensee's QA personnel at the vendor facilities, which covered three months of surveillance activities during the fabrication of the RPVH. The inspectors reviewed the activities indicated in the report and verified their adequacy.

In addition, the inspectors reviewed an audit conducted by the licensee as part of the Nuclear Procurement Issues Committee (NUPIC), which took place at two manufacturer's facilities and evaluated the adequacy of the manufacture's QA program in the following areas: Order Entry, Design, Software QA, Procurement, Fabrication and Assembly, Material Control, Handling, Storage and Shipping, Special Processes, Tests and Inspections, Calibration, Document Control, Organization and Program, Nonconforming Items/Part 21, Internal Audits, Corrective Action, and Training/Certification and Records. The inspectors reviewed the thoroughness of the audit and the impact of audit findings on the fabrication of the RPVH.

b. Findings

No findings of significance were identified.

.8 Reactor Vessel Head Replacement Radiation Protection Inspection

a. Inspection Scope

The inspectors reviewed and evaluated planning and implementation of radiological controls for ORVCH replacement activities.

Radiation, contamination, and airborne radioactivity surveys were reviewed for radiological work conditions and the adequacy of prescribed postings and HP controls. The inspectors reviewed RWPs and evaluated ED alarm settings, worker and HPT instructions, special dosimetry needs, and protective clothing/equipment requirements. ALARA planning, dose reduction initiatives, and actual doses received were reviewed and discussed with the RVCH replacement project and ALARA staff. The inspectors also reviewed plans for temporary storage of vessel head. A review of the training provided to workers is detailed in section 2PS2. Individual worker doses and project dose expenditures for the U1 ORVCH disassembly, removal from containment, packaging, and staging at the onsite temporary storage location were reviewed and discussed in detail with licensee representatives.

Radiological controls associated with the ORVCH prior to its removal from containment and subsequent to its placement in the onsite temporary storage facility were observed and evaluated. The inspectors observed implementation of radiological controls including posting, shielding, and air monitoring around the old ORVCH and selected components in containment. The inspectors also toured the onsite temporary storage facility and conducted independent surveys of the temporary storage environs and reviewed radiological controls and applicable environmental monitoring programs. Packaging used to ship the ORVCH was reviewed and plans for shipment of the ORVCH to a radwaste processing vendor were discussed with licensee staff.

RVCH replacement activities were evaluated against 10 CFR Parts 19 and 20, and approved licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in Sections 2OS1 and 2PS2 of the report Attachment.

The inspectors completed the specified radiation protection line-item samples detailed in IP 71007.

b. Findings

No findings of significance were identified.

.9 (Closed) Temporary Instruction (TI) 2515/161, Transport of Control Rod Drive (CRD) in Type A Packages

a. Inspection Scope

The inspectors reviewed shipping logs and discussed shipment of CRDs in Type A packages with shipping staff. The inspectors noted that no shipments of CRDs in Type A packages have been made since January 1, 2002. The only recent uses of a Type A package, i.e., shipments of in-core instrument cutting equipment, were reviewed and evaluated as part of routine inspection activities documented in Section 2PS2 of this report. The inspectors completed all the relevant inspection activities detailed in TI 2515/161.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

On January 4, 2006, the resident inspectors presented the inspection results to Mr. Bill Jefferson and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

Interim Exit Meetings

Interim exit meetings were conducted for:

- IP 50003 - Welding of SCV Construction Hatch on November 22, 2005, and Lifting and Transportation Program and Design Review of Activities Associated With the Pressurizer Replacement on December 2, 2005. The licensee confirmed that none of the potential report input discussed was considered proprietary.

- Baseline procedure 71111.08 on November 4, 2005. The licensee confirmed that none of the potential report input discussed was considered proprietary.

4OA7 Licensee Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispostioned as NCV's.

- Technical Specification 3.3.1.1 and 3.3.2.1 requires that reactor protective system instrumentation channels be placed in trip if not operable within one hour. Contrary to this, steam generator water level instrument LIC-9023A was inoperable on February 19, 2005, and was not placed in the bypassed or tripped condition until March 22, 2005. This issue has been captured by the licensee's CAP as CR 2005-8049. The finding is of very low safety significance because it does not represent an actual loss of a safety function.

- 10 CFR Part 20.1902(b) requires each HRA to be conspicuously posted "Caution, High Radiation Area" or "Danger, High Radiation Area". Contrary to the above, from November 18, through November 20, 2005, the U1 Volume Control Tank (VCT) cubicle was not posted as a HRA even though general area dose rates exceeding 100 mrem in one hour existed in the area. Specifically, at 19:20 hours (hrs) on November 18, an HPT down-posted the VCT cubicle from an HRA to a radiation area based on limited surveys. At 22:00 hrs on November 20, an HPT conducting routine pre-job surveys in the VCT cubicle identified the posting discrepancy and took immediate corrective actions. Licensee personnel entered and worked in the VCT cubicle while the area was posted improperly. This event is documented in the licensee's CAP as CR 2005-32778. Although this event involved failure to maintain proper controls for an HRA, this finding is of very low safety significance because workers in the VCT cubicle were on appropriate RWPs, had proper dosimetry, and no unexpected/ unintended radiation exposures occurred.

ATTACHMENT: SUPPLEMENTAL INFORMATION

Supplemental Information

KEY POINTS OF CONTACT

Licensee Personnel

L. Neely, Work Control Manager
R. De La Espriella, Security Manager
C. Buehrig, Maintenance Rule Coordinator
D. Calabrese, Emergency Planning Supervisor
C. Costanzo, Operations Manager
E. Armando, Site Quality Manager
L. Edwards, Training Manager
K. Frehafer, Licensing Engineer
R. Hughes, Site Engineering Manager
E. Katzman, Performance Improvement Department Manager
G. Johnston, Plant General Manager
W. Jefferson, Site Vice President
R. McDaniel, Fire Protection Supervisor
W. Nurberg, Chemistry Manager
W. Parks, Operations Supervisor
M. O'keefe, Acting Licensing Manager
J. Porter, Operations Support Engineering Manager
G. Swider, Systems Engineering Manager
J. Tucker, Maintenance Manager
S. Wisla, Health Physics Manager
D. Nowakoski, NDE Level III
E. Avella, Manager Projects
M. Moran, Pressurizer Replacement Engineering Project Manager
T. Coste, ISI Coordinator
B. Moss, BACCP Manager

NRC Personnel

B. Moroney, NRR Project Manager
S. Ninh, Region II Project Engineer

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Open/Closed

5000389/200505-01	NCV	Failure to Control Tooling for Use Only on Stainless Steel (Section 1RO8).
05000335/2005005-02	NCV	Failure to Accomplish Prescribed Procedure Steps Resulting in Starting the 1B LPSI Pump With its Suction Valve Closed (Section 1R20).
5000389/200505-03	NCV	Loss of FME Integrity When Material Was Found in RCB Which Was Not on the FME Log (Section 1R20).
05000335, 389/2005005-04	NCV	Failure to Implement Required HP Controls for U1 Pressurizer Valve Work (Section 2OS1).
05000335, 389/2005005-05	NCV	Failure to Implement Appropriate DOT Type A Package Closure Requirements (Section 2PS2).
05000335/2005005-06	NCV	Failure to take timely and effective corrective actions to control hot work activities while being planned or performed to prevent recurrence (Section 4OA2).

Closed

05000335/2005003	LER	1A Emergency Diesel Generator Actuation (Section 4OA3).
05000389/2004004	LER	Manual Reactor Trip Due to Condensate Pump Motor Malfunction (Section 4OA3).
05000335/2005001	LER	Condition Prohibited by Technical Specifications (TS) due to Inoperable Reactor Protection Instrumentation (Section 4OA3).
05000335/2515/161	TI	Transport of Control Rod Drive (CRD) in Type A Packages (Section 4OA5)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1R08, Inservice Inspection (ISI) Activities

Nondestructive Examination
- Welding Procedure Specification (WPS) - 43, Revision 10
- ISI-PSL-1-Program, St. Lucie Nuclear Power Plant Unit 1 - Third Inservice Inspection Interval Program Plan, Revision 4
- ISI-PSL-1-Plan, St. Lucie Nuclear Power Plant Unit 1 - Third Inservice Inspection Interval Plan
- NDE 5.4, Ultrasonic Examination of Austenitic Piping Welds, Revision 17
- NDE 5.2, Ultrasonic Examination of Ferritic Piping Welds, Revision 13
- NDE 4.2, Visual Examination VT-2 Conducted During System Pressure Tests, Revision 9
- ADM-29.03, Boric Acid Corrosion Control Program, Revision 4 and Revision 5

Corrective Action Documents (Condition Reports (CR))
- CR 2005-28797, Steam Leak on instrument valve downstream of V08105, Primary Root Valve Downstream of FT-08-1A.
- CR 2004-6056, Operating Experience Assessment: Review issues associated with the Mihama-3 pipe rupture of August 9, 2004, for applicability to St. Lucie and Turkey Point
- CR 2005-29636, FAC Component 12ES1-P-7-16 (Loc. 13) - Degradation in HP Extraction Steam feedwater heater 5A
- CR 2005-28909, Dry Boric Acid on tubing at VFT-1199
- CR 2005-28198, Boric Acid Issues. Discovered during NOP/NOT walkdowns
- CR 2005-20547, BACC Program Review
- CR 2005-3757, Tracking CR for development of continuing training for the Boric Acid Corrosion Control Program
- CR 2005-28880, Loss of Power to SMAW Ovens during Hurricane Wilma
- CR 2005-28906, Control of Tooling for Use on Stainless Steel Material

1R20 Refueling and Other Outage Activities

Procedures
- ADM-27.13, Foreign Material Exclusion, Revision 3
- ADM-09.05, Containment Entries Mode 1-4, Revision 10

Corrective Action Documents (Condition Reports (CR))
- CR 2005-29019, FME Log Book Missing Pages
- CR 2005-28985, Several Tools Used To Support Weld Repair (V-08105) Found in Unit 2 RCB Without Corresponding FME Log Sheets/Entries
- CR 2005-29068, Management Containment Close-Out Inspection Identifies Welding Lead Adapter That Was Not Entered as Single Component During FME Area Entry

Section 2OS1: Access Controls to Radiologically Significant Areas

Procedures, Manuals, and Guidance Documents

Administrative Procedure (ADM) -05.02, HP Controls of Spent Fuel Pool Non-SNM, Revision (Rev.) 1B
Health Physics Procedure (HPP) -3 High Radiation Area, Rev. 17A
HP-74, Access Control Using Alarming Dosimeters, Rev. 6
HP-112, Multibadging, Rev. 21B
HPP-22, Air Sampling, Rev. 16A
HPP-20, Area Radiation and Contamination Surveys, Rev. 24
HPP-30, Personnel Monitoring, Rev. 36
HPP-70, Personnel Contamination Monitoring, Rev. 23
HPP-72, Determination of Dose to the Skin from Skin Contamination, Rev. 5A
Radiation Protection Instruction, Reactor Vessel Closure Head (RVCH) Replacement Radiation Protection Controls, dated 10/08/05
Radiation Protection Instruction, Move In/Move Out Major Components and Waste, dated 10/11/05
Nuclear Administrative Procedure (NAP) - 204, Condition Reporting, Rev. 6
Radiation Work Permit (RWP) 05-710, U1 RCB 62' Top of PZR & 1100 E&F Platform (LHRA), Rev. 1
RWP 05-517, U1 RAB and RCA (HRA), Rev. 0
RWP 05-524, U1 RAB/RCA/Decon Room/Hot Machine Shop/U2 Test Bench (HRA), Rev. 1
RWP 05-1028, U1 RCB 62', 45', 23', 18' Ele. All Areas (HRA Access Allowed)

Records and Data

Unit 1 (U1) / Unit 2 (U2) High Radiation Area (HRA) and Locked High Radiation (LHRA) Locations, September 28, 2005
Personnel Contamination Event Data Lists: Calendar Year (CY) 2004 and January 2005 through November 30, 2005
CY 2004 U2 Refueling Outage 15 Internal Dose Assignments from Positive Whole Body Counts/ Airborne Radioactivity Area
Plant St Lucie (PSL) Maximum Total Effective Dose Equivalent (TEDE) Results: CY 2004 and January 2005 through October 2005
HPS-64 Radiation Survey, Boric Acid Pre Concentrator Filters - EL-0.5 Ft, 05/16/05
HPS-64 Radiation Survey, Concrete Plugs Located at Temporary Storage Facility, 11/03/05
HPS-64 Radiation Survey, Controlled Area Boundary Survey, 11/04/05
HPS-64 Radiation Survey, Temporary Storage Area, 11/06/05
HPS-64 Radiation Survey, U1 Containment Inside Construction Service Hatch with Reactor Head Laying Flat, 11/07
U1 ORVCH Smear and Dose Rate Surveys including: Interior Gamma Survey, Interior Smear, CRDM Vertical and Azimuthal Gamma Survey11/06/05
HPS-64 Old RVCH Storage Location, 11/07/05
HPS-109 Survey, U1 Pressurizer, 11/08/05
HPS-334 Survey, U1 Pressurizer Prior to Release from Containment, 11/08/05, 11/09/05,
HPS-64, Dose Rate Verification of Boundaries After Old Pressurizer Placed in Shielded Area, 11/09/05

HPS-64 Survey, St. Lucie Owner Controlled Area Perimeter, 11/12/05
HPS-64 Survey, Original Reactor Vessel Head Laydown Area, 11/12/05
Survey 2005-1998, U1 CTMT 62' PZR, 11/26/05
Survey 2005-2066, U1 CTMT 62' PZR Valve #1249, 11/27/05
Survey, U1 VCT Area El. 19.5', 11/18/05
Survey 2005-1709, U1 VCT Area El. 19.5', 11/20/05
Survey 2005-1707, U1 VCT Piping in VCT Room El. 19.5', 11/20/05
Temporary Shield Package (TSP) No: 02-035, Attachment 1, Temporary Shielding Placement Form, Unit 1 BA Pre-Conc At Center Entrance Door Overhead Line Between H/U to Spent Resin Tank Check & F16650; Pre and Post Shielding Surveys HPS-6 Surveys of Boric Acid Pre Concentrator Filters - EL-.5' conducted 09/17/02 and 09/19/02
RCA Exit Transaction Records, 11/18/05 - 11/20/05
Form HPP-20.1, HRA Postings Change Checklists, 11/18/05 and 11/20/05
Form HPP-30.12, Radiation Exposure Extension Requests, U1 Pressurizer Replacement Project
EPD Dose/Dose Rate Alarm Logs, 1/1/05 - 11/30/05
Form HPP-70.1, Personnel Skin and Clothing Contamination Reports, 11/21/05 and 11/22/05

Corrective Action Program (CAP) Documents

Plant Saint Lucie (PSL) Daily Quality Summary, Transport of the Original Reactor Vessel Closure Head (ORVCH); ORVCH Travel and Laydown; U1 Pressurizer Removal Preparations 11/07/05
PSL Daily Quality Summary, ORVCH Canister Assembly, Pressurizer Removal; 11/09/05
Condition Report (CR) 2005-32859, Workers Contaminated While Inspecting Pressurizer Valve V1249, 11/25/05
CR 2005-32778, U1 VCT Cubicle Inappropriately Down-Posted from HRA to RA, 11/18/05
CR 2005-1800, HP technician discovered LHRA dose rates on the 2A S/G hotleg drain valve, 1/18/05

Section 2OS2: As Low As Reasonably Achievable

Procedures, Manuals, and Guidance Documents

ADM 05.01, ALARA Program, Rev. 9
ADM 05.04, Cobalt Reduction Program, Rev. 0B
HP-55, Portable Shielding, 17B
HPP-23, Health Physics Activities in the Reactor Containment Building during Shutdown, Rev. 17
HPP-38, Surveys for Chemical Crudburst and Cleanup of Reactor Coolant System, Rev. 1
St Lucie Nuclear Plant Dose Reduction Plan 2003 - 2007, Rev. 08/01/05
Temporary Procedure (TP) - 04, Saint Lucie Nuclear Power Station, Unit One Reactor Head Replacement ALARA Plan, 10/06/05
TP-05, Saint Lucie Nuclear Power Station, Unit One Pressurizer Replacement ALARA Plan, 10/06/05
AREVA Total Estimated Exposure Data for PSL 1 Reactor Head Disassembly/Reassembly, 10/08/05
AREVA Total Estimated Exposure Data for PSL 1 Old Reactor Head Disassembly and New

Head Assembly in Containment, 08/21/05
AREVA Total Estimated Exposure Data for PSL 1 Pressurizer Replacement, Maximum Shield Package and Cleaning Out Piping in Place, 08/16/05; Re-Evaluation Data 11/01/05
AREVA Total Estimated Exposure Data for PSL 1 Containment/Spent Fuel Pool Refueling (Core Off Load & Reload 10/08/05
AREVA Total Estimated Exposure Data for PSL Project Activities, 10/08/05
RWP 05-700, Unit 1 (U1) Reactor Containment Building (RCB) - Reactor (Rx) Head, Rx Cavity, (Locked High Radiation Access [LHRA] Allowed), Health Physics Job Coverage & Surveys in Support of Reactor Vessel Closure Head (RVCH) Removal & Replacement & Refueling Activities, Rev. 0
RWP 05-701, U1 RCB 62 foot ('), 23', 18' Pressurizer (PZR) (Upper/Lower) 1100 E&F (LHRA Access Allowed), HP Job Coverage, Surveys in Support of PZR Removal/Replacement, Rev. 0
RWP 05-704, U1 RCB All Areas to Include 62 ' PZR, Rx Head Area, & Rx Cavity (High Rad Area) Decon Support for Rx Head & PZR Replacement Project, Includes Shielding, Transport of High Radiation Trash and Components, Rev. 0
RWP 05-706, U1 RCB PZR All Elevations. (HRA Access Allowed), Cut Out & Replace Pressurizer Vessel: Includes, Pipe Cuts, Welding, Removing & Replacing Pipe Supports / Hanger, Coatings Application & Photogrammetry, Rev. 0
RWP 05-707, U1 RCB PZR All Elevations (High Radiation Area Access Allowed) Install Scaffolding in Support of PZR Replacement Project, Rev. 0
RWP 05-708, U1 RCB PZR All Elevations (HRA), Remove/Install Insulation for PZR Replacement, Rev. 0
RWP 05-710, U1 RCB 62 foot Top of PZR and 1100 E&F Platform (LHRA), Perform Hydrolance of PZR Piping in Preparation of PZR Vessel Removal, Includes Valve Removal and Replacement of the Vales Listed in the ALARA Section of this RWP, Also to Include Insulation Removal and Shielding on V-1100E&F, Rev. 0
RWP 05-711, U1 RCB PZR All Elevations (LHRA Access Allowed) Cut Out and Replace Pressurizer Vessel Surge Line: Includes Pipe Cuts, Welding, Removing & Replacing Pipe Supports/Hangers & Bottom Side Hydrolance, Rev. 0
RWP-05-714, Temporary RCA (HRA Access Allowed), Support OPZR and ORVCH Changeout and Storage, Shielding, Packaging, Includes HP Support, Rev. 0
RWP-05-715, U1 Outside RCB, Construction Service Platform, Yard & Temporary RCA, Remove from RCB and Package Old Rx Head and Old PZR for Transport (Includes Offload) Move Replacement Pressurizer & RVCH into RCB, Rev. 0
RWP-05-717, U1 RCB 62' Elevation (LHRA Access Allowed), Disassemble ORVCH Includes Shroud and Superstructure Removal, O-Ring Cut Up and Removal, Coating Application, Shield Wall Shielding & Other Support Work, Rev. 0
RWP 05-1416, U1 RCB 18' RCS Hot Legs (HRA Access Allowed), Remove, Replace, Repair RTD Nozzles, to Include Machining, Cutting, Grinding, and Welding, Rev. 0

<u>Records and Data</u>

Plant St Lucie Unit 1 (PSL 1) - Refueling Outage 20 Daily ALARA Report Data, 11/01-12/04/2005
PSL-1 - Daily Outage RWP Report Data, 11/01-12/04/05
PSL Non-Outage, General Entry RWP Report Data, 11/01/05 and 11/27/05
PSL-ENG-SEMJ-05-062, Temporary Onsite Storage of the Original Pressurizer and the

Original Reactor Vessel Closure Head, Rev. 1
St Lucie Nuclear Plant, Dose Reduction Plan, 2003-2007, Rev. 08/01/05
2005 St Lucie Unit 2 Cycle 15 - ALARA Report, Refueling Outage Summary, 3/4/05
2004 St Lucie Unit 1 Cycle 19 - ALARA Report, Refueling Outage Summary, 4/2/04
ADM-05.01, Appendix B, ALARA Re-evaluation Form Data for: RWP No. 05-701, 11/20/05; RWP No. 05-704, 11/20/05; RWP No. 05-706, 11/20/05; RWP No. 05-708, 11/20/05; RWP No. 05-723, 11/20/05; RWP No. 05-710, 11/5/05; RWP No. 1006, 11/28/05; RWP 05-1008, 11/28/05; RWP No. 1011, 11/28/05; RWP No. 05-1030, 11/04/05; RWP 05-1405, 11/11/05

Corrective Action Program (CAP) Documents

PSL Condition Report Data Base Results Sorted on ALARA: October 1, 2004 through September 17
2005 ALARA Self-Assessment, St Lucie Nuclear Power Plant, 06/20-24/05
CR 2004-7065, Poor ALARA Practice by Reactor Crew for Stud Nut Removal and Cleaning, 08/25/04
CR 2005-0877, Lack of Formal Process for Delivering ALARA Board Decisions to Work Groups, 01/11/05
CR 2005-1365, Failure to Use Power Tools on Reactor Head Duct Work Removal, 01/14/05

Section 2PS2: Radioactive Material Processing and Transportation

Procedures, Manuals, and Guidance Documents

St. Lucie Plant Process Control Program, Administrative Procedure 0520025, Rev. 13A
HP-40, Shipment of Radioactive Material, Rev. 52
HP-45, Packaging of Dry Active Waste Into Bulk Containers, Rev. 5B
HP-47, Classification of Radioactive Waste Material for Land Disposal, Rev. 27A
HP-48, Activity Determinations for Radioactive Material Shipments, Rev 6B
HP-49, Dewatering Radioactive Bead Resins, Rev. 12
HP-49A, Transfer of Radioactive Bead Resins, Rev. 20
HP-53, Transfer of Plant Process or Tri-Nuc Filters and High Dose Rate Radioactive Waste, Rev. 9
HPP-42, Identification, Survey, and Release of Material, Rev. 3
HPP-45, Packaging of Dry Active Waste into Bulk Containers, Rev. 5B
Mechanical Maintenance Procedure Unit 2 (2-MMP)-06.02, Radioactive Filter Cartridge Replacement, Rev. 6
Operating Procedure 2-0540020, Boron Recovery System Lineup, Rev. 25B

Records and Data

St. Lucie RPT-C Training and Qualification Summary Printout, printed on 12/30/2005
Work Order (WO) Package, Task No. 2001477501, Replace U2B Boric Acid Pre-Concentrator Filter, 01/23/1990
WO Package, Task No. 3200338401, Replace U2A Boric Acid Pre-Concentrator Filter, 02/19/2002
St. Lucie Plant Annual Effluent Release Report January 1, 2003 - December 31, 2003

St. Lucie Plant Annual Effluent Release Report January 1, 2004 - December 31, 2004
Framatome ANP Environmental Laboratory 10 CFR Part 50/61 Analysis Report for DAW Waste Stream sampled on February 11, 2005
Framatome ANP Environmental Laboratory 10 CFR Part 50/61 Analysis Report for Secondary Resin Waste Stream sampled on October 25, 2004
Framatome ANP Environmental Laboratory 10 CFR Part 50/61 Analysis Report for U1 Spent Resin Tank Bead Resin shipment sampled on July 20, 2004
Framatome ANP Environmental Laboratory 10 CFR Part 50/61 Analysis Report for Old Reactor Head sampled on August 19, 2004
Framatome ANP Environmental Laboratory 10 CFR Part 50/61 Analysis Report for Old Pressurizer sampled on October 25, 2004
Evaluation of PSL Dry Active Waste Stream Data and Recommendations, including RADMAN Data Base Comparison Reports for Scaling Factors, Sample Abundance, and Data Values, dated 11/28/2005
Evaluation of PSL 2 SRT Resin Waste Stream Data and Recommendations, including RADMAN Data Base Comparison Reports for Scaling Factors, Sample Abundance, and Data Values, dated 10/03/2004
U-2 SRT Inventory from Radwaste Log, 06/23/1998 - 08/05/2005
U-1 SRT Inventory from Radwaste Log, 08/06/2002 - 07/28/2005
Container Products Corporation Certificate of Compliance and Final Inspection Report for DOT 7A Type A Container, Serial No. 208756, 05/23/1997
Shipment No. 03-54, Surface Contaminated Object (SCO), Incore cutter, 10/03/2003
Shipment No. 04-43, Low Specific Activity (LSA), DAW, 05/03/2004
Shipment No. 05-23, Type A Package, Incore cutter, 02/16/2005
Shipment No. 05-37, Type B Package, Spent Filters, 04/14/2005
Shipment No. 05-62, LSA, spent resin, 08/09/2005
HP-107, Survey of Dry Waste Storage Building - El 19.5 Ft., 11/01/2005

<u>CAP Documents</u>

FPL Nuclear Division Quality Assurance Audit Report, Radiation Protection Functional Area Audit QSL-RP-03-04, June 24 - August 11, 2003
FPL Nuclear Division Quality Assurance Audit Report, Radiation Protection Functional Area Audit QSL-RP-05-07, June 8 - August 1, 2005
St. Lucie HP Program Radioactive Material Control Assessment Report, May 24 - June 04/2004
CR 03-2793, Two packages shipped to PTN were missing the required exception notices upon receipt, 07/13/2003
CR 2004-5208, A review of the Frequently Asked Questions About Health Physics Based on 10 CFR Part 20 (Q&A # 53) revealed an opportunity for improvement, 07/30/2004
CR 2005-5530, The contact dose rate on a container of Radioactive material shipped from PSL to the Ft Calhoun Nuclear Station was found to be higher than reported on the shipping papers, 02/18/2005
CR 2005-25727, Evaluate package closure requirements for DOT 7A Type A packages, 09/20/2005
OE19531, Lid Bolt Torque Requirement Missed for Type A Control Rod Drive Shipping, 11/18/2004

Section 4OA1: Performance Indicator Verification

Procedures, Manuals, and Guidance Documents

ADM-25.02, NRC Performance Indicators, Rev. 13A

Records and Data

EPD Dose/Dose Rate Alarms: YDT 2005
Occupational Exposure Control Effectiveness Data Sheets: 4th Quarter CY 2004 through 2nd Quarter CY 2005
RETS/ODCM Radiological Effluent Occurrence Data Sheets: 4th Quarter CY 2004 through 2nd Quarter CY 2005
U1 & U2 Gaseous Effluent I-131,I-133, Tritium & Particulate Dose Report, 11/09/05
U1 & U2 Noble Gas Effluent Air dose and Projected Dose Report, 11/09/05
U1 & U2 Liquid Effluent Dose Summation, 11/09/05
U1 Liquid Release Permit No. 1-04-83
Operations Department Chronological Logs Unit 2 11/23/05, 11/25/05
Equipment Run Time Log - PSL- 2; 11/19/-25/05 for SGBTF Exhaust Fans (HVE-41A/41B)

CAP Documents

CR 2004-8038, Elevated Dose Rates in Both Unit 1 and Unit 2 Reactor Auxiliary Building, 09/12/04
CR 2005-1707, High Radiation Area Created During Spent Fuel Transfer, 01/18/05
CR 2005-21572, Transfer of Resin from Spent Resin Tank Resulted in the Generation of a High Radiation Area in the Unit 1 Drumming Room, 08/05/05
CR 2005-1399, Grating (Diamond Plate) Removed from Reactor Sump Creating a Posting Inconsistency, 01/14/05
CR 2005-1800, Locked High Radiation Area on the 2A Steam Generator Hotleg Drain, Valve No. 1214
CR 2005-19648, Several Keys Were Assigned to a HPSS /VHRA in the Security Data-Base, but could not be Readily Located During a QA Audit of Health Physics, 07/14/05
CR 2005-22365, Following Unit 2 Startup, Dose Rates in the Letdown Cubicle Were Found to be 2 R/hr with a teletector in the Vicinity of LCV-210Q, 08/13/05

Section 4OA5, Other

Review of Heavy Lifting

d. Procedure 1-3011, Rev. AFU, Transport of the Replacement Reactor Vessel Closure Head
e. Procedure 1-3330, Rev. AFU, Rig and Handle the Replacement Reactor Vessel Head
f. Procedure 1-3016, Rev. AFU, Transport of Pressurizer from Temporary Laydown Area to Reactor Containment Building
g. Procedure 103340, Rev. AFU, Rig and Handle the Replacement Pressurizer
h. QEP 10.05, Rev. 1E1, Rigging and Handling
i. Procedure AP 0010438, Rev. 41, Control of Heavy Loads

j. SGT Work Order (WO) 35008564-02, Transport New Reactor Head to Containment via Haul Route
k. Work Package Master Index for Package 1-3011
l. WO 35009300-02, Transport for New Pressurizer to Containment via Haul Route
m. WO 35007472-01, Rig and Install the New Pressurizer
n. Mammoet Dwg. 0010003769-NL02-D-H04, Rev. 01, Rigging Arrangement Old PZR/Replacement PZR for Outside Lifting System (OLS)
o. Mammoet Dwg. 0010003769-HL02-D-H08, Rev.00, Hauling Sequence Replacement RVCH (Sheets 1 to 6)
p. Plant Change & Modification (PC/M) Packages PC/M No. 04136M, Rev. 0, Outside Service Platform and Outside Lifting System for Unit 1 RVCH/PZR
q. PC/M No. 04137M, Rev. 0, Hatch Transfer System and Heavy Rigging for Unit 1 RVCH/PZR
r. SGT Calculation No. 0010003769-NL02-D-C02, Rev. 0/AFU, Load and Function Test Procedure
s. PC/M No. 04138M, Rev. 0, RVCH Offload & Transport for Unit 1 RVCH Replacement Project
t. PC/M No. 04152M, Rev. 0, Pressurizer Offload and Transport for Unit 1 Pressurizer Replacement Project
u. Polar Crane Inspection and Maintenance Records
v. 10 CFR 50.59 Screening for PC/M

<u>Pressurizer Replacement</u>

w. Plant Change/Modification (PC/M) No.04149, Rev. 2, Replacement Pressurizer Components for Unit 1 Pressurizer Replacement Project
b. Specification SPEC -—099, Rev. 3, Project Specification for Replacement Pressurizer
c. Specification SPEC-C-057, Rev. 1, Technical Specification for the Pressurizer Installation at St. Lucie Unit 1
d. Areva NDE Procedure 54-PT-200-04, Rev. 4, Color Contrast Solvent Removable Liquid Penetrant Examination of Components
e. Areva Procedure 55-SPP02-09, Rev. 9, General Procedure for Arc Welding - including VT
f. Areva Procedure 55-WCP08-07, Rev. 7, Welding Visual Inspection
g. Areva NDE Procedure 54-RT-1027734, Rev. 8, Radiographic Testing
h. FPL Procedure NDE 5.4, Rev. 17, Ultrasonic Examination of Austenitic Piping Welds
i FPL Procedure NDE 5.23, Rev. 2, Ultrasonic Examination of Welds Adjoining Cast Materials
j. Procedure ADM-27.13, Rev. 3, Foreign Material Exclusion
k. FPL Pressurizer Manufacturer Report, PSL 1 RPRZ QADP Doc. 23-9003280-003 Volume 1, & 8
XXIV. QADP Pressurizer Volume 18, Data Package No. 23-9001574-001, Rev. 001, Pressurizer Heater Element Assembly Quality Assurance Data Package
m. 10 CFR 50.59 Screening/Evaluation for PC/M No. 04149
n. Areva WO 35007564-01-SL1-20, Re-Installation Piping at Top Pressurizer for MEP-4150
o. OWO 35007562-01, Re-Installation Pressurizer/ Piping at Bottom Pressurizer
p. Condition Report (CR) 2005-28126, 27815, 4205, 5192, and 5271
q. NDE Reports for VT, UT, PT, and RT

r. Dwg. BUMPSL/NPR1801, Rev. C, Replacement Pressurizer for St. Lucie Plant Unit 1
s. Dwgs. 5052895 to 5052902, Pressurizer Top and Bottom Pipe Installation

Containment Restoration Activities

PC/M No. 04135, Removal and Reinstallation of Containment Building Steel Construction Hatch for Unit 1 RVCH/PZR Replacement Projects, Revision 2

Ebasco Specification FLO-8770-757, "Steel Containment Vessel," Revision 6

QEP 20.04, Welder Performance Qualification, Revision 3

WPS No. SM/1.1-2, Manual Shielded Metal Arc Welding (SMAW), Revision 1

PQR No. GT-SM/1.1-Q6, Procedure Qualification Record, Revision 1

WDC No. 1-3720-01, In-process Weld Data Card for FW-2007

NCR 01-115, Two drilled holes in weld prep area barrel side of construction hatch

NCR 01-109, Leak Channel remnants on Construction Hatch / Containment Side

Specifications & Procedures

Specification No. 7012-SPEC-C-007, Concrete and Grout Specification (Excluding Turkey Point Containment Opening Work) Rev. 1

EBASCO Specification No. FLO 8770.474, Concrete Reinforcing Steel Fuirnishing, Fabrication, and Delivery, Rev. 1

Plant Change/Modification (PC/M) Package 04-134M, Removal and Restoration of Shield Building Concrete Construction Hatch for Unit 1 RVCH/PZR Replacement Project

Specification CN-2.9, Specification for Concrete Materials and Mixes, Concrete Mixing, and Transportation, Rev. 4

Specification CN-2.11, Specification for Concrete Testing, Placing, Curing, and Finishing, Rev. 6

FPL Drawing 8770-G-051, Reactor Building Cylinder Wall - Plans and Sections, Rev. 10

CRN 04134-13033, Rebar Installation Tolerances

CRN 04134-13013, Rebar Lap Splices

Quality Records

Commercial Grade Dedication Plan for concrete mix for shield building repair

Vendor Surveillance Reports performed on Rinker concrete and concrete trucks

Concrete mix design data (Rinker mix design Mix Code 1211599)

Concrete mix design qualification test results performed by Soils and Materials Engineers, S&ME, Inc

S&ME test equipment calibration records

Result of testing performed on concrete materials: Type I cement (ASTM C-150), water reducer admixture WRDA-60, high range water reducer ADVA 120, fine aggregate (ASTM C-33), number 57 coarse aggregate (ASTM C-33), Ice, and batch plant water

Concrete placement records which included the pre-pour check list, the concrete pour card, concrete batch tickets, and the results of testing performed on the plastic concrete (slump, air content, temperature and unit weight) at the batch plant and point of placement.

Post pour inspection report

Reactor Pressure Vessel Head (RPVH) Replacement

Quality Assurance Data Package (Doc. # 23-5072592-00) - "QADP Saint Lucie 1 Reactor Vessel Head Replacement," Volumes 1 - 8

Quality Assurance Data Package (Order # 000673) - "CEDM Pressure Housing Assembly"

Plant Change/Modification Document - PCM 04129, RVCH Replacement Modification for Unit 1 RVCH Replacement Project, including 10 CFR 50.59 Applicability Determination and 10 CFR-50.59 Screen

Framatome ANP Certified Design Specification 08-5028068-03, Reactor Vessel Closure Head Replacement Florida Power and Light St. Lucie Unit 1

St. Lucie Unit 1 Replacement RV Closure Head Reconciliation Document 5047583-00

JQA-04-282, NUPIC Joint Audit 08.03.FANFR.04.1, dated July 15, 2004

Report No. 08.06.FANFR.05.7, QA Surveillance Report for July 1-31, 2005

Report No. 08.06.FANFR.05.8, QA Surveillance Report for August 1-31, 2005

Report No. 08.06.FANFR.05.9, QA Surveillance Report for September 1-30, 2005

WESDYNE Examination Program Plan WDI-PJF-1303240-EPP-001, Examination of CEDM Similar and Dissimilar Metal Wels for PF&L St. Lucie 1

St. Lucie Unit 1 RVCH Replacement Baseline NDE Final Report (EIR Report No. 51-9005256-000)

Quality Assurance Data Package (Doc. # 23-5072592-00) - “QADP Saint Lucie 1 Reactor Vessel Head Replacement,” Volumes 1 - 8

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St. Lucie Unit 1 Replacement RV Closure Head Reconciliation Document 5047583-00

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