UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

July 26, 2004

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 50-335/04-04 AND 50-389/04-04 AND OI SYNOPSIS

Dear Mr. Stall:

On June 27, 2004, 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 July 8, 2004, 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. In addition, this report includes the results of an investigation completed by the NRC's Office of Investigations (OI). The investigation involved a review to determine if a senior reactor operator (SRO) deliberately failed to comply with license conditions. Based on the evidence developed during this investigation, the allegation was not substantiated. A copy of the synopsis to the OI report is included as Enclosure 2 to this letter.

Based on the results of this inspection, one inspector identified and two self-revealing findings of very low safety significance (Green) were identified. These findings were determined to involve violations of NRC requirements. 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 Senior 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 enclosures 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 1. Inspection Report 50-335/04-04, 50-389/04-04 w/Attachment - Supplemental Information
2: OI Synopsis

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.:	50-335/04-04, 50-389/04-04
Licensee:	Florida Power & Light Company (FPL)
Facility:	St. Lucie Nuclear Plant, Units 1 & 2
Location:	6351 South Ocean Drive Jensen Beach, FL 34957
Dates:	March 28 - June 26, 2004
Inspectors:	T. Ross, Senior Resident Inspector S. Sanchez, Resident Inspector B. Crowley, Reactor Inspector (Section 4OA5.1) S. Ninh, Senior Project Engineer (Sections 1R12 and 1EP)) R. Chou, Reactor Inspector (Section 1R08) A. Vargas-Mendez, Reactor Inspector (Section 1R08) S. Rudisail, Project Inspector (Sections 1R1, 1R23, and 4OA1)
Approved by:	Joel Munday, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure 1

SUMMARY OF FINDINGS

IR 05000335/2004-04, 05000389/2004-04; 03/28/2004 - 06/26/2004; Florida Power & Light; St. Lucie Nuclear Plant, Units 1 & 2; Licensed Operator Requalification, Maintenance Effectiveness, and Personnel Performance During Nonroutine Plant Evolutions and Events.

The report covered a three month period of inspection by resident inspectors and several inspectors from Region II. Three 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. Inspector Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified two examples of a non-cited violation (NCV) of Technical Specification 6.8.1.(a) for failure to follow procedures in maintaining and reactivating SRO licenses. This resulted in two senior reactor licensed operators standing watch without the appropriate qualifications.

The finding, which involves the mitigating systems cornerstone, is greater than minor because it is associated with human performance attributes that affect the availability, reliability, and capability of licensed operators to respond to initiating events to prevent undesirable consequences. The NRC considers the maintenance and proficiency of licensed operators an element of the defense in depth philosophy, and the compliance with procedures which implement the requirements of 10 CFR 55.53(f), to be significant. (Section 1R11)

Green. A self-revealing NCV of Criterion XVI of 10 CFR 50, Appendix B, Corrective Action was identified for the licensee’s failure to implement adequate corrective actions to address water intrusion events which resulted in repetitive torque switch failures in the close control circuit of the Unit 1 MV-21-3, the “A” train ICW to non-safety related TCW piping isolation valve.

The finding was greater than minor because it involved the equipment performance attribute of the mitigating system cornerstone and affected the objective of ensuring that equipment is available and capable to respond to an event. The finding was determined to be of very low safety significance in accordance with the Significance Determination Process (SDP) phase 1, since another independent intake cooling water (ICW) train remained operable and available to perform the safety function. In addition, the valve was repaired and returned to service within the required 72 hour Technical Specification limit when the condition was identified. (Section 1R12)

Cornerstone: Barriers

Green. A self-revealing NCV of Technical Specifications 6.8.1(a) was identified for failing to maintain configuration control of the Unit 1 shutdown cooling (SDC) purification system in accordance with normal operating procedure 1-NOP-03.05, Shutdown Cooling.

The finding was considered greater than minor because it involved the reactor coolant system (RCS) barrier and if left uncorrected could have resulted in more significant safety consequences such as a continued loss of reactor coolant inventory eventually resulting in loss of radiological shielding and core cooling. The finding was determined to be of very low safety significance according to Appendix G for Shutdown Operations of the Significance Determination Process since there was not a significant loss of RCS inventory control. (Section 1R14)

B. Licensee Identified Violations

None.

Report Details

Summary of Plant Status

Unit 1 began the report period already shutdown for its 19th refueling outage (SL1-19). The unit was restarted on April 25, 2004 and returned to full power operation. Unit 1 remained at full power for the duration of the report period except for one unplanned power reduction. On June 3, operators reduced reactor power to about 90% due to a trip of the 1B Heater Drain Pump. Repairs were made to the 1B Heater Drain Pump discharge level control valve and Unit 1 was returned to full power operation.

Unit 2 operated at essentially full power the entire inspection report period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity [Reactor-R]

1R01 Adverse Weather Protection

Site Specific Weather Related Condition: Hurricanes

a. Inspection Scope

During the week of June 21, the inspectors verified the status of licensee actions in accordance with administrative procedure (ADM)-04.01, Hurricane Season Preparation. This verification included physical walkdowns of the licensee's property and discussions with responsible licensee personnel regarding systems, structures, and components (SSCs) vulnerable to high winds and potential flooding during a hurricane. The inspectors reviewed applicable Technical Specifications (TS), a memo issued by the site Vice President regarding "Hurricane Season," and outstanding Plant Manager Action Items (PMAIs) used to track incomplete items from ADM-04.01. The inspectors also reviewed administrative procedure AP-0005753, Severe Weather Preparations, and specifically examined the following exterior areas:

- Unit 1 and Unit 2 Reactor Auxiliary Buildings (RAB)
- Unit 1 and Unit 2 Turbine Buildings

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial Equipment Walkdowns

The inspectors conducted three partial 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. These inspections included reviews of applicable TS, plant lineup procedures, operating procedures, and/or piping and instrumentation drawings (PI&D) which were compared with observed equipment configurations to identify any discrepancies that could affect operability of the redundant train or backup system. The inspectors also reviewed applicable reactor control operator (RCO) logs; out of service (OOS) and operator work around (OWA) lists; active temporary system alterations (TSA); and outstanding condition reports (CR) regarding system alignment and operability.

- Unit 1 Spent Fuel Pool Cooling After Being Returned to Service per OP 1-0350020, Fuel Pool Cooling and Purification System - Normal Operations
- Unit 2 Intake Cooling Water System per P&ID 2998-G-082
- 1B Auxiliary Feedwater Pump Train per P&ID 8770-G-080

b. Findings

No findings of significance were identified.

.2 Complete Equipment Walkdown

a. Inspection Scope

During the week of April 18, an inspector completed a detailed alignment verification of the Unit 1 Safety Injection Tanks (SIT). The inspector used OP 1-0410021, SIT Normal Operation, Appendix A, SIT Initial Alignment, and the applicable P&ID, to conduct a thorough walkdown of the Unit 1 SITs inside and outside of containment. This walkdown included verification of critical SIT parameters (i.e., pressure, level); and local and control board key lock positions of SIT discharge valves. The inspector also reviewed applicable sections of the Updated Final Safety Analysis Report (UFSAR) and TS. Furthermore, the inspector reviewed applicable RCO logs; OOS and OWA lists; active TSAs and outstanding CRs regarding system alignment and operability. In addition, the inspector examined boron concentration sample results of each SIT to verify TS compliance.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Annual Inspections

a. Inspection Scope

On April 13 and May 28, the inspectors observed fire brigade response for a reported fire of the Unit 2 Gantry Crane and an unannounced drill, respectively. The inspectors verified that fire brigade protective clothing and gear were properly worn and used; and that fire hose lines and/or portable suppression equipment were capable of reaching all necessary fire hazard locations. The inspectors monitored communications between the fire brigade leader, other fire fighters and plant operators. The inspectors also attended a post-drill critique and discussed fire brigade performance with the Fire Protection Training Coordinator.

b. Findings

No findings of significance were identified.

.2 Routine Inspections

a. Inspection Scope

The inspectors conducted tours of the following 12 fire areas or witnessed associated activities listed below during the inspection period to verify they conformed with Administrative Procedure AP-1800022, 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 1 Pipe Tunnel Area (Fire Area J)
- Unit 1 Reactor Containment Building (Fire Area K)
- Unit 1 Containment Hot Work
- Unit 2 Steam Trestle and AFW Pumps (Fire Area FF)
- Unit 1 Condensate Storage Tank Enclosure (Fire Area X-X)
- Unit 2 Condensate Storage Tank Enclosure (Fire Area J-J)
- Unit 1 Steam Trestle and AFW Pumps (Fire Area S-S)
- Unit 1 Vital Switchgear Rooms (Fire Areas A and C)
- Unit 2 Vital Switchgear Rooms (Fire Areas A and C)
- Unit 1 Component Cooling Water Platform (Fire Area U-U)
- 2B Emergency Diesel Generator (EDG) Room (Fire Area II)
- 1A EDG Room (Fire Area H-H)

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI)

a. Inspection Scope

The inspectors observed inspection activities and reviewed the documentation and selected supporting records for ISI work activities conducted during St. Lucie Unit 1 refueling outage 19. The inspection activities, documentation, and supporting records were reviewed for compliance to the TS; the ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, with no Addenda; and other appropriate industry and NRC guidance and standards.

The inspectors reviewed the following nondestructive examinations (NDE):

Ultrasonic Test (UT): Weld Nos: RC-112-FW-2-500H & RC-115-FW-2-500F for Intermediate Pipe Welds from Steam Generator 1A to reactor coolant Pump 1A1 and 1A2

Visual Inspection (VT): Pressurizer Heater Penetrations

Qualification and certification records for NDE procedures, examiners, and equipment and consumables (i.e., UT oscilloscopes, transducers, calibration blocks, and couplant) for the inspected ISI examinations were reviewed.

The inspectors reviewed the "Owner's Activity Report" dated September 29, 2003 for the last outage Unit 1, SL1-18 which contained no recordable indications that required evaluation for continued service.

The inspectors reviewed Maintenance Work Order (MWO) 31024074 for the replacement of the Class 2 vent valves CH-210 and CH-208 for Charging Pump 1C and 1A with butt welds in the reactor coolant pressure boundary. The inspectors reviewed radiographs for the welds to ensure they met ASME code requirements.

The inspectors reviewed ASME Section XI Code repairs or replacements including Class 2 vent valves CH-210 and CH-208 (MWO 31024074) for Charging Pump 1C and 1A and Class 3 Safety Relief Valve V2345 (MWO 30011450) for letdown control valve station to ensure they met ASME code requirements.

The inspectors reviewed the steam generator degradation assessment, qualified Eddy Current examination technique, engineering evaluation and justification for deviation from the EPRI Pressurized Water Reactor (PWR) Steam Generator examination guidelines, and Eddy Current equipment qualification for the steam generator examination.

The inspectors reviewed the licensee's Boric Acid corrosion control program inspection results. The inspectors walked down the containment to assess selected licensee findings and to independently check for any additional leakages. In addition the inspectors reviewed Quality Assurance audits, Self Assessments and evaluations for thoroughness and appropriate corrective action.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Routine Quarterly Inspection

a. Inspection Scope

On May 13, 2004, the inspector observed and assessed licensed operator actions of two separate crews during simulator evaluations. During these simulator evaluations, the operating crews responded to simulated accident events (i.e., excess steam demand outside of containment, and control element assembly ejection), coincident with the failure of critical equipment, in accordance with applicable emergency operating procedures (EOP). The inspector specifically evaluated the following attributes related to the operating crews' performance:

- Clarity and formality of communication
- Ability to take timely action to safely control the unit
- Prioritization, interpretation, and verification of alarms
- Correct use and implementation of EOP-01, Standard Post Trip Actions; EOP-3, LOCA; and EOP-5, Excess Steam Demand
- Timely and appropriate Emergency Action Level declarations per Emergency Plan Implementing Procedure (EPIP) - 01, Classification of Emergencies
- 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.

.2 Operator License Maintenance and Reactivation

a. Inspection Scope

During the course of routine control room tours, the inspector frequently interviewed operators, and reviewed applicable documentation (e.g., RCO logs and License

Watchstanding Records), regarding the status of operator license qualification. The requirements for "Maintenance of Active License status," are prescribed by AP-0005720, Licensed Operator Continuing Training Program.

b. Findings

Introduction. A Green NCV, with two examples, was identified for failure to follow the requirements of procedure AP-0005720, "Licensed Operator Continuing Training Program," as required by Technical Specification 6.8.1.(a).

Description. The inspectors identified two instances where the licensee did not adhere to the requirements of procedure AP-0005720, "Licensed Operator Continuing Training Program." One Senior Reactor Operator (SRO) failed to properly reactivate his license, in that, he failed to complete all elements of a full plant tour required by AP-0005720. Another SRO failed to maintain watch standing proficiency by standing proficiency watches in a position other than a Technical Specification defined minimum staff position as required by AP-0005720.

The inspectors, along with OI (OI Report No. 2-2004-016), found that one SRO stood three SRO and three Reactor Operator (RO) proficiency watches of 12 hours each during the last quarter of 2003. While this would satisfy the proficiency requirement of 10 CFR 55.53(e) for the RO position, the inspector discovered that for the three RO watches, the individual stood watch as an extra-RO in the control room, not as the board or desk RO (Technical Specification defined minimum staff position). Procedure AP-0005720, Section 7.6.2.A., stated that to maintain an active license the operator must actively perform the functions of an RO or SRO "in a Technical Specification defined minimum staff position." The SRO in question failed to stand watch as a board or desk RO during his three 12-hour shifts as an RO, therefore his license was inactive for the first quarter of 2004 for the RO position, along with already being inactive for the SRO position. The inspector identified that this SRO stood watch on January 16, 2004 as a Unit 1 reactor operator (RO, a position that requires an active license) with an inactive license.

The licensee took prompt corrective action when informed by the inspector and entered this issue into their corrective action program (Condition Report # 04-0838).

The licensee was informed of another non-compliance with procedure AP-0005720 in a letter dated April 16, 2004 (EA-04-077) which contained synopses of two NRC Office of Investigations (OI) completed reports. During an inspection conducted in September 2003, with a subsequent follow-up OI investigation (OI Report No. 2-2003-073) the NRC substantiated that an Assistant Nuclear Plant Supervisor (ANPS) failed to follow procedure AP-005720 by not completing a plant tour during license reactivation. The NRC identified that the licensee improperly certified on March 18, 2003 that the requirements for reactivation of an SRO license had been met, in that, the SRO had not completed a plant tour of areas required by AP-0005720 and had so indicated by placing an N/A in the space provided on the reactivation form. The licensee reactivated the SRO's license without completion of all requirements.

The licensee took prompt corrective action and entered this issue into their corrective action program (Condition Report # 03-3418). During the evaluation of the extent of condition the licensee did not identify any additional discrepancies.

Analysis. The inspector determined that the licensee's failure to properly certify reactivation records and to properly ensure proficiency of operators is a performance deficiency because the licensee is required to certify that the qualifications and status of licensed operators are current and valid and that they meet the requirements of AP-0005720, "Licensed Operator Continuing Training Program."

The finding, which involves the mitigating systems cornerstone, is greater than minor because it is associated with human performance attributes that affect the availability, reliability, and capability of licensed operators to respond to initiating events to prevent undesirable consequences. The NRC considers the maintenance and proficiency of licensed operators an element of the defense in depth philosophy, and the compliance with procedures which implement the requirements of 10 CFR 55.53(f), to be significant.

The finding is not suitable for SDP evaluation, but has been reviewed by NRC management and is determined to be a Green finding of very low safety significance. In these two cases, the individual's failure to complete the plant tour or to properly maintain a license had little effect on the overall knowledge level of each individual's knowledge of plant operations or plant status, and additional watch standers were on shift that were cognizant of the current plant conditions when the individual's were placed on watch.

Enforcement. TS 6.8.1.(a) requires written procedures be established, implemented and maintained covering the applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978, and those required for implementing the requirements of NUREG - 0737. Procedure 0005720 is a written procedure that is required by TS 6.8.1.(a). Contrary to the above, the inspectors identified two examples where the licensee did not adhere to the requirements of this procedure. (1) The NRC identified that on March 18, 2003, the licensee improperly certified that the requirements for reactivation of an SRO license had been met, in that, the SRO had not completed a plant tour of areas required by AP-0005720 and had so indicated by placing an N/A in the space provided on the reactivation form. The licensee reactivated the SRO without completion of all requirements. (2) The NRC identified that during the fourth quarter of 2003, an SRO had failed to maintain his license proficiency, in that he failed to stand proficiency watches in a technical specification defined minimum staff position, as required by AP-0005720. Because the failure to reactivate an inactive license and maintain proficiency in accordance with procedure AP-0005720 is of very low safety significance and has been entered in the licensee corrective action program (Condition Report # 03-3418 and #04-0838), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000335, 389/2004004-01, Failure to Follow Procedures per TS 6.8.1.(a).

1R12 Maintenance Rule Implementation

.1 Routine Inspection

a. Inspection Scope

The inspectors reviewed the reliability and problems associated with the 4 systems, structures, and components (SSCs) listed below, including associated condition reports. 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' efforts focused on the licensee's work practices and ability to identify and address common causes, maintenance rule scoping, characterization of reliability issues and assigning unavailability time, determination of a(1) and a(2) classification, corrective actions, and the appropriateness of established performance goals and monitoring criteria. The inspectors also attended applicable expert panel meetings, interviewed responsible engineers, and observed some of the corrective maintenance activities. Furthermore, the inspectors verified that equipment problems were being identified at the appropriate level and entered into the corrective action program

- CRs 01-2077, 03-2973, and CR 03-3319; Unit 1 and 2 Repetitive Functional Failures of valve MV-21-3 to close on demand
- CRs 01-3202, 03-4018, and 04-2005, Unit 1 and 2 ICW Systems Exceeded Unavailability Criteria
- CRs 03-0115 and 03-497; Unit 1 and 2 Engineering Safety Feature Actuation System Actuations
- CRs 02-3173, 03-0098, 034178, and 034548; Unit 1 and 2 Reactor Protection Systems (RPS) Multiple Equipment Performance Problems

b. Findings

Introduction. A Green self-revealing non-cited violation (NCV) of 10 CFR 50 Appendix B, Criterion XVI, was identified for the licensee's failure to implement adequate corrective actions to address water intrusion events which resulted in repetitive torque switch failures in the close control circuit of Unit 1 valve MV-21-3.

Description On August 15 and September 17, 2003, Unit 1 valve MV-21-3 failed to close on the demand from the control room hand switch while Operations was initiating the line up for the intake cooling water (ICW)/turbine cooling water (TCW) heat exchanger back wash. The licensee's investigation determined that the apparent cause for the failure was water intrusion through the limit switch housing cover flange that affected torque switch performance and continuity. A similar water intrusion torque switch failure of Unit 2 valve MV-21-3 occurred on August 6, 2001. The inspectors determined that the valve was successfully repaired and returned to service within the required 72 hour Technical Specification action statement each time. The licensee's investigation concluded that prior corrective actions were not successful to address the water intrusion events. The licensee concluded that the second failure occurred due to weak procedural guidance, a poor practice for installation of protective covers, and material condition of the limit switch compartment cover. Corrective actions included maintenance procedure enhancements, cleaning of the limit switch compartment, sealing of the water entry locations and installation of additional barriers. In addition, the licensee installed protective covers on MOVs that have safety functions and are located in an outdoor environment.

Analysis MV-21-3 is the "A" train ICW supply to non-safety related TCW piping isolation valve. It is required to perform a safety related function to automatically close on receipt of a safety injection actuation signal (SIAS). The inspectors determined that the licensee's failure to implement adequate corrective actions to address water intrusion events was a performance deficiency. The finding was greater than minor because it involved the equipment performance attribute of the mitigating system cornerstone and affected the objective of ensuring that equipment is available and capable to respond to an event. The finding was determined to be of very low safety significance in accordance with the Significance Determination Process (SDP) phase 1, since another independent intake cooling water (ICW) train remained operable and available to perform the safety function. In addition, the valve was repaired and returned to service within the required 72 hour Technical Specification limit when the condition was identified.

Enforcement. Criterion XVI of Appendix B to 10 CFR 50, Corrective Action, states in part that, "Measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition." Contrary to this, on September 17, 2003, Unit 1 valve MV-21-3 experienced a repetitive functional failure due to the licensee not having adequate corrective actions to address water intrusion events which resulted in torque switch failures that occurred on August 6, 2001 and August 15, 2003. However, because this violation is of very low safety significance and was addressed by the licensee's corrective action program as CRs 03-2973 and 03-3319, this violation is being treated as a non-cited (NCV) consistent with Section VI.A of the NRC Enforcement Policy - NCV 05000335/2004004-02, Inadequate Corrective Actions to Preclude Repetitive Torque Switch Failure in Close Control Circuit of Unit 1 Valve MV-21-3.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the risk assessments for the following six SSCs that were OOS for 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 procedures such as ADM-17.16, Implementation of the Configuration Risk Management Program. Furthermore, 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 OOS risk significant SSCs listed below:

- U1 Entering Mode 3 with multiple risk significant systems OOS
- 2B EDG Critical Maintenance Management (CMM) Work Week, including multiple OOS risk significant systems
- 2A EDG CMM Work Week, including multiple OOS risk significant systems
- 2C ICW, 2A CCW Heat Exchanger (HX), 2BB Battery Charger, 2D Instrument Air Compressor (IAC), and Boric Acid Makeup Tank (BAMT) Gravity Drain Valve
- 1A ECCS CMM, including multiple OOS risk significant systems
- 1A AFW Pump CMM, including multiple OOS risk significant systems

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events

a. Inspection Scope

For the non-routine evolutions described below, the inspectors evaluated operator performance through interviews, observations, examining available information (e.g.,operator logs, plant computer data, and strip charts), and reviewing applicable CRs to determine what occurred, how the operators responded, and to verify that the response was in accordance with plant procedures(e.g., normal and abnormal operating procedures, EOPs, etc.).

- On March 31, 2004, the licensee declared a notification of Unusual Event (UE) on Unit 1 due to RCS leakage in excess of 10 gpm. This event occurred during SL1-19, while Unit 1 was being defueled and reactor cavity level was greater than 23 feet above the fuel (see Section 4OA3 of this report). The cause of this event was attributed to human performance errors (see Section 4OA4 of this report).

- On June 1, 2004, the inspector attended an “infrequently performed evolution” briefing by management, and a Health Physics briefing, of personnel involved in the physical inventory of failed fuel rods in the Unit 1 and 2 Fuel Rod Storage Baskets (FRSB). During the first two weeks of June, the inspector observed preparations for, and witnessed execution of, the Unit 1 spent fuel pool FRSB inspection in accordance with MRS-SSP-1641, Inventory of the Fuel Rod Storage Baskets at St. Lucie Units 1 and 2, and WO #34010089. The inspector focused particular attention to monitoring personnel actions, communications, and teamwork during this infrequent evolution due to the multiple fuel rod manipulations required by the procedure.

b. Findings

Introduction. A Green self-revealing non-cited violation (NCV) was identified for failing to maintain configuration control of the Unit 1 shutdown cooling (SDC) purification system in accordance with the equipment clearance order (ECO) special instructions that would have required removing SDC purification from service per 1-NOP-03.05, Shutdown Cooling.

Description. On March 30, 2004, the onshift operating crew executed ECO #CV090 in order to allow for work on SDC Purification valve 2345. The special instructions of ECO #CV090 stated "Remove SDC Purif. from service prior to hanging this ECO." Although the removal of the SDC purification system in accordance with the normal operating procedure 1-NOP-03.5 was discussed by the responsible senior reactor operator during the initial ECO briefing, ECO execution was interrupted for an extended period due to delays in field work preparations. Subsequent poor communications, and inadequate shift turnover, regarding system conditions and the status of ECO #CV090, resulted in a failure to properly implement Step 6.8, Remove SDC Purification From Service, during the following shift when the ECO tags were hung. As a consequence of failing to properly secure the SDC Purification system per 1-NOP-03.05, the SDC purification outlet valve (V2013) was left open. Then on March 31, while Unit 1 was being defueled with the reactor cavity level greater than 23 feet above the fuel, Operations authorized maintenance personnel to conduct testing on the Volume Control Tank level divert valve (V2500). This testing required stroking V2500 several times, and was scheduled based on the SDC Purification system being removed from service. Subsequent stroking of V2500, with V2013 left open, resulted in an inadvertent flow path from the RCS and refueling cavity inventory to the 1D Holdup tank. Later on that same day, operators noticed an unexpected decreasing level in the refueling cavity with a commensurate increasing level in the 1D Holdup tank. Operations promptly investigated and located and secured the leak path. During the event approximately two inches of level was observed lost from the refueling cavity which resulted in approximately 6000 gallons of reactor coolant being drained to the 1D Holdup tank.

Analysis. The principal cause of this event was the failure of Operations supervision to maintain configuration control of the Unit 1 SDC system in accordance with normal operating procedures and ECO instructions. Failure to maintain configuration control of SDC purification system is a finding considered greater than minor because if left uncorrected could have resulted in more significant safety consequences such as a continued loss of reactor coolant inventory eventually resulting in loss of radiological shielding and core cooling. The finding was determined to be of very low safety significance according to Appendix G for Shutdown Operations of the Significance Determination Process since there was not a significant loss of RCS inventory control.

Enforcement. TS 6.8.1.a requires that written procedures shall be established, implemented, and maintained as recommended in Appendix A of Regulatory Guide (RG) 1.33, Revision 2, February 1978, of which Section 3.c specifically identifies the Shutdown Cooling System. Contrary to TS 6.8.1.a, on March 30, 2004, Unit 1 operators failed to properly remove the Unit 1 SDC purification system from service in accordance with 1-NOP-03.05. However, because this violation is of very low safety significance and was addressed by the licensee's corrective action program (i.e., CR 04-1749), this violation is being treated as a non-cited (NCV) consistent with Section VI.A of the NRC Enforcement Policy - NCV 05000335/2004004-03, Improper Configuration Control Of Shutdown Cooling Purification System Led To Inadvertent Reactor Coolant System Leak That Resulted In An Unusual Event.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following seven 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 unrecognized increase in risk. As applicable, the inspectors reviewed the UFSAR, and associated supporting documents and procedures, and interviewed plant personnel to assess the adequacy of the interim CR disposition.

- CR 04-1973, Unit 1 SDC Piping Restraints
- CR 04-1046, Mis-wired 480V Breakers in Safety-related Applications
- CR 04-2381, 2A EDG Startup Air Tank 2A4 Relief Valve
- CR 04-2779, 2A EDG Fuel System Fault Annunciator
- CR 04-2849, Battery Charger 2A/B Output Breakers
- CR 04-0762 and Supplement 1, Agastat type DSC Time Delay Relays
- CR 04-1956, Unit 1 Main Control Board (MCB) Fuse Blocks

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors routinely reviewed the Operator Work Around (OWA) log for both units and discussed new items with Operations supervision. The inspectors also routinely walked down unit MCBs, reviewed operator chronological logs and equipment OOS logs, and examined MCB plant work order (PWO) tags for potential OWAs and minor operator burdens.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors witnessed and reviewed post maintenance test (PMT) activities of the 6 risk significant SSCs listed below. The following aspects were specifically 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 the PMTs to ensure that they were correctly identified and appropriately entered into the corrective action program.

- 1C AFW Pump per OP 1-0700050, AFW Periodic Test
- 1C AFW Pump Governor per 0-IMP-09.01, AFW Turbine Governor Oil Change and Tuning Instruction
- 2HVS-1C Relay per PWO #33007023-01
- U2-HVE-9A Fan DSC Relay per PWO # 33007228-01
- 2A EDG Engine Inspection per PWO #33015313-01
- 2B EDG Operability Test per OP 2-2200050B, 2B EDG Periodic Test

b. Findings

No findings of significance were identified.

1R20 Refueling and 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 SL1-19, in particular the Risk Assessment Team (RAT) notebook. The inspectors also examined the licensee’s implementation of shutdown safety assessments during SL1-19 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 March 21, 2004. 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

Through SL1-19, the inspectors examined specific, critical 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 reactor coolant system (RCS) pressure, level, flow, and temperature instruments during various modes of operation
- Verified electrical systems availability and alignment
- Monitored important control room plant parameters
- Verified shutdown cooling system and spent fuel pool cooling system operation
- Evaluated implementation of reactivity controls
- Reviewed control of containment penetrations
- Examined foreign material exclusion (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

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 spent fuel pool. 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 their procedural controls (e.g., 1-M-0060, Containment Closure).

Heatup, Mode Transition, and Reactor Startup Activities

The inspectors examined selected TS, license conditions, and other commitments and verified administrative prerequisites were being met prior to mode changes. The inspectors also specifically reviewed the initial RCS inventory balances used to measure RCS leakage, and verified containment integrity was properly established. The inspectors performed a detailed containment sump closeout inspection on April 16, during Mode 5. The inspectors also conducted a thorough containment walkdown on April 21 after the Unit 1 reactor plant had reached Mode 3 and was at normal operating pressure and temperature. Lastly, the inspectors witnessed portions of the reactor coolant system heatup, reactor startup and power ascension in accordance with plant procedures, such as: Preoperational Test Procedure (POP) 1-3200088, Unit 1 Initial Criticality Following Refueling; POP 0-3200092, Reactor Engineering Power Ascension Program; and 1-GOP-201, Reactor Plant Startup. 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.

Correction Action Program

The inspectors reviewed CRs generated during SL1-19 to evaluate the licensee's threshold for initiating CRs. The inspectors also selected, numerous CRs to verify appropriate priorities, mode holds, and significance levels were being assigned. Resolution and implementation of corrective actions of several CRs were also examined. Furthermore, the inspectors routinely reviewed the results of Quality Assurance daily surveillances of outage activities.

b. Findings

No findings of significance were identified.

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 (OP), and operations surveillance procedures (OSP). The actual 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 corrective action program for resolution. The tests reviewed included 1 IST and 1 containment isolation valve surveillance.

- OP 1-0400050, Periodic Test of the Engineered Safety Features
- 1-IMP-01.05D, Reactor Coolant Temperature Calibration Channel D
- 0-EMP-50.01, 125VDC System Battery Charger 18 Month Operability Test, of the 2BB Battery Charger

- OP 2-0700050, 2C AFW Pump Inservice Test Code Run
- 1-OSP-68.02, Local Leak Rate Testing, of Penetration 52B
- 1-OSP-03.02A, 1A Low Pressure Safety Injection Flow Test

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors continued to periodically screen active temporary modifications, especially for risk significant systems. The inspectors specifically examined the technical evaluation and associated 10CFR50.59 screening of TSA #2-04-004 regarding the 2A EDG electric fuel oil priming pump control circuit. This TSA was reviewed 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 warrant prior NRC approval. The inspector also walked down the installation of the TSA to verify configuration control was maintained. Furthermore, the inspectors verified and reviewed required condition monitoring by Operations, and discussed compensatory actions detailed by the TSA with Operations supervision.

b. Findings

No findings of significance were identified.

1REP Equipment Availability, Reliability and Functional Capability - Pilot

.1 Maintenance Rule Periodic Evaluation

a. Inspection Scope

The inspectors reviewed the licensee’s latest Maintenance Rule periodic assessment, “St. Lucie Plant Maintenance Rule Periodic Assessment,” issued in July, 2003. The report was issued in accordance with paragraph (a)(3) of 10 CFR 50.65, and covered the period September 2001 through March 2003 for both units. The (a)(3) report was reviewed to determine that the assessment satisfied the time requirement of the Maintenance Rule and included evaluation of: balancing reliability and unavailability, (a)(1) activities, (a)(2) activities, and use of industry operating experience. To verify compliance with 10 CFR 50.65 and the effectiveness of the licensee’s (a)(3) activities, the inspectors reviewed selected work history and maintenance rule activities for the following risk significant systems for both units: AFW, ESFAS, RPS, and ICW. The documents, procedures and condition reports reviewed during the inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors assessed the accuracy of the Unit 1 and 2 HPSI Unavailability Performance Indicator (PI) reported to the NRC in accordance with the criteria specified in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, and ADM-25.02, NRC Performance Indicators. The inspectors reviewed the PI data of both Units 1 and 2 for the previous four quarters. Applicable operator logs, condition reports, Maintenance Rule history, and Licensee Events Reports were reviewed to verify the reported PI data was complete and accurate. Furthermore, the inspectors interviewed the responsible engineering and licensing personnel.

b. Findings

No findings of significance were identified.

.2 Barrier Integrity Cornerstone

a. Inspection Scope

The inspectors assessed the accuracy of the following PIs reported to the NRC in accordance with the criteria specified in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, and ADM-25.02, NRC Performance Indicators:

- Unit 1 and 2 Reactor Coolant System Leakage
- Unit 1 and 2 Reactor Coolant System Activity

The inspectors reviewed the PI data of both Units 1 and 2 for the previous four quarters. The inspectors also interviewed the responsible system engineer and Chemistry technician regarding the acquisition, review, validation, and reporting of PI data to the PI program coordinator. Furthermore, the inspectors reviewed a large number of selected source documents (i.e., RCS inventory balance data sheets and primary chemistry sample analysis results) during the past twelve months to verify accuracy and completeness of PI data.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Routine Review Of Condition Reports

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems", and to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of all condition reports as they were entered into the licensee's corrective action program.

b. Findings And Observations

There were no specific findings identified from this overall review of the CRs issued each day.

.2 Semi Annual Trend Review

a. Inspection Scope (71152)

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspector's review included daily screening of individual condition reports (see section 4OA2.1 above), licensee trending efforts, and licensee human performance self-assessments. The inspectors review nominally considered the six month period of January 2004 through June 2004, although some examples expanded beyond those dates when the scope of the trend warranted. This review also specifically examined equipment issues identified in selected System Health Reports; human performance issues identified in the 1st quarter Departmental Human Performance Corrective Action Program (CAP) Rollup Reports; and adverse and negative trends identified by the Cross Functional Trend Coordinator Team Reports for the months of January, February and May 2004. Furthermore, the inspectors verified whether adverse or negative trends and issues identified in the licensees reports were entered into the licensee's CAP.

b. Findings and Observations

There were no findings of significance identified. However, based on the CAP Rollup results the inspectors identified potential human performance related trends associated with self-checking and procedural compliance. This was based on multiple CR's identified in the Operations, Maintenance, and Health Physics departments, which had cause codes associated with these two areas. Although, the licensee had assigned the cause codes, they had not identified the trends. In response to the inspectors' observations, the licensee conducted a more detailed evaluation of 1ST Quarter CR cause codes per department, from which they concluded that two negative trends involving "Inadequate Self-Checking" and "Work Documents Not Followed" existed in the Maintenance Department. These negative trends were then entered into the CAP as CR 2004-3471.

.3 Annual Sample Review

.1 Damaged Unit 1 Main Control Board Fuse Blocks

a. Inspection Scope

The inspector evaluated the licensee's actions to address damaged (i.e., melted due to arcing) fuse block holders that were discovered inside the Unit 1 Reactor/Turbine Generator Board (RTGB) 106. Condition Report 04-1956, and its Supplement, were reviewed to verify that the problem identification was timely, complete and accurate; safety concerns were properly classified and prioritized for resolution; technical issues were evaluated and dispositioned to address operability; the cause determination was sufficiently thorough; the extent of condition was adequately examined; and appropriate corrective actions (short and long term) were implemented or planned in a manner consistent with safety and TS compliance. The inspector evaluated the CR against the requirements of the licensee's corrective action program as delineated in Nuclear Administrative Procedure NAP-400, Condition Reports; and 10 CFR 50, Appendix B. The inspector also physically inspected the condition of the damaged RTGB 106 fuse blocks, and physically inspected the entire population of fuse blocks in RTGB 101 through 105. Furthermore, the inspector interviewed responsible Operations and Engineering personnel; and, reviewed Operations Night Orders and operator training implemented as part of the interim corrective actions.

b. Findings and Observations

No safety significant findings were identified. However, the inspector noted that the initial disposition of CR 04-1956 did not include an operability evaluation, or extent of condition review. The CR was originally classified as a Significance Level 3, with an investigation type of Correction Only, and flagged as a Mode 4 hold. Per NAP-400, this level and type of CR classification did not require an extent of condition review. But after further independent examination in the field, the inspector identified numerous other broken Bakelite insulating barriers between RTGB fuse blocks. Based on the inspector's findings, the licensee conducted their own comprehensive inspection of the Unit 1 RTGB fuse blocks and discovered 17 additional broken barriers affecting 34 fuse blocks. A supplement to CR 04-1956 was then initiated that adequately addressed the expanded extent of condition, generic implications, potential operability concerns for Mode 4, and established broader scope corrective actions. Failure to properly classify the level and type of CR necessary to adequately assess the extent of condition of the degraded fuse blocks was considered a weakness in the guidance of NAP-400 and its implementation. The inspector also reviewed the licensee's interim corrective actions which involved a revised technique for removing and installing fuses as described in an Operations Night Orders. However, the Unit 1 crew interviewed by the inspector was unaware of the revised fuse pulling/inserting technique and the Operations Night Order was not well disseminated. So, in order to better formalize the new fuse pulling/inserting method, the licensee subsequently implemented a procedure change to OP-1250020, Valve, Breaker, Motor And Instrument Instructions.

.2 Multiple Agastat DSC Time Delay Relay Failures

a. Inspection Scope

The inspector evaluated the licensee's actions to address early end-of- life failures of Agastat DSC time delay relays used in numerous safety-related systems. Condition Report 04-0762, and its Supplement, were reviewed to verify that the problem identification was timely, complete and accurate; safety concerns were properly classified and prioritized for resolution; technical aspects were evaluated and dispositioned to address operability; the cause determination was sufficiently thorough; the extent of condition was adequately examined; generic implications, common causes, and previous history were fully considered; and appropriate corrective actions (short and long term) were implemented or planned in a manner consistent with safety, risk significance, and TS compliance. The inspector evaluated the CR against the requirements of the licensee's corrective action program as delineated in Nuclear Administrative Procedure NAP-400, Condition Reports; and 10 CFR 50, Appendix B. The inspector also met several times with engineering personnel, supervision and management to discuss the aforementioned attributes as related to CR 04-0762, including Supplement 1. Furthermore, the inspectors observed several Agastat relay replacements in the field.

b. Findings and Observations

No safety significant findings were identified. However, the licensee's initial operability evaluation did not adequately explain or bound the impact or scope of past and future, premature end-of-life failures upon risk significant, safety-related systems. The CR was originally classified as a Significance Level 3, with an investigation type of Correction Only, which did not require an extent of condition review or corrective actions to preclude recurrence (CAPR). Although not specifically required by the CR type, the licensee did evaluate the extent of condition. But the licensee's corrective actions to replace the DSC relays based solely on age did not address system risk significance, service application of the relay, or relay failure mode in order to institute effective CAPRs. In addition, the licensee had not established a comprehensive and detailed plan to affect replacement of all safety-related relays in a timely and prioritized manner to preclude future failures nor did the licensee consider an interim testing schedule to identify relays early on that could fail in an undetectable manner prior to replacement. All these aspects were necessary elements of effective CAPRs. Failure to properly classify the level and type of CR necessary to require effective CAPRs prior to a repeat DSC relay failure was considered a weakness in the guidance of NAP-400 and its implementation. A subsequent supplement to CR 04-0762 fully addressed the inspector issues associated with the operability evaluation and CAPRs.

.4 Unit 1 MV-21-3 Repeat Functional Failures (Cross-cutting issues)

Section 1R12 describes the licensee's failure to implement adequate corrective actions to address water intrusion events which resulted in a repetitive torque switch failure in the close circuit of Unit 1 valve MV-21-3.

4OA3 Event Follow-up

Declaration of Unusual Event Due to RCS Leakage in Excess of 10 Gallons per Minute

a. Inspection Scope

On March 31, 2004, the licensee identified an inadvertent leak path from the Unit 1 RCS to the 1D Holdup Tank during flowscan testing of the volume control tank divert valve (V2500). During this event, the unit was in Mode 6 with the reactor vessel head removed and the reactor cavity flooded up to 60 feet. The licensee declared an Unusual Event due to RCS leakage exceeding 10 gallons per minute (gpm) in accordance with Emergency Plan Implementing Procedure (EPIP) 01, Classification of Emergencies. An inspector reviewed operator logs, assessed plant conditions, and evaluated operator actions according to applicable off normal operating procedures. The inspector also interviewed responsible Operations and Engineering personnel, plant management and attended an Event Review Team meeting. Furthermore, the inspector reviewed the 10 CFR 50.72 notification regarding the entry and termination of the Unusual Event, and discussed the event with regional management.

b. Findings

No findings of significance were identified, except for the human performance deficiency described in Section 1R14 of this report.

4OA4 Cross Cutting Issues

Section 1R14 describes a human performance error where operators failed to properly remove the Unit 1 SDC purification system from service in accordance with 1-NOP-03.05. As a result, on March 21, 2004, approximately two inches of level was observed lost from the refueling cavity, and about 6000 gallons of reactor coolant had been drained to the 1D Holdup tank.

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction 2515/150, Reactor Pressure Vessel Head and Head Penetration Nozzles (NRC Order EA-03-009) (Unit 1)

a. Inspection Scope

The inspectors observed activities relative to inspection of the reactor pressure vessel head (RPVH) nozzles in response to NRC Bulletins 2001-01, 2002-01, 2002-02 and NRC Order EA-03-009 Modifying Licenses dated February 20, 2004. The inspection included review of nondestructive examination (NDE) procedures, assessment of NDE personnel training and qualification, and observation and assessment of visual (VT) eddy current (ET) and ultrasonic (UT) examinations. Discussions were also held with contractor representatives and other licensee personnel. The activities were examined to verify licensee compliance with regulatory requirements and gather information to help the NRC staff identify possible further regulatory positions and generic communications. Specifically, the inspectors reviewed or observed the following:

(1) Bare Metal VT Examination

a. observed all or a portion of in-process RPVH bare metal remote video VT inspection of RPVH Nozzle Nos. 2, 17, 24, 37, 70, 71, 32, and 56 (including space around the nozzles)

b. reviewed all or a portion of RPVH bare metal VT video tape for RPVH Nozzle Nos. 11, 13, 14, 19, 27, 31, 34, 39, 41, 43, 46, 47, 51, 55, 59, 60, 65, and 66 and still digital pictures for Nozzle Nos. 11, 13, 14, 19, 27, 30, 31, 34, 39, 41, 43, 46, 47, 48, 49, 51, 55, 56, 59, 65, 66, 67,68, 74, 75, 76, and 77

The inspections were observed/reviewed in order to verify absence of boron crystals indicative of a leak and to verify the integrity of the RPVH.

(2) Volumetric UT Examination of RPVH Nozzles

a. observed a portion of in-process UT scanning of RPVH Nozzle Nos. 5, 11, 12, 33, 39 and 77

b. reviewed the UT results for RPVH Nozzle Nos. 6, 11, 34, 39, 72, 77, and vent line

c. observed data analysis and reviewed the UT results for RPVH Nozzle Nos. 1 and 20

UT observations/reviews included review of results intended to assess for leakage into the interference fit zone of the nozzles.

(3) Surface ET Examination

The inspectors observed initial in-process ET inspection activities for examining the surface of the vent nozzle J-groove weld. The examination was completed subsequent to the NRC inspection and, based on discussions with the licensee, no indications were identified.

(4) The inspectors reviewed and discussed with licensee personnel the susceptibility ranking calculation and the basis for the RPVH temperatures used in the calculation. The basis for RPVH temperature input was reviewed to verify appropriate plant specific information was used in the time-at-temperature model for determining RPVH susceptibility ranking.

(5) The inspectors reviewed licensee procedures and inspection results for visual examinations to identify potential boric acid leaks from pressure-retaining components above the RPVH.

b. Observations and Findings

1) Verification that the examinations were performed by qualified and knowledgeable personnel.

The inspectors found that visual and NDE inspections were being performed in accordance with approved and demonstrated procedures with trained and qualified inspection personnel. All examiners had significant experience, including experience inspecting RPVHs. In addition to qualification to Code requirements, VT, UT and ET personnel had additional training on RPVH inspections.

2) Verification that the examinations were performed in accordance with approved and demonstrated procedures.

The St. Lucie Unit 1 RPVH has 69 Control Element Drive Mechanism (CEDM) nozzles, 8 Incore Instrumentation (ICI) nozzles, and one vent nozzle, or a total of 78 nozzles. The bare head remote visual inspection was performed in accordance with Framatome Procedure 54-ISI-367. The procedure used high-resolution miniaturized cameras delivered by a flexible inspection guide tube (CIGAR - Combined Inspection Grappling and Retrieval) which scanned a portion of each nozzle and surrounding head material with each pass. The scans covered the full circumference at the nozzle-to-top-of-head interface areas of all of the 78 nozzles and surrounding head surfaces. Prior to the inspection, the licensee requested relaxation from full coverage of the bare metal visual inspection based on limitations encountered in the first bare metal inspection. The relaxation was documented in Relaxation Request Letter L-2003-283, dated November 21, 2003. However, due to improved inspection tooling, the licensee was able to meet the visual coverage requirements of the NRC Order. Therefore, the Relaxation Request was withdrawn by Letter L-2004-085 dated April 6, 2004.

All 78 nozzles received remote mechanized UT examination from the inside surface in accordance with Framatome approved Procedures 54-ISI-100-11 and 54-ISI-137-03 (vent nozzle only). Sixty-seven of the 69 CEDM Nozzles had thermal sleeves and were UT inspected with a blade probe. The blade probe employed “time-of-flight” (TOFD), 7.0 MHz, 50 degree, L-Wave (longitudinal) transducers with the ultrasonic beam directed circumferentially and scanning vertically. The 8 ICI nozzles and 2 of the CEDM nozzles with previously severed thermal sleeves were UT inspected using an open-bore tool with all UT transducers mounted in a single inspection module and scanning axially (vertical up and vertical down). For these nozzles, the examination employed the TOFD technique using two sets (one 30 degree and one 45 degree) of 5 MHz, L-Wave transducers with the 30 degree directed in the axial direction and the 45 degree directed in the circumferential direction. In addition, the nozzle volume was scanned with two 60 degree, 2.25 MHz, shear wave transducers (one directed axially and one directed circumferentially) and a 0 degree, 5 MHZ L-Wave transducer. The inspection area extended from a minimum of 2" above the J-groove welds to the bottom of the nozzles, except for 17 of the CEDM nozzles. For these 17 nozzles a step counterbore above the J-groove welds limited the minimum inspection distance above the weld to less than 2" (1.65" to 1.95") on the uphill arc of the welds.

This limitation was submitted to the NRC by Relaxation Request Letter L-2003-283, dated November 21, 2003, as modified by Letters L-2004-085, dated April 6, 2004, and L-2004-088, dated April 8, 2004.

The vent nozzle inside surface was scanned with a 0 degree, 5.0 MHz, L-Wave transducer; a 45 degree, 5.0 MHz, shear wave transducer (axial flaw detection); and a 70 degree, 5.0 MHz, shear wave transducer (circumferential flaw detection). The surface of the J-groove weld for the vent nozzle was examined using ET inspection to evaluate for leakage through the J-groove weld.

The inspectors reviewed the Framatome procedures and observed in-process examinations as noted above. Approved acceptance criteria and/or critical parameters for RPVH leakage were applied in accordance with the demonstrated procedures.

3) Verification that the licensee was able to identify, disposition, and resolve deficiencies.

All indications of cracks, leakage or head wastage were required to be reported for further inspection and disposition. Based on observation of the inspection process, the inspectors considered deficiencies would be appropriately identified, dispositioned and resolved. No cracks, leakage or wastage were identified.

4) Verification that the licensee was capable of identifying the primary water stress corrosion cracking (PWSCC) and/or RPVH corrosion phenomenon described in NRC Order EA-03-009.

The licensee performed NDE examinations and bare metal visual inspection of all of the RPVH nozzles and the RPVH surfaces during the outage. As noted above, the NDE techniques had been previously demonstrated under the MRP Inspection Demonstration Program as capable of detecting PWSCC type manufactured cracks as well as cracks from actual samples from another site. Based on the demonstration, observation of in-process inspections, and review of inspection data for NDE and bare metal visual inspections, the inspectors concluded the licensee was capable of identifying cracking and/or corrosion as described in the NRC Order.

5) Evaluate condition of the reactor vessel head (debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions).

Although debris was observed, it did not appear to be associated with leaks from above the head or with nozzle leaks. The inspection process required documenting any debris that might interfere with observation of the head to nozzle interface area and later removal of the debris with followup re-inspection to ensure the debris had not masked any boric acid deposits. This allowed 100 percent visual inspection of each of the RPVH nozzles with no significant obstructions impeding the examination.

6) Evaluate ability for small boron deposits, as described in NRC Bulletin 2001-01, to be identified and characterized.

The inspectors observed that the resolution of the video camera provided capability of detecting any debris or small boron deposits on the bare metal head. As noted above there were no obstructions to preclude essentially 100% visual inspection of the RPVH penetrations. As noted above, the loose debris noted at the RPVH to nozzle areas, was to be removed and the area re-inspected. In addition to the video, a series of good resolution digital still pictures were taken of each nozzle to head area to aid in interpretation of the video results.

7) Determine the extent of material deficiencies (i.e., cracks, corrosion, etc.) that required repair.

No examples of RPVH leakage or material deficiencies were identified during the visual or NDE examinations.

8) For each inspection method, determine if any significant impediments (e.g., centering rings, insulation, thermal sleeves, nozzle distortion, etc.) to effective examinations were identified.

Sixty-seven nozzles had thermal sleeves, requiring the use of “blade probe” inspections. However, this did not preclude UT inspections of the nozzle inside diameters. As noted above, a step counterbore precluded UT inspection of the required minimum of 2" above the J-groove weld for 17 CEDM nozzles. Loose insulation particles and debris had to be removed to allow visual inspection of some nozzles. No other significant items that could impede the examination processes were noted during observation of the visual or NDE examinations.

(9) Determine the basis for the temperatures used in the susceptibility ranking calculation. Were the temperatures plan-specific measurements, generic calculations, etc.?

The inspectors reviewed the susceptibility calculation and the basis for the RPVH temperatures used in the calculation, as documented in FPL Engineering Evaluations and FPL Letters listed in List of Documents Reviewed (Attachment 1) below. The RPVH temperature used for the calculation was taken from Combustion Engineering Owners Group (CEOG) Report CE-NPS-1074, which documented an analysis of core bypass flow to determine a reduction from T-hot called T-mix.

10) Determine if the methods used for disposition of NDE identified flaws were consistent with NRC flaw evaluation guidance. If not, was the method more restrictive?

No flaws were identified.

11) Determine if procedures existed to identify potential boric acid leaks from pressure-retaining components above the RPVH and if the licensee performed proper followup for indications of boric acid leaks.

Operating Procedure 1-0120022 requires inspection of the reactor vessel head area and components above the head for evidence of leakage. Documentation and disposition of inspection findings are performed in accordance with Plant Administrative Procedure ADM-29.03, Boric Acid Corrosion Control Program. The inspectors reviewed the completed inspection results for Procedure 1-0120022 for the current Unit 1 outage. Evidence of leakage (boric acid crystals) was identified at the swage lock fittings on the ICI flanges and a mechanical plug on CEDM No. 65. These conditions were documented and dispositioned in CRs 04-1250 and 04-1265. There was a history of previous ICI flange leakage.

.2 (Discussed) NRC TI 2515/154; Spent Fuel Material Control and Accounting at Nuclear Power Plants (Units 1 and 2)

Temporary Instruction 2515/154, Spent Fuel Material Control and Accounting at Nuclear Power Plants, Phase I and II, were completed during this inspection period for Unit 1 and Unit 2. Appropriate documentation was provided to NRC management as required. Although the inspection was completed during this report period, this TI will remain open pending further review and analysis. It should also be noted, that the licensee initiated CR 04-2846 to address issues identified during the course of the inspection.

.3 (Discussed) NRC TI 2515/156, Offsite Power System Operational Readiness (Units 1 and 2)

a. Inspection Scope

The inspectors collected data from licensee maintenance records, event reports, corrective action documents and procedures and through interviews of station engineering, maintenance, and operations staff, as required by the Temporary Instruction (TI) 2515/156. The data was gathered to assess the operational readiness of the offsite power systems in accordance with NRC requirements such as Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 17; Criterion XVI of Appendix B to10 CFR Part 50, Plant Technical Specifications (TS) for offsite power systems; 10 CFR 50.63; 10 CFR 50.65 (a)(4), and licensee procedures. Documents reviewed for this TI are listed in the attachment.

b. Findings

No findings of significance were identified. Based on the inspection, no immediate operability issues were identified. In accordance with TI 2515/156 reporting requirements, the inspectors provided the required data to the headquarters staff for further analysis. This TI will remain open pending completion of that analysis. It should also be noted, that during the course of the inspection, the licensee identified and reported a condition prohibited by TS due to nonconservatisms in the original electrical distribution analysis (refer to licensee event report (LER) 2004-001-00 dated June 4, 2004).

.4 World Association of Nuclear Operators (WANO) Peer Review and Institute of Nuclear Power Operations (INPO) Plant Evaluation Report

In June 2003, inspectors reviewed the WANO/INPO Final Report for the February 2003 peer review and plant evaluation. The inspectors did not identify any significant safety issues that warranted additional NRC followup.

4OA6 Meetings

Exit Meeting Summary

The inspector presented the inspection results to Mr. Bill Jefferson and other members of licensee management on July 8, 2004. Several interim exits were also held during the report periods by regional inspectors. The licensee acknowledged the findings presented. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

Supplemental Information

KEY POINTS OF PERSONS CONTACT

Licensee Personnel

G. Alexander, ISI coordinator
M. Alfonso, Corrective Action Program Supervisor
R. Boggs, Materials Specialist, CSI
D. Calabrese, Emergency Planning Supervisor
C. Costanzo, Operations Manager
R. De La Espriella, Site Quality Manager
L. Edwards, Training Manager
R. Gil, CSI Manager
R. Hughes, Site Engineering Manager
E. Katzman, Performance Improvement Department Manager
G. Johnston, Plant General Manager
W. Jefferson, Site Vice President
J. Martin, Operations Support Supervisor
R. McDaniel, Fire Protection Supervisor
D. Mothena, Manager - Plant Support Services
R. Murle, Acting Maintenance Manager
D. Nowokowski, CSI NDE Specialist, Level III Examiner
W. Parks, Operations Supervisor
T. Patterson, Licensing Manager
J. Porter, Operations Support Engineering Manager
G. Swider, Systems Engineering Manager
J. Tucker, Work Control Manager
G. Hollinger, Acting Security Manager
S. Wisla, Health Physics Manager

Contractors

R. Garrison, Senior Outage Manager, Framatome
T. Rockwood, Reactor Head Inspection Task Leader, Framatome

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

NRC personnel

B. Moroney, NRR Project Manager

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Closed

2515/150	TI	Reactor Pressure Vessel Head and Head Penetration Nozzles (NRC Order EA-03-009) (Section 4OA5.1)

Opened and Closed

05000335, 389/2004004-01	NCV	Failure to Follow Procedures per TS 6.8.1.(a)
05000335/2004004-02	NCV	Inadequate Corrective Actions to Preclude Repetitive Torque Switch Failure in Close Control Circuit of Unit 1 Valve MV-21-3 (Section 1R12)
05000335/2004004-03	NCV	Improper Configuration Control Of Shutdown Cooling Purification System Led To Inadvertent Reactor Coolant System Leak That Resulted In An Unusual Event (Section 1R14)

Discussed

2515/154	TI	Spent Fuel Material Control and Accounting at Nuclear Power Plants (Units 1 and 2) (Section 4OA5.2)
2515/156	TI	Offsite Power System Operational Readiness (Units 1 and 2) (Section 4OA5.3)

Documents Reviewed

<u>1R08 - Inservice Inspection (ISI)</u>

<u>Procedures:</u>

- Procedure 1-0120022, “Reactor Coolant System Leak Test”, Rev. 38
- Procedure 51- 5037733-00, “St. Lucie (PSL) Unit 1 Eddy Current Data Analysis Guidelines 2004"
- Procedure GMP-14, “Inspection and Maintenance of Crosby Relief Valves”, Rev. 8A
- Procedure 0-GMP-27, “Relief Valve Testing Using the <u>Old</u> Test Bench”, Rev. 2B
- Procedure QI-3-PSL-1, “Design Control”, Rev. 10
- Procedure ADM-29.03, “Boric Acid Corrosion Control Program”, Rev. 1
- Procedure CSI-NDE-03-098, “Eddy Current Examination Implementation for Steam Generator Tubing at St. Lucie Unit #1", Rev. 1
- Procedure 54-ISI-400-12, “Multi-Frequency Eddy Current Examination of Tubing”, Rev. July 2, 2003
- NDE 9.3, Rev. 0, “Radiographic Examination General Requirements
- NDE 5.2, Rev. 13, Ultrasonic Examination of Ferritic Piping Welds

<u>ASME Section XI Code Repairs/Replacements:</u>

- SL1-02043, ASME Class 2 Replacement, “Replace Charging Block Vent Assembly in Charging Pump with Butt Weld”
- SL1-02042, ASME Class 3 Replacement, “Replace Relief Valve V2345 and Replace with New Valve”

<u>Condition Reports (CR):</u>

- 04-1208, Dried Boric Acid Identified at the RCP Seal Bolting During Initial Walkdown, 4/07/2004
- 04-1215, 19 Packing Leaks (DBA) that were identified on the Initial RCB Walkdown, 4/10/2004
- 04-1241, Evidence of Dry Boric Acid Noted at Body to Bonnet Flange, 4/07/2004
- 04-1246, Evidence of Dry Boric Acid at Valve Packing Leak/ or Body Bonnet, 04/07/2004
- 04-1280, Packing Leaks at all Valves found during initial RCS Walkdown, 3/27/2004

<u>QA Audits, Self Assessments and Evaluations:</u>

- CSI-NDE-03-103, “ISI Program Procedure Adequacy and Implementation”, 12/26/03
- PSL-ENG-SEMS-03-065, “Degradation Assessment for St. Lucie Unit 1 Steam Generators Update for End-Of-Cycle 18, Refueling Outage”, Rev. 0
- QSL-ENG-03-06, “Site Engineering Functional Area Audit”, 8/27/2003-11/17/2003
- PSL-ENG-LRAM-00-090, “Boric Acid Wastage Surveillance Program- License Renewal Basis Document”, Rev 2
- PSL-ENG-LRAM-00-111, Alloy 600 Inspection Program- License Renewal Basis Document”, Rev 1
- AES 03095162-1-1, “Degradation Assessment for St. Lucie Unit 1, Steam generators Update for End-Of-Cycle 18 Refueling Outage”, Rev 1

Other Documents:

- SPEC-M-038, “Safety Related Relief Valve Setpoints”, Rev. 3
- NDE 9.3, “Radiographic Check List Weld #: CH-210-2004”, Rev 0
- NDE 9.3, “Radiographic Check List Weld #: CC-38-2000 (P-22)”, Rev 0
- ISI-PSL-1-2002, “Form OAR-1 Owner’s Activity Report”
- Ultrasonic Examination “Calibration Data Sheet and Summary” for Weld RC-115-FW-2-500F for RCP, Loop A SG-1A to RCP 1A2"
- Ultrasonic Examination “Calibration Data Sheet and Summary” for Weld RC-112- FW-2-500H for RCP, Loop A SG-1A to RCP 1A1"

4OA5.1 - NRC Temporary Instruction 2515/150, Reactor Pressure Vessel Head and Head Penetration Nozzles (NRC Order EA-03-009) (Unit 1)

Document 104-1221117, St. Lucie - LUCIE 1 (EOC18) Integrated Outage Project Plan, Revision 0

Framatome St. Lucie Unit 1 Head Inspection Workscope March 2004

Framatome St. Luce Unit 1 Bare Head Visual Inspection Workscope 2/16/04

Document 6015743A Reactor Head Nozzle Penetration Remote Visual Inspection Plan For St. Lucie Unit 1, Revision 03

Framatome ANP Nondestructive Examination Procedure 54-ISI-367-06, Procedure for Visual Examination for Leakage of Reactor Head Penetrations, Revision 06

Framatome NDE 108.0, Task Lesson Plan Bare Head Inspection, Revision 1

Framatome ANP Nondestructive Examination Procedure 54-ISI-100-11, CRDM, CEDM and ICI Accusonex Automated Data Acquisition and Analysis System, Revision 11

Framatome ANP Nondestructive Examination Procedure 54-ISI-137-03, Remote Ultrasonic Examination of Reactor Vessel Head Vent Line Penetrations, Revision 03

Materials Reliability Program: Demonstrations of Vendor Equipment and Procedures for the Inspection of Control Rod Drive Mechanism Head Penetrations (MRP-89) (September 2003)

Framatome ANP Nondestructive Examination Procedure 54-ISI-460-01, Eddy Current Method, Revision 01

Framatome ANP Procedure Qualification 54-PQ-460-01-00, Eddy Current Method

EPRI NDE Center Demonstration of the Framatome ANP Eddy Current Examination for RVHP Attachment Weld

FPL Letter FPL-2002-061, St. Lucie Units 1 and 2, Turkey Point Units 3 and 4, Response to NRC Bulletin 2002-001, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, dated April 3, 2002

FPL Letter FPL-2003-007, NRC Bulletin 2002-01, Request for Additional Information Response, dated January 31, 2003

FPL Letter No. L-2003-283, St. Lucie Unit 1 Order EA-03-009, Relaxation Requests 1 and 2, dated November 21, 2003

FPL Letter No. L-2004-071, St. Lucie Unit 1 Order EA-03-009 Relaxation Requests 1 and 2, dated March 23, 2004

FPL Letter No. L-2004-085, St. Lucie Unit 1 Order EA-03-009 Relaxation Requests 1 and 2, dated April 6, 2004

FPL Letter No. L-2004-088, St. Lucie Unit 1 Order EA-03-009 Relaxation Request 2, dated April 8, 2004

PSL-ENG-SESJ-03-053, St. Lucie Unit 1 Engineering Evaluation, Relaxation from the Requirements of NRC Order EA-03-009

St. Lucie Plant Administrative Procedure ADM-29.03, Boric Acid Corrosion Control Program, Revision 0

St. Lucie Unit 1 Operating Procedure 1-0120022, Reactor Coolant System Leak Test, Revision 38, including Appendix C (Reactor Coolant System Leak Test) for the current outage

Engineering Examination Report per St. Lucie Plant Administrative Procedure ADM-29.03, Boric Acid Corrosion Control Program, documenting the results of the inspection evidence of leaks of components above the Unit 1 RVH

Condition Report 04-1265, Trace Indications of Leakage at ICI Flanges

Condition Report 04-1250, Dry Boric Acid Deposit at Vent Ball Plug on CEDM 65

Personnel Certification Records for Framatome Inspection Personnel, including:

- Personnel Training Letters dated 9/9/02 and 9/18/03
- St. Lucie - LUCIE1 (EOC18) Training Matrix dated 3/22-4/23/2004
- St. Lucie - LUCIE1 (EOC18) CRDM Nozzle Inspection W/SUMO-ROCKY BUT Training Matrix dated 3/22/2004-4/23/2004
- Individual Examiner Certification, Training, and Eye Test Records for 7 NDE Examiners

Framatome Equipment Certification Records for the following Inspection Equipment

- μTOMOSCAN Pulser-Receivers VH-8168, VH-8719, and VH-9036
- UT Blade Probes VH-7351, VH-8085, VH-8875, VH-9031, VH-9051, and VH-9057
- Multifunction NDE Unit VH-8726
- Eddy Current Probes 0403401 and 0403402
- Light Meters VH-8055 and VH-8927
- Calibration Standards 6011680-001, 02-5025415E, and 6012827D-A

FPL Letter L-2002-185, St. Lucie Units 1 and 2, Turkey Point Units 3 and 4, Response to NRC Bulletin 2002-02, Reactor Pressure Vessel Head Penetration Nozzle Inspection Programs

MRP 48 (PWR Materials Reliability Program)

PSL-ENG-SESJ-02-045, St. Lucie Units 1 & 2 Engineering Evaluation For Response to NRC Bulletin 2002-02, Revision 1

PSL-ENG-SESJ-01-049, Engineering Evaluation, Response to the NRC Bulletin 2001-01 For St. Lucie Units 1 & 2, Revision 0

Spread Sheet Calculation for Effective Degradation Years (EDY)

Printout of St. Lucie Effective Full Power Hours

1R12 Maintenance Rule Implementation and
1REP Equipment Availability, Reliability and Functional Capability - Pilot

Maintenance Rule Quarterly Report Summary (Period: 2003-03)
Maintenance Rule Periodic Assessment (Period: September 2001 through March 2003)
System Checklist / Health Reports (Period: 2003-04)
ADM -17.08, Implementation of 10 CFR50.65, The Maintenance Rule
Systems and Components Engineering Department Guideline SCEG -008, Guideline for Maintenance Rule Periodic Assessments
QI-16-PSL-3, Corrective Action
ADM-17.2, Duties and Responsibilities of System and Component Engineering
Selected Expert Panel Meeting Minutes in 2003
Selected ADM-17.08, Figure 4 Attachments, Goal Setting and Monitoring

The following selected condition reports (CRs), associated with equipment problems for the Auxiliary Feedwater, Engineering Safety Feature Actuation Signal, Reactor Protection, and Intake Cooling Water systems, were specifically reviewed. This review included a sample of completed corrective actions, root cause determinations, and Maintenance Rule Program (a)(1) determinations, goal setting and status.

CR 01-2077	CR 03-1023	CR 02-3173	CR 01-2515
CR 01-3202	CR 03-3574	CR 03-0098	CR 03-0115
CR 03-2973	CR 03-3893	CR 03-4178	CR 03-0497
CR 03-3319	CR 03-4548	CR 03-4515	
CR 03-4018	CR 03-4552		

4OA5.3 - NRC Temporary Instruction TI 2515/156, Offsite Power System Operational Readiness (Units 1 and 2)

PSL-ENG-SEES-04-225, Revision 1, Address PSL Offsite Power System Operational Readiness Questions in NRC RIS 2004-05 and NRC Inspection Manual TI 2515/156

Power Systems and St. Lucie Plant Transmission Switchyard Interface Agreement

Memo to Power Supply System Operators and Dispatch Personnel dated April 21, 2004, regarding St. Lucie 230KV Switchyard Voltage Limits

SYNOPSIS

This investigation was initiated by the U.S. Nuclear Regulatory Commission, Office of Investigations, Region II, on March 12, 2004, to determine if a senior reactor operator (SRO), nuclear watch engineer, at the Florida Power and Light Company owned St. Lucie Nuclear Plant deliberately failed to comply with license conditions, specifically, not performing the functions of an operator or senior operator, as appropriate, in a technical specification-defined minimum staff position regarding the Licensed Operator Continuing Training Program (LOCTP) procedure.

Based on the evidence developed during this investigation, the allegation that the SRO deliberately circumvented license conditions associated with the LOCTP was not substantiated.

Approved for release on 7/26/04 - SES