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

May 2, 2003

Florida Power & Light Company
ATTN: Mr. J. A. Stall
Senior Vice President of Nuclear Operations
PO Box 14000
Juno Beach, FL 33408-0420

SUBJECT: TURKEY POINT NUCLEAR PLANT - NRC INTEGRATED INSPECTION REPORT 50-250/03-02, 50-251/03-02

Dear Mr. Stall:

On April 05, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Turkey Point Units 3 and 4. The enclosed integrated inspection report documents the inspection findings which were discussed on April 9, 2003, with Mr. T. Jones 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 two NRC-identified findings and one self-revealing finding of very low safety significance (Green). Two of the findings were determined to involve violations of NRC requirements. One violation was evaluated under the risk significance determination process and one was evaluated in accordance with Section IV of the NRC's Enforcement Policy and determined to be a Severity Level IV violation. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these two findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the 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 Turkey Point facility.

In accordance with 10 CFR 2.790 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 the NRC's document system (ADAMS). Adams is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

Docket Nos. 50-250, 50-251
License Nos. DPR-31, DPR-41

Enclosure: Inspection Report 50-250/03-02, 50-251/03-02
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:	50-250, 50-251
License Nos:	DPR-31, DPR-41
Report No:	50-250/03-02, 50-251/03-02
Licensee:	Florida Power & Light Company
Facility:	Turkey Point Nuclear Plant, Units 3 & 4
Location:	9760 S. W. 344th Street Florida City, FL 33035
Dates:	January 5, 2003 - April 5, 2003
Inspectors:	C. Patterson, Senior Resident Inspector K. Green-Bates, Resident Inspector (Also Sections 1R08.2, 4OA5) R. Reyes, Resident Inspector S. Vias, Senior Reactor Inspector (Section 1R08.1) B. Crowley, Senior Reactor Inspector (Section 4OA5) S. Ninh, Senior Project Engineer F. Jape, Senior Project Manager (Sections 1R02, 1R17) R. Chou, Reactor Inspector (Sections 1R02, 1R17) L. Mellen, Senior Emergency Preparedness Inspector (Sections 1EP1, 1EP4, 4OA1) W. Sartor, Senior Emergency Preparedness Inspector (Sections 1EP1, 1EP4, 4OA1) R. Baldwin, Senior License Examiner (Sections 1EP1, 1EP4, 4OA1) R. Aiello, Senior License Examiner (Sections 1EP1, 1EP4, 4OA1)
Approved by:	Joel T. Munday, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000250/03-02, IR 05000251/03-02; Florida Power & Light; 01/05/03-04/05/03; Turkey Point Nuclear Power Plant, Units 3 and 4; Inservice Inspection Activities, Refueling and Outage Activities, and Identification and Resolution of Problems.

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

A. Inspector Identified and Self-Revealing Findings

Cornerstone: Initiating Events Cornerstone

- Green. Inadequate root cause determination and corrective action of a failed Chemical and Volume Control System (CVCS) pipe support led to an additional failure.

  A Non-Cited violation of 10CFR50.55a(g)(4) and 10CFR50, Appendix B, Criterion XVI was identified in that measures taken to evaluate the suitability of replacement and to correct the cause for failure of CVCS pipe support H-4 in 1998 were not adequate and the same pipe support was found failed again in 2003. This finding was greater than minor because inadequate evaluation and corrective action to modify the pipe support and correct the cause for the 1998 failure could have challenged the ability of this line to supply reactor coolant pump seal cooling. The finding was of very low safety significance because the licensee determined that the loss of this support did not cause loss of function of the CVCS system. Specifically, the stress analysis showed that the pipe would not have been over stressed seismically or thermally with the loss of this hanger. (Section 1R08.1)

Cornerstone: N/A

- Green. Inappropriate blanket overtime authorization for operators, health physics personnel, and maintenance personnel was granted for the entire Unit 3 refueling outage.

  This finding is greater than minor because inappropriate deviations from overtime limits can be a significant contributor to worker fatigue and potential for human errors which, if left uncorrected, could become a more significant safety concern. This finding is of very low safety significance because once this issue was presented to licensee management at the start of the outage, action was taken to prevent inappropriate deviations from the guidelines and no violation of regulatory requirements occurred. (Section 1R20)

- Green. Main Steam Safety Relief Valve lift pressures were outside the ± 3% Technical Specification (TS) requirements for the past several refueling outages and were not reported as required in Licensee Event Reports (LERs).

  A non-cited violation of 10 CFR 50.73 (a)(2)(i)(B) was identified. This finding is greater than minor because failure to accurately report events could impact the NRC's ability to perform its regulatory function. This finding is of very low safety significance because the as-found main steam safety relief valve lift pressures of the affected valves were bounded by accident analyses. (Section 4OA2)

B. Licensee Identified Findings

None

REPORT DETAILS

Summary of Plant Status:

Unit 3 was manually tripped on January 27, 2003, when a partial loss of instrument air pressure resulted in a low steam generator water level. The unit returned to full power on January 31, 2003. On February 18, 2003, power was reduced to 60% when a feedwater pump was removed from service due to high vibration. On February 27, 2003 power was reduced to 46% for main steam safety valve testing, and the unit remained at reduced power until a refueling outage started on March 1, 2003. The unit returned to service on March 28, 2003.

Unit 4 reduced power to 40% on January 13, 2003, for turbine valve testing and to perform planned maintenance. The unit returned to full power on January 15, 2003.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity (Reactor-R)

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors performed a walkdown of the Unit 3A and Unit 3B emergency diesel generators (EDGs) to verify these systems would remain functional during cold weather conditions. The inspectors verified that compensatory actions using temporary space heaters were implemented for a degraded heater element. On January 14, 2003, the inspectors reviewed that the actions taken by plant operations due to the record cold weather were consistent with procedure 0-ONOP-03.2, Cold/Hot Weather Conditions. The inspectors reviewed the control room logs and checked that actions were taken at the appropriate thresholds.

b. Findings

No findings of significance were identified**.**

1R02 Evaluations of Changes, Tests or Experiments

a. Inspection Scope

The inspectors reviewed selected samples of evaluations to confirm that the licensee had appropriately considered the conditions under which changes to the facility or procedures may be made, and tests conducted, without prior NRC approval. The inspectors reviewed evaluations for 4 changes and additional information, such as calculations, supporting analyses, the Updated Final Safety Analysis Report (UFSAR), and drawings to confirm that the licensee had appropriately concluded that the changes could be accomplished without obtaining a license amendment.

The inspectors also reviewed 15 samples of changes which included design changes, commercial grade dedication packages, a temporary modification, and a procedure change for which the licensee had determined that evaluations were not required, to confirm that the licensee’s conclusions to "screen out" these changes were correct and consistent with 10 CFR 50.59. The safety evaluations and "screen outs" are listed in Section 1R17.

The inspectors also reviewed a recent audit of the 10 CFR 50.59 process to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated.

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 assess the operability of required redundant trains or backup systems while the other trains were inoperable or out of service. These inspections included reviews of plant lineup procedures, operating procedures, and piping and instrumentation drawings which were compared with observed equipment configurations to identify any discrepancies that could affect operability of the redundant train or backup system. The inspectors reviewed the following systems:

- 3A, 4A, 4B EDGs while 3B EDG was out of service to repair a heater element
- 3B boric acid transfer (BAT) pump while the 3A BAT pump was out of service
- 3A and 3B EDGs starting air system while cross-tied with two 3B EDG air receivers drained.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors toured the following eleven plant areas to evaluate conditions related to control of transient combustibles and ignition sources, the material condition and operational status of fire protection systems, and selected fire barriers used to prevent fire damage or fire propagation. The inspectors evaluated these conditions against provisions in the licensee’s off Normal Operating Procedure, 0-ONOP-016.8, Response to a Fire/Smoke Detection System Alarm, 0-SME-091.1, Fire and Smoke Detection System Annual Test, 0-ADM-016, Fire Protection Plan, and 10 CFR Part 50, Appendix R. The following areas were inspected:

- U3 and U4 Turbine Building Deck Area (Fire Zone 117)
- U3 and U4 Turbine Building Mezzanine Deck (Fire Zone 105)
- U3 480V Load Center (Fire Zone 94)
- U4 480V Load Center (Fire Zone 96)
- U3B 4160V Switchgear Rooms (Fire Zone 70)
- U4B 4160V Switchgear Rooms (Fire Zone 67)
- U3A EDG Engine Rooms (Fire Zone 73)
- U4B EDG Engine Rooms (Fire Zone 113)
- U3 West Electrical Penetration Rooms (Fire Zone 19)
- U4 West Electrical Penetration Rooms (Fire Zones 27)
- U3 Component Cooling Pump and Heat Exchanger Area (Fire Zone 54)

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI)

.1 Inservice Inspection Activities

a. Inspection Scope

The inspectors observed Unit 3 in-process ISI work activities during the second and final outage of the 2nd interval, 3rd ISI period and reviewed selected ISI records. The observations and records were reviewed for compliance to the Technical Specifications (TS) and the applicable Code (ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, with no Addenda). The following Unit 3 ISI examinations were observed:

•	Ultrasonic	(UT)	3-SGA-Y	C-A C1.30	(Tube sheet to SG stub barrel)
			3-SGA-N	C-A C1.10	(SG stub barrel to lower shell)
•	Visual	(VT-3)	3-CCH-27	F-A F1.30	
			3-CCH-25	F-A F1.30	
			3-CVC-4	F-A F1.30	
		(VT-1)	3-CCH-27	D-A D-1.20	

Qualification and certification records for examiners, equipment and consumables, and nondestructive examination (NDE) procedures for the above ISI examination activities were reviewed. In addition, a sample of ISI issues in the licensee's corrective action program were reviewed for adequacy. The following records/documents were reviewed:

NDE Examiner/QC Inspector Qualification Certification and Visual Acuity Records Examined

Examiner	Method-Level
MM	UT-II (PDI), VT-1-LII, VT-3-LII
PJ	UT-II (PDI)

NDE Equipment and Consumables

Probes:	45° SN: J26607, 60° SN: M31135, 0° SN: 48807
Thermometer:	SN:185141
Scope:	Sonic 136, SN: 136P1106C031382
Couplant:	Ultragel II - Batch 98325
Cal Block:	316 CS-7-TK-4

b. Findings

Introduction: The inspectors identified a self-revealing Non-Cited Violation (NCV) with very low safety significance (Green). The violation of 10CFR50.55a(g)(4), which requires meeting the ASME Boiler and Pressure Vessel Code, Section XI, and of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, resulted from the failure to properly evaluate and correct a pipe support failure on the chemical and volume control system (CVCS) pipe support H-4.

Description: In March, 2003, while performing a scheduled ASME Section XI VT-3 exam on CVCS pipe support H-4, in accordance with the Turkey Point Nuclear Power Plant Unit 3 Inservice Inspection Plan, ISI-PTN-3-Plan, Rev. 3, the licensee found that the pipe support structure had been torn from the base plate and was dangling from the pipe. The hanger was on a 2-inch diameter supply line from the charging pumps to the reactor coolant pump ‘C’ #1 seal. The inspectors determined that this same pipe support had failed similarly in 1998. Upon detection of the 2003 failure, the licensee evaluated the need to expand the sample size of the visual exams, and in accordance with the requirements of Code Case N-491-1, supports immediately adjacent to the broken support (H-2 & H-5) were determined to require an examination, plus 10 similar supports on this system. A VT-3 inspection of the 2 Hilti anchor bolts for support H-4 was also performed as well as a UT examination of the 2" schedule 160 pipe upstream and downstream of the broken support. To assess any potential damage to adjacent piping components, the licensee also performed a liquid penetrant (PT) examination on the adjacent 3/4" to 2" welded connection, and a PT examination on the welded connection of the 1-1/2" seal injection line to the 3P200C reactor coolant pump. For the expanded sample there were no rejected items reported. Engineering also performed an inspection of the H-4 support during the Reactor Coolant System Overpressure Leak Test (per 2-OSP-041.25), at operating temperature, and found sufficient clearance between the outside of the coupling weld and the H-4 support. The apparent cause was determined after the 2003 failure, to be thermal binding between the piping and the support. The clearance between the support frame and the weld of the adjacent 3/4" branch pipe coupling was not sufficient to accommodate the expected axial thermal movement (1-3/8"). An inspection of the as-found support condition indicated that at most, there was a ½" clearance between the outside of the weld and the box frame of the support. Another contributing factor was that the orientation of the angle from the wall plate was not perpendicular to the axial direction of the associated 2" pipe. This resulted in the application of a horizontal load on the support that caused the failure of the weld at the connection of the angle to the base plate.

Analysis: The inspectors determined that this finding was associated with inadequate design evaluation and analysis and affected the Initiating Events Cornerstone. Inadequate evaluation and corrective action to modify the pipe support and correct the cause for the 1998 pipe support failure could have challenged the ability of this line to supply reactor coolant pump seal cooling. The licensee performed a review of the TES Technical Report TR-5322-28, Rev. 1 CVCS, Inside Containment, Stress Problem CVCS-17 & 18 and determined that had a design basis event occurred with the degraded support condition, an adequate margin existed in the piping system to accommodate piping stresses allowed under functionality criteria of Standard CN-3.01, Rev. 3, Piping and Support Analysis Requirements for Turkey Point Units 3 & 4. Since there was no loss of function of the system, the finding was evaluated as Green (very low safety significance).

Enforcement: 10CFR50.55a(g)(4) specifies in part that components classified as ASME Code Class 1, Class 2, and Class 3 meet the requirements set forth in Section XI of the ASME Boiler and Pressure Vessel Code. The ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, with no Addenda, subsection IWA-7220, states in part that "Prior to authorizing the installation of an item to be used for replacement, the Owner shall conduct an evaluation of the suitability of that item. If a replacement is required because of a failure of an item, the evaluation shall consider cause(s) of failure of the existing item to ensure that the selected item is suitable."

10CFR50, Appendix B, Criterion XVI, Corrective Action, states in part that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

Contrary to the above, as of March 6, 2003, measures taken to evaluate the suitability of replacement and to correct the cause for failure of CVCS pipe support H-4 in 1998 were not adequate, and resulted in a repeat failure of the same pipe support. Because the finding is of very low safety significance and is in the licensee's corrective action program as CR 03-0452, it is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-250/03-02-01, Inadequate Corrective Action For a Failed CVCS Pipe Support.

.2 Unit 3 Steam Generator (SG) Inspection

a. Inspection Scope

The inspector reviewed the implementation of the licensee's inservice inspection program for monitoring degradation of the U3 steam generators (SG), a reactor coolant system boundary component. The inspectors observed and/or reviewed selected inspection records for:

- Eddy current examination (ET) and data acquisition for eight inservice SG tubes (Framatome).

- Unit 3 SG Secondary Side Integrity Plan foreign object search & retrieval (FOSAR) visual secondary side inspections of SGA, SGB and SGC due to the potential of steam generator feed pump casing debris entering the generators.

- 2003 ET data analysis and history (Framatome) and resolution (Zetec) for three inservice tubes in SGA, two in SGB and two in SGC.

a. 3 SG tube repairs (plugging) determined as a result of the Unit 3 SG ET inspection.

The records were compared to the TS, License Amendments, and Licensee Commitments such as: NEI 97-06 Steam Generator Program Guidelines, EPRI PWR Steam Generator Examination Guidelines, and applicable industry results from examinations of similarly designed (Model F) steam generators to verify compliance. The inspectors also verified that the ET equipment setup parameters, methodology and equipment used were in accordance with Turkey Point Unit 3 Component Specific Technique Sheets and that Steam Generator Integrity Program commitments made as part of the License Renewal Aging Management program (as stated in Chapter 16 of the UFSAR) were met.

As the licensee had changed inspection vendors from Westinghouse to Framatome, the inspectors verified that past SG database history files were compatible with the software being used by the new vendor. Inspectors interviewed the independent qualified data analyst (Zetec) and evaluated to determine that the ET performed consistently detected previously identified tube imperfections such as dents, tube wear, and manufactured burnish marks at the expected locations, which would indicate an effective inspection. The inspectors evaluated to determine that SG plugging limits had not been exceeded and that site procedures had been reviewed and accepted by the Authorized Nuclear Inservice Inspector. The inspectors reviewed selected condition reports from the outage to assess whether the identification of SG problems was at an appropriate threshold.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On February 28, the inspectors observed and assessed licensed operator actions on the Unit 3 simulator concerning residual heat removal system scenarios. Procedures 3-ONOP-050, Loss of Residual Heat Removal System (RHR); 3-GOP-305, Hot Standby to Cold Shutdown; 3-OP-050 Section 5.1, Place RHR in Operation for Cooldown, were observed to assess licensee implementation. The inspectors specifically evaluated the following attributes related to operating crew 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 Emergency Operating Procedures and Emergency Plan Implementing Procedures
- 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 training critique

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the following three equipment problems and associated CRs to verify that the licensee's maintenance efforts met the requirements of 10 CFR 50.65 "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" and Administrative Procedure 0-ADM-728. The inspectors' efforts focused on maintenance rule scoping, characterization of the failed components, risk significance, determination of a(1) 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.

- CR 02-2157 4B EDG Relay Failure
- CR 03-0304 4C Charging Pump Failure
- CR 03-0147 Process Radiation Monitor, R-15, Spiking

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the following six emergent items, as described in the referenced CRs or safety evaluation. The inspectors verified that the emergent work activities were adequately planned and controlled, as described in 0-ADM-068, Work Week Management and O-ADM-225, On Line Risk Assessment and Management. The inspectors verified that, as appropriate, contingencies were in place to reduce risk, minimize time spent in increased risk configurations, and avoid initiating events. The following items were reviewed:

- CR 03-0195 — Charging Line Relief Valve Leaking
- CR 02-2157 — 4B EDG Relay Failure
- CR 03-0310 — 3B Main Feedwater Pump Vibration/Missing Parts
- CR 03-0275 — Part 21 Coatings - Issue
- CR 03-0380 — Intake Cooling Water (ICW) Pump - Voluntary Entry into Limiting Condition for Operation
- PTN-ENG-SENS-01-0097 — Temporary Lowering of Spent Fuel Pool Water Level for Maintenance Activities

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Non-Routine Plant Evolutions and Events

a. Inspection Scope

For the non-routine events described below, the inspectors reviewed operator logs, plant computer data, and strip charts to determine what occurred and how the operators responded, and to verify that the response was in accordance with plant procedures for the following:

- Manual Reactor trip of Unit 3 on January 27, 2003, due to degraded instrument air pressure and subsequent low level in the 3C steam generator. Plant operators quickly restarted locally the two diesel driven instrument air compressors that automatically started but tripped. The quick response to restore instrument air pressure helped to reduce the effects of this transient. The unit operator was proactive and manually tripped the reactor prior to an automatic trip on low steam generator water level (CRs 03-140, 03-146).

- Unit 3 power was reduced to 60% on February 18, 2003, due to high vibration of the 3B steam generator feed pump. A plant operator noticed this vibration and action was taken in time to remove the pump from service in an orderly fashion before complete failure of the pump (CRs 03-0301, 03-0488).

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following six operability determinations to ensure that TS operability was properly supported and the system, structure or component remained available to perform its safety function with no unrecognized increase in risk. The inspectors reviewed the UFSAR, applicable supporting documents and procedures, and interviewed plant personnel to assess the adequacy of the CR disposition.

- CR 03-0022 — Turbine Plant Cooling Water (TPCW) Heat Exchanger Isolation Valve Leakage
- CR 02-2445 — Auxiliary Feedwater (AFW) Flow Controllers found in Manual following test
- CR 03-0012 — High Head Safety Injection (HHSI) pump casing leakage
- CR 03-0197 — 4C ICW Pump Check Valve
- CR 03-0195 — Charging Line Relief Valve Leaking
- CR 03-0541 — Pin Hole Leak in the Unit 4 Letdown Line

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated design change request (DCR) packages and commercial grade dedication (CGD) packages for selected modifications, in the Initiating Events and Mitigating Systems cornerstone areas. The packages were evaluated to assess any adverse affects on system availability, reliability, and functional capability. The modifications and the associated attributes reviewed were as follows:

- PC/M-02-024 Intake Structure Strut Beam Repairs, Rev. 0 (Mitigating systems)
  Affected flowpaths to remaining loads
  Seismic considerations
  Critical characteristics, acceptance criteria, and method of acceptance
  Inspection requirements
  Material compatibility with original design for type, classification, and dimensions

- PC/M-02-012 Permanent Platforms/Scaffolds/Components in Containment, Rev. 0, (Initiating events)
  Materials/Replacement Components material compatibility,
  Code requirements
  Seismic considerations

- PC/M 00-009 Steam Generator Flexible Tube Stakes, Rev. 1 (Mitigating systems)
  Critical characteristics, acceptance criteria, and method of acceptance
  Inspection requirements
  License basis documents updated
  Materials/Replacement Components material compatibility, Code requirements
  Material compatibility with original design for type, classification, and dimensions
  Materials type/classification/pressure boundary
  Necessary pressure boundaries re-established
  Plant procedure and critical drawing updating
  Seismic considerations
  Supporting license basis and safety evaluation documentation

- JPN-PTN-SEMS-96-003 Unit 4 Steam Generator Secondary Side Foreign Objects, Rev. 4 (Mitigating systems & Initiating events)
  Associated temporary modification risk assessment
  Corrective actions for post modification problems
  Critical characteristics, acceptance criteria, and method of acceptance
  Functional requirements to support design bases for flow and pressure control
  Inspection requirements
  Material compatibility with original design for type, classification, and dimensions

- PC/M-97-052 1 Hour Thermolag Upgrade for Outdoor Fire Zones, Rev. 1 (Mitigating system)
  Maintenance work order
  Engineering Evaluation
  Physical Walk-Down
  Procurement Documentation

- PC/M-97-057 25 Minute Thermolag Upgrade for Outdoor Fire Zones, Rev. 3 (Mitigating systems)
  Maintenance work order
  Engineering Evaluation
  Physical Walk-Down
  Procurement Documentation

- PC/M-02-111 Unit 3 SGBD HX Discharge to Canal, Rev. 1, (Mitigating systems)
  Material compatibility evaluation
  Heat removal
  Drawing updated
  Procedure updated
  Radiation monitoring
  Calculation to support modification
  Post maintenance testing
  Plant procedure and critical drawing updating

- PTN-ENG-00-0367 Safety Evaluation for Throttling CCW Manual Valves (Mitigating systems)
  Associated temporary modification risk assessment
  Corrective actions for post modification problems
  Critical characteristics, acceptance criteria, and method of acceptance

- PC/M-96-096 Plant/C Bus Reliability Improvements Modifications, Rev. 1, (Mitigating systems)
  Inspection requirements
  Materials type/classification/pressure boundary
  Updating of drawings and affected plant procedures

- PTN-ENG-SECS 98-058 Storage of Tools and Equipment in Containment During All Modes of Operation, Rev. 3, (Mitigating systems)
  Inspection Requirements
  Plant procedure and critical drawing updating

- PC/M-02-002 Unit 3 Containment Polar Crane Upgrade - 1B, Rev. 0 (Mitigating systems)
  Material Replacement compatibility
  Seismic consideration
  Improvement of operability and functionality
  Environmental consideration
  Safety enhancement

- PTN-ENG-SENS-01-0057 Temporary Lowering of PTN-3 SFP Level, Rev. 1 (Initiating events)
  Radiation dose rate evaluation
  FSAR and Technical Specification Updated
  Alarm system

- MSP 02-010 Provide Cable Supports for 3A Battery Inter-Tier Cables, Rev. 0, (Initiating events)
  Seismic consideration
  Fire protection consideration
  Electric input limit
  Support evaluation
  Plant procedure and drawings updated
  Testing acceptance criteria

- MSP-02-029 TE-4-454 Use Spare Green Wire to Fix Low IR on White Wire, 4/2/02, (Initiating events)
  Inspection requirements
  Plant procedure and critical drawing updating

- MSP-02-123 Change Breaker Trip Setting on 40753 & 0863 for 4P25A & 4P24B Motor Replacement, 4/2/02, (Mitigating system)
  Characteristics, acceptance criteria, and method of acceptance
  Inspection requirements
  Plant procedure and critical drawing updating
  Post modification testing criteria and results

- MSP-02-030 TE-4-60A Repair Field Cable in Hagen Rack 19, 4/25/02, (Mitigating system)
  Critical characteristics, acceptance criteria, and method of acceptance
  Post modification testing criteria and results
  Testing acceptance criteria

- MSP-02-013 Replace Thermal Overload Heater Elements for 3A & 3C ECFF Motors, 2/28/02, (Mitigating system)
  Critical characteristics, acceptance criteria, and method of acceptance
  Inspection requirements
  Plant procedure and critical drawing updating
  Testing acceptance criteria

- JPN-PTN-SEMS-96-04 Safety Evaluation for the Temporary Installation of Drain Hoses and Performance of Hot-Spot Flushes on the RHR System, Rev. 1, (Mitigating systems).
  Associated temporary modification risk assessment
  Affected flowpaths of PSW to remaining loads
  Inspection requirements
  Updating of procedures reflect new setpoint

- MSP-02-126 Removal of Rigid Support 80117-R-334-05 from the AFW steam supply system, Rev. 0 (Mitigating systems)
  Functional evaluation
  Justification evaluation
  Drawing updated

The inspectors observed the as-built configuration for selected modification packages. Documents reviewed included procedures, engineering calculations, modifications, work orders, site drawings, corrective action documents, applicable sections of the living UFSAR, supporting analysis, Technical Specifications, and design basis information.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

For the following six post maintenance tests listed below, the inspectors reviewed the test procedures and either witnessed the testing and/or reviewed test records to determine whether the scope of testing adequately verified that the work performed was completed correctly and demonstrated that the affected equipment was functional and operable. The inspectors verified that the requirements of procedure 0-ADM-737, Post Maintenance Testing, were incorporated into test requirements. The inspectors reviewed the following list of tests:

- 4-OSP-047.1 — Charging Pump/Valves Inservice Test Following Charging Pump Maintenance
- WO 33000151-01 — HHSI Pump
- WO 32011508-03 — 3B EDG Heaters
- WO 32018259-01 — ICW to Component Cooling Water (CCW) Basket Strainer
- WO 32018259 — Calibration of ICW to CCW Heat Exchanger Gauge
- WO 33049534 — 3B EDG Erratic Voltage Control

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors reviewed the outage plans and contingency plans for the Unit 3 refueling outage, conducted March 1-28, 2003, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

Outage Risk

Prior to the start of the refueling outage the inspectors reviewed the outage risk assessment with the licensee. The outage risk status or color and plant evolutions during the outage were reviewed. The risk assessment was planned according to plant procedure, O-ADM-051, Outage Risk Management. During the outage the inspectors reviewed that the outage unit risk as described in daily status sheets was consistent with the plan.

Clearance Activities

The inspectors performed random checks of clearance activities during the outage to verify that activities were in accordance with procedure O-ADM-212 “In-Plant Equipment Clearance Orders”, and O-ADM-212.1 “Operations In-Plant Equipment Clearance

Orders". A detailed review was performed of a clearance error that rendered both trains of AFW inoperable during plant cooldown. The licensee intends to submit an LER for this error (CR 03-046).

Refueling Activities
The inspectors observed fuel offload activities in the control room and spent fuel pool areas. Core reload activities were observed and activities verified in accordance with procedure 3-OSP-040.2, Refueling Shuffle. The inspectors also reviewed the videotape of the core reload, and verified fuel bundles were placed in accordance with O-OSP-O59 Core Mapping Following Core Loading.

Containment Closeout
The inspectors conducted several walkdowns of containment during the refueling outage. On March 27, 2003, a final walkdown of containment was conducted while the unit was at normal operating temperature and pressure to inspect for reactor coolant system leaks and debris that could enter the containment sumps.

Instrumentation
The inspectors verified the cooldown rate during the initial plant cooldown did not exceed TS limits. System pressures and level indications were observed for proper operation during periods of reduced inventory to ensure adequate core cooling was maintained.

Electrical
The inspector monitored that electrical lineups during the outage were in accordance with the risk assessment plan. System configurations were monitored during planned electrical bus outages and engineered safeguards integrated testing to verify adequate power sources were maintained.

Spent Fuel Pool Cooling
The inspectors verified that the spent fuel pool cooling system was protected as described in the outage risk assessment. Temperatures were monitored when the core was completely offloaded to verify proper cooling. Activities that could affect water level were assessed using procedure 3-OSP-075.4, "Filling/Draining the Refueling Cavity and the SFP Transfer Canal".

Inventory Control
The inspectors monitored inventory control during the outage, and again when reduced inventory conditions occurred when the 3C reactor coolant pump seal had to be reworked, to verify proper indication of water level was maintained.

Startup
The inspector monitored plant heatup, initial criticality, and power ascension to verify mode changes were made with the required equipment operable. Reactor coolant system boundary leakage was monitored to verify leakage requirements were met.

b. Findings

Introduction: A Green finding was identified by the inspectors when blanket authorization was granted to exceed the working hour guidelines for the plant staff who perform safety-related functions. The inspectors determined that at the start of the Unit 3 refueling outage, authorization was given to all plant operators, health physics technicians, and maintenance personnel to exceed the TS 6.8.5 requirement for administrative control of working hours.

Description: On March 3, 2003, the inspectors identified that Operations personnel were scheduled to work continuous 12 hour shifts for the duration of the outage. The inspectors noted that this schedule would exceed the established limit of 72 hours a week.

The inspectors discussed this issue with licensee management and learned that executive correspondence on March 1, 2003 gave blanket authorization to deviate from the overtime guidelines per QI I-PTN-1. This authorization was given to operations, health physics, and maintenance departments.

Analysis: The inspectors reviewed TS 6.8.5 that required administrative procedures to limit working hours. The plant procedure QI I-PTN-1, Organization, states the following:

1. No more than 24 hours worked in any 48 hour period.
2. No more than 72 hours worked in any consecutive 7 day period.
3. At least 8 hours break in between work periods.

The executive correspondence gave authorization to exceed all of these limitations.

The inspectors determined that this practice would be in violation of the TS. NRC Generic Letter 82-12, Nuclear Power Plant Staff Working Hours, specified limits on overtime, and stated that deviations from the limits were to be for “very unusual circumstances”. Inappropriate deviations for exceeding the overtime limits can be a significant contributor to worker fatigue and potential for human errors which, if left uncorrected, could become a more significant safety concern. The inspectors concluded that blanket authorization for the entire 18 day refueling outage was not a “very unusual circumstance”. This issue was promptly discussed with licensee management and action was taken to limit overtime in accordance with the established guidelines. The inspectors determined that the finding was of very low safety significance (Green) because the blanket authorization was retracted prior to the TS limits being challenged. This finding is in the licensee’s corrective action program as CR 03-0647

Enforcement: No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a non-compliance in that action was taken by the licensee to restrict the hours worked prior to exceeding the stated limits.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors either reviewed or witnessed the following surveillance tests to verify that the tests met the TS, the UFSAR, and licensee procedure requirements and demonstrated the systems were capable of performing their intended safety functions and their operational readiness.

- 3-OSP-075.1 Auxiliary Feedwater Train I Operability Verification (‘A’ Pump)
- 4-OSP-075.7 Auxiliary Feedwater Train 2 Backup Nitrogen Test
- 0-OSP-200.5 Miscellaneous Tests, Checks and Operating Evolutions
- 0-SMI-067.4 Control Room HVAC Radiation Monitors RAI-6642 and RAI-6643 Monthly Operational Test
- 3-OSP.041.1 Reactor Coolant System Leak Rate Calculation
- 4-OSP-024.2 U4 EDG Load Sequencing Test
- 3-OSP-072.5 Main Steam Safety Valve Setpoint Verification
- 3-OSP-203.1 Train A Engineered Safeguards Integrated Test
- 3-OSP-203.2 Train B Engineered Safeguards Integrated Test
- 3-OSP-0061 Load Center 3H Transfer Function and 480 VAC Degraded Voltage
- 3-OSP-075.4 Auxiliary Feedwater Auto-Start Test

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following two active temporary modifications to verify that risk significant items did not adversely affect the operation of a system that was altered. The inspectors reviewed plant procedure 0-ADM-503, “Control and Use of Temporary System Alterations (TSA)”, to verify that the modifications were controlled as required by procedure. In addition, the inspectors toured plant areas and specifically looked for any temporary modifications that the licensee might not have identified. The following active temporary modifications were reviewed:

- TSA-09-03-013.01 Temporary System Alternative to the Unit 3 and 4 Instrument Air Systems
- TSA-04-03-047-01 RV-4-311 Setpoint Adjustment

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness (EP)

1EP1 Exercise Evaluation

a. Inspection Scope

The inspectors reviewed the emergency exercise and scenario for the biennial, full participation 2003 emergency response exercise for Turkey Point. The review covered whether the licensee created a scenario suitable to test the major elements of their emergency plan in accordance with 10 CFR 50, Appendix E.

During the period February 18 - 21, 2003, the inspectors observed and evaluated the licensee's performance in the exercise, as well as selected activities related to the licensee's conduct and self-assessment of the exercise. The exercise was conducted on February 19, 2003. Licensee activities inspected during the exercise included those occurring in the Control Room Simulator (CRS), Technical Support Center (TSC), Operational Support Center (OSC), and Emergency Operations Facility (EOF). The NRC's evaluation focused on the risk-significant activities of event classification, notification of governmental authorities, onsite protective actions, offsite protective action recommendations, and accident mitigation. The inspectors also evaluated command and control, the transfer of emergency responsibilities between facilities, communications, adherence to procedures, and the overall implementation of the emergency plan. The inspectors attended the post-exercise critique to evaluate the licensee's self-assessment process, as well as the presentation of critique results to plant management.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Response Plan Changes

a. Inspection Scope

The inspector reviewed changes to the Radiological Emergency Plan (REP) and determined there had been no plan changes since the last review.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

On January 16, 2003, the inspectors monitored from the Technical Support Center the first quarter EP drill of the site emergency response organization. During this drill the inspectors assessed if actions taken for emergency classification, notification, and protective action recommendations were made in accordance with implementing procedures. Additionally, the inspectors evaluated the adequacy of the post drill critiques conducted.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors reviewed licensee submittals for the performance indicators (PIs) listed below for the period from January 2002 through December 2002 for Units 3 and 4. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, “Regulatory Assessment Indicator Guideline,” Rev. 2, were used to verify the basis in reporting for data element.

Reactor Safety Cornerstone

- Unplanned scrams per 7000 critical hours
- Scrams with loss of normal heat removal
- Unplanned power changes per 7000 critical hours

The inspectors reviewed operating reports, plant procedure 0-ADM-032, NRC Performance Indicators, and NRC Inspection reports to verify the reported PI data was complete and accurate.

Emergency Preparedness Cornerstone

- Emergency Response Organization (ERO) Drill/Exercise Performance
- ERO Drill Participation
- Alert and Notification System Reliability

The inspector assessed the accuracy of the PI for ERO drill and exercise performance (DEP) over the past eight quarters through review of a sample of drill and event records. The inspector assessed the accuracy of the PI for ERO drill participation during the previous 8 quarters for personnel assigned to key positions in the ERO. The inspector assessed the accuracy of the PI for the alert and notification system reliability through review of a sample of the licensee’s records of the biweekly silent tests and quarterly full-cycle tests and observations of the physical condition of installed sirens.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

Annual Sample Review

a. Inspection Scope

The inspectors selected the following CRs for detailed review and discussion with the licensee. These CRs were examined to verify whether 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 and reportability; root cause or apparent cause determinations were sufficiently thorough; extent of condition, generic implications, common causes, and previous history were adequately considered; and appropriate corrective actions (short and long-term) were implemented or planned in a manner consistent with safety and TS compliance. The inspectors evaluated the CRs against the requirements of the licensee's corrective action program as delineated in Administrative Procedures ADM-518, Condition Reports, ADM-059, Root Cause Analysis, and 10 CFR 50, Appendix B.

- CR 03-0368 — Contingency CR to Address Potential Failure of Main Steam Safety Valve (MSSV) During Surveillance Testing
- CR 01-1818 — MSSV RV-3-1412 Tested Higher Than 3% High; RV-3-1411 Post Test Leakage Exceeded OSP Acceptance.
- CR 01-2196-0,1,2 — Main Steam Safety Valve Setpoint Verification Test; Failure of RV-3-1406 to Lift at it's Set Pressure
- CR 00-0676 — Overhaul of Relief Valve RV-3-1406; Valve Setpoint Tolerance

b. Findings and Observations

Introduction: A Green NCV of 10 CFR 50.73 (a)(2)(i)(B) was identified by the inspectors for failure to report past conditions prohibited by plant TS. The inspectors determined that over the past several years multiple main steam safety relief valves (MSSRVs) exceeded the lift setting tolerance of ± 3% as listed in TS Table 3.7-2.

Description: On February 27, 2003, during a Unit 3 MSSRV setpoint verification test, the inspector observed MSSRV RV-3-1402 exceed the lift setting by 4.8%. This exceeded the allowable tolerance of ± 3% and the licensee entered a 4 hour LCO as required by TS 3.7.1.1.b. The inspector discussed this issue with control room operators and determined that earlier that day another valve, RV-3-1410, was tested and found to actuate at 6% greater than the lift setting. However, the 4 hour LCO for RV-3-1410 was already exited based on two satisfactory retests and an operability evaluation documented under CR 03-0368. The cause of the higher lift pressure was concluded to be micro-bonding of the nozzle and disc (often called sticking). A total of four valves tested higher than the TS allowable limit. After discussion with the inspector, the licensee indicated that this issue would be documented in an LER.

The inspector reviewed historical records to assess past performance of MSSRVs and determined that multiple failures had occurred in past outages. The review included past completed surveillance test data since 1999 from procedure OSP-072.5, Main Steam Safety Valve Setpoint Verification Test as well as several completed CRs. In most cases there was one or more MSSRVs which lifted outside the TS limit. The inspectors identified several issues associated with the identification and resolution of these failures.

First, the past operability of the valve failures was not addressed. For example, CR 01-1818 discussed the fact that RV-3-1412 tested at 3.5% above set pressure. The CR documented the cause to be micro-bonding of the nozzle and disc, however nothing in the CR discussed how the micro-bonding occurred nor how it was resolved. Secondly, the reportability determination concluded that this condition occurred on a single valve and was not reportable in accordance with 10 CFR 50.73 (a)(2)(i)(B). This conclusion was based on having satisfied the TS requirement for each failed MSSRV and declaring it operable before the next MSSRV was tested. Third, four valves initially tested during startup following the outage, lifted outside the 3% band. CR's were not written on those failures because the licensee considered the tests to be as-left data, even though no work or tests had been performed on the valves during the shutdown or the outage. Several examples of past MSSRV test results are listed below:

Unit 3 March 24, 2000

Valve	Setpoint	Lift Pressure	± %
RV 1402	1100	969	-12%
RV 1406	1115	1174	+5.3%

Unit 3 September 29, 2001

Valve	Setpoint	Lift Pressure	± %
RV 1406	1100	1149	+4.5%
RV 1412	1115	1154	+3.5%

The inspectors reviewed NUREG-1022, Rev. 2, Event Reporting Guidelines 10 CFR 50.72 and 50.73, section 3.2.2, which discusses multiple test failures of safety valves found to lift with setpoints outside of TS limits. NUREG-1022 states that, "The existence of similar discrepancies in multiple valves is an indication that the discrepancies may well have arisen over a period of time and the failure mode should be evaluated to make this determination. If so, the condition existed during plant operation and the event is reportable under 50.73 (a)(2)(i)(B), "Any operation or condition prohibited by the plant's Technical Specifications"." The inspector concluded that the licensee had not reported past MSSRV lift test failures in accordance with this guidance.

Analysis: This finding was not assessed through the SDP but was reviewed by NRC management and was determined to be greater than minor because failure to accurately report events could impact the NRC's ability to perform its regulatory function. The finding is of very low safety significance because previous setpoint drifts were bounded by accident analyses.

Enforcement: 10 CFR 50.73 (a)(2)(i)(B) requires that any condition or operation prohibited by TS be reported in an LER. Contrary to the above, the licensee failed to report in an LER multiple cases where MSSRV testing results were greater than the 3% permitted by TS Table 3-7-2. This violation was evaluated in accordance with Section IV of the NRC's Enforcement Policy and determined to be a Severity Level IV violation. However, because the failure to report is considered to be of very low safety significance and has been entered into the licensee's corrective action program as CR 03-0762, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-250,251/03-02-02, Failure to Report Main Steam Safety Relief Valve Test Results Outside TS Limits.

The inspectors concluded that corrective actions for MSSRV setpoint drift problems were not comprehensive. Examples were identified where operabilty was not completely addressed. The CRs discussed reportability but did not include the test results of all valves or review past operability. Some of the test results which were initially outside the ±3% TS requirement, were not considered to be as-found data and therefore were not captured in the corrective action program. The valves were simply adjusted and re-tested until satisfactory results were obtained.

4OA3 Event Follow-up

(Closed) LER 50-250/2002-002-00, Operation with One Component Cooling Water Pump in Excess of Technical Specification Allowable Limits.

This issue was dispositioned as part of a Green Finding in NRC Inspection Report 250, 251/02-05 as a noncited violation 50-250, 251/2002-05-01. This LER is closed.

4OA5 Other Activities

(Closed) NRC Temporary Instruction 2515/150, Reactor Pressure Vessel Head and Head Penetration Nozzles (NRC Bulletin 2002-02)

a. Inspection Scope

The inspectors observed activities relative to inspection of the reactor vessel head penetration (VHP) nozzles in response to NRC Bulletins 2001-01, 2002-01, 2002-02 and NRC Order Modifying Licenses dated February 11, 2003. The inspection included review of nondestructive examination (NDE) procedures, assessment of NDE personnel training and qualification, and observation and assessment of visual (VT), 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. VT inspection using remote video of VHP Nozzle Nos. 1, 10, 14, 26, 30, 46, 59, and 60 - also observed partial inspection (at least one quadrant) of penetrations 4, 5, 9, 12, 13, 19, 20, 21, 22, 25, 28, 35, 36, 37, 38, 43, 44, 45, 47, 48, 55, 61, 62, and 67

2. UT in-process scanning and analysis of results for VHP Nozzle Nos. 45, 48, 61 and the vent line - also reviewed the UT results for VHP Nozzle Nos: 23, 48, and RVLMS Nozzles 59 and 60 (Two VHP nozzles modified for the reactor vessel level measurement with a guide sleeve installed along with a welded end plate, had to be removed for the inspection)

3. Liquid Penetrant (PT) inspection of Head Vent Line Nozzle J-groove weld performed to compensate for not having an interference fit.

Additionally, the inspectors reviewed the susceptibility ranking calculation, including the basis for head temperature input, and verified appropriate plant specific information was used in the time-at-temperature model for determining RPV head susceptibility ranking.

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

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 VHPs.

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

   The bare head remote visual inspection was performed in accordance with procedure VP 03-10. The procedure used crawler mounted cameras that required 4 separate passes (90 degree view each) for each of the 66 nozzles. The entire bare metal surface was covered with these scans.

   All 66 nozzles (65 large nozzles and 1 vent nozzle) received remote mechanized UT examination from the inside surface in accordance with approved Procedures 54-ISI-100-09 (large nozzles) and 54-ISI-137-01(vent line). Procedure 54-ISI-100-09 used a blade probe and the ‘Time of Flight’ technique. This technique employed two 5 Mhz, 50 degree L (Longitudinal) transducers with scanning in the vertical direction. Procedure 54-ISI-137-01 used the open bore technique with 5.0 Mhz shear wave transducers. Two 45 degree transducers were used for scanning in the clockwise and counterclockwise directions. Two 70 degree transducers were used for scanning in the vertical up and vertical down directions. For both procedures an automated UT data acquisition and analysis system (Accusonex) was used.

   The inspectors reviewed the Framatome procedures and the inspection plan developed for the VHP inspection. The inspectors noted that the approved acceptance criteria and/or critical parameters for VHP leakage were applied in accordance with the procedures.

However, nine VHP nozzles did not receive full coverage, as described in the order (examination from 2 inches above the J-groove weld to the bottom of the nozzle). The reduced coverage was caused by nozzle configuration and limitations of the UT probe design. Actual coverage below the weld, in the non-pressure boundary portion of the nozzle, did not extend to the bottom of the nozzle. These 9 nozzles were approved by NRR via Turkey Point U3 Order EA-03-009 Relaxation Request Examination Coverage of RPV Penetration Nozzles, Letters L-2003-067 and L-2003-068. In all cases examination included 2 inches above the J-groove weld to 1 inch below the weld of the nozzle.

In addition, since the head vent nozzle did not have an interference fit, the J-groove weld was PT inspected to assess if leakage had occurred through the weld. No relevant indications were identified.

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

   All potential crack indications 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. Although indications were identified in two nozzles, they were resolved and determined not to be crack indications and non-recordable.

4. Verification that the licensee was capable of identifying the primary water stress corrosion cracking (PWSCC) phenomenon described in the bulletins.

   The licensee performed NDE examinations on all of the CRDM and vent line nozzles during the outage. The inspection techniques had been previously demonstrated capable of detecting PWSCC type manufactured cracks as well as actual cracks verified by liquid penetrant examination.

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

   The inspectors noted that no significant examples of insulation, leakage sources, debris, or dirt, impeded the examination. The licensee was able to view 100% of each of the 65 large nozzles and the reactor head vent nozzle during the visual examinations.

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 was very good and capable of detecting any debris or small boron deposits on the bare metal head. There were no obstructions to preclude a 100% visual inspection. No boron deposits were noted.

7. Determine extent of material deficiencies (associated with the concerns identified in the three bulletins) which were identified that required repair.

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

8. Determine any significant items that could impede effective examinations.

   No significant items that could impede the examination process were noted during observation of the visual or NDE examinations.

9. Determine the basis for the temperatures used in the susceptibility calculation.

   The temperatures used in the susceptibility calculation were based on evaluations and calculations using plant specific fluid temperatures and flow rates. Because of changes in flow rates after steam generator replacement, calculations resulted in a decrease in head temperature. Therefore, the susceptibility calculation used one temperature for the period before steam generator replacement and a lower temperature after steam generator replacement.

4OA6 Meetings, including Exit

.1 Exit Meeting Summary

On April 9, 2003, the resident inspectors presented the inspection results to Mr. T. Jones and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

.2 Annual Assessment Meeting Summary

On April 2, 2003, the NRC's Chief of Reactor Projects Branch 3, Project Engineer, and Resident staff assigned to the Turkey Point Nuclear Plant (TP) met with Florida Power and Light Company (FP&L) to discuss the NRC's Reactor Oversight Process (ROP) and the TP annual assessment of safety performance for the period of January 1, 2002 - December 31, 2002. The major topics addressed were: the NRC's assessment program, the results of the TP assessment, and NRC security activities. Attendees included TP site management, members of site staff, two reporters, and three members of the public.

This meeting was open to the public. The NRC's presentation material used for the discussion is available from the NRC's document system (ADAMS) as accession number ML031000148. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

E. Avella, Acting Plant General Manager
M. Cornell, Training Manager
G. Hollinger, Protection Services Manager
W. Jefferson Jr, Site Vice-President (during refueling outage)
T. Jones, Site Vice-President
M. Lacal, Operations Manager
T. Miller, Acting Maintenance Manager
M. Moore, Health Physics Supervisor
W. Parker, Licensing Manager
W. Prevatt, Work Control Manager
G. Warriner, Quality Assurance Manager
A. Zielonka, Site Engineering Manager

NRC personnel:

J. Munday, Branch Chief
C. Patterson, Senior Resident Inspector
K. Green-Bates, Resident Inspector
R. Reyes, Resident Inspector
S. Ninh, Project Engineer
S. Vias, Senior Reactor Inspector
W. Sartor, Senior Emergency Preparedness Inspector
B. Crowley, Senior Reactor Inspector
F. Jape, Senior Project Manager
R. Chou, Reactor Inspector

ITEMS OPENED AND CLOSED

Opened/Closed

50-250/03-02-01	NCV	Inadequate Corrective Action For a Failed CVCS Pipe Support (Section 1R08.1)
50-250, 251/03-02-02	NCV	Failure to report Main Steam Safety Relief Valve Test Results Outside TS Limits (Section 4OA2)
TI 2515/150 (Unit 3)		Reactor Pressure Vessel Head and Head Penetration Nozzles (NRC Bulletin 2002-02) (Section 4OA5)

Closed

50-250/2002-002-00	LER	Operation with One Component Cooling Water Pump in Excess of Technical Specification Allowable Limits (Section 4OA3)

PARTIAL LIST OF DOCUMENTS REVIEWED

Section 1R08.1: Inservice Inspection (ISI)

Procedures
NDE Manual Examination Procedure, NDE - 4.1, Visual Examination VT-1, Weld/Bolting/Bushings/Washers, Rev. 11
NDE Manual Examination Procedure, NDE - 4.3, Visual Examination VT-3, Rev. 9
External Corrosion (XCI), Monitoring Program for Insulated Piping, Rev. 0
NDE Manual Examination Procedure, NDE - 5.1, Ultrasonic Examination of Pressure Vessel Welds, Rev. 10
NDE Manual Examination Procedure, NDE - 9.3, Radiographic Examination General Requirements, Rev. 0

Other Documents
Turkey Point Nuclear Power Plant Unit 3 Inservice Inspection Plan, ISI-PTN-3-Plan, Rev. 3
Letter TPN to NRC, Examination Schedule for Remaining RCS Components, June 26, 2002
TPN Units 3 & 4, Third Inservice Inspection Interval, Relief Request Number 33, "Alternative Requirements for Implementation of Appendix VIII, Supplement 10", PTN-ENG-SOES-03-006, Rev. 0
STD-C-011, Acceptance Criteria for As-Built Safety Related Piping and Pipe Supports, Rev. 3
Condition Reports: 01-2031, 01-1984, 01-2028, 03-0125, 02-1211, 01-1842, 03-0122, 03-0486, 03-0506, 03-0452, 98-1357
PMAI PM98-10-044
NCR-91-0766
PTN-ENG-LRAM-00-0028, Rev. 3, Boric Acid Wastage Surveillance Program - License Renewal Basis Document
OP-0206.7, Containment Visual Leak Inspection
Boric Acid Corrosion Control, NP-919, Rev. 0

Section 1R08.2: Inservice Inspection (ISI)

Florida Power & Light Co. Turkey Point Units 3 and 4; Steam Generator Secondary Side Integrity Plan, dated November 2002

Florida Power & Light Co. Turkey Point Units 3 and 4 Engineering Evaluation PTN-ENG-SEMS-02-060; Degradation Assessment for the Turkey Point Unit 3 & 4 Steam Generators Update for the Turkey Point Unit 3 EOC Refueling Outage, Revision 0

Letter No. L-2002-032; Turkey Point Unit 3 Steam Generator Tube Plugging Inservice Inspection 12-Month Special report - Revision, dated February 28, 2002

Framatome ANP Shift Turnover Status reports for 3/11/03, 3/12/03 3/13/03.

0-ADM-651; FL&P Turkey Point Nuclear Plant Nuclear Chemistry Manual; July 31, 2001

EPRI TR-107569-V1R5; PWR Steam Generator Examination Guidelines; Revision 5

Turkey Point Steam Generator Eddy Current Analysis Guideline & Performance Demonstration, Revision 0

Turkey Point Unit 3 Steam Generator Examination Plan, Revision 0

2003 ET Data Analysis as compared with 2001 Data Analysis & Tube History for PTN-3 SG 10 Row 33 Column 44; October 8, 2001

2003 ET Data Analysis as compared with 2001 ET Data Analysis & Tube History for PTN-3 SG 10 Row 32 Column 15; October 9, 2001

2003 ET Data Analysis as compared with 2001 ET Data Analysis & Tube History for PTN-3 SG 20 Row 34 Column 51; October 10, 2001

2003 ET Data Analysis as compared with 2001 ET Data Analysis & Tube History for PTN-3 SG 20 Row 28 Column 41; October 10, 2001

2003 ET Data Analysis as compared with 2001 ET Data Analysis & Tube History for PTN-3 SG 20 Row 27 Column 42; October 10, 2001

2003 ET Data Analysis as compared with 2001 ET Data Analysis & Tube History for PTN-3 SG 30 Row 32 Column 64; October 9, 2001

Framatome Specification Sheet for SG3A Plug, Row 21, Col 38, Revision 0

QA Daily Observation Sheet for 3/11/03 ; Steam Generator ECT dated March 12, 2003

<u>Condition Reports</u>

02-0966	During FOSAR piece of FME identified on tube sheet of "B" SG in annulus of cold leg, dated May 13, 2002
03-0310	Damage of 3B FW Pump investigate where debris went, dated February 21, 2003
03-0488	Damage of 3B SGFP, Extensive amount of debris, dated March 4, 2003
03-0658	SG Nozzle Covers, dated March 13, 2003

03-0664 3 SG Tubes Require Plugging, dated March 13, 2003
03-0674 Row 1 Col 86 plug in 3B SG Hot Leg, dated March 13, 2003

03-0684 Evaluation of Foreign Objects remaining in SG3A, SG3B and SG3C
JPN-PTN-SEMS-96-038 DUE 5/22/03, dated March 14, 2003

Section 1R17: Permanent Plant Modifications Self Assessment Documents

Self Assessment Documents
PTN-ENG-00-0327Engineering Department Self Assessment of Maintenance Specifications & the MRA Process
Engineering Department Design Process Self assessment

Other Documents
Living FSAR
Technical Specifications
Condition Report (CR) No. 02-1794, Supplement 1, Support R-334-5 Was Missing

Calculations
PTN- 4FSC-02-2002, Rev. 0
PTN-BFSC-02-1002, Rev. 0, Assessment for Steam Generator Blowdown Discharge Line to the Existing Storm Drain System
Offsite Dose Calculation Manual for Gaseous and Liquid Effluents from the Turkey Point Units 3 & 4, Rev. 11

Drawings
5614-C-1801, Shts. 1, 1A, 1B, 2, 2A
5610-M-410-239 Rev. 0
5610-C-1244, Rev. 13, Control Building Battery Room EL. 30'-0" Masonry Wall Plan, Sections, & Details

Change Request Notice (CRNs)
CRNs for each PC/M listed
CR-01-2435
CR-01-1196
CR-02-2154 (Self Assessment of the 50.59 Process)

Procedures
0-ADM-012, Scaffold Controls, 2/6/03
0-ADM-701, Control of Plant Work Activities
0-ADM-104, 10CFR50,59 Applicability/Screening Reviews, 1/28/02
0-ADM-503, Control and Use of Temporary System Alterations, 1/13/03
ENG-QI-2.1,10.CFR-50.59 Applicability/Screening/Evaluations, Rev. 5
ENG-QI-2.1,10.CFR-50.59 Applicability/Screening/Evaluations, Rev. 3
Guidance for Performing 10CFR50.59 Evaluations. Rev. 4
Eng-QI 2.3 Operability Determinations, Rev 5
ENG-QI 4.4, Rev. 2 Procurement Classifications
ENG-QI 4.2, Rev. 12, Procurement Engineering Control
ISC TS 7.1, Rev. 7, Receiving Inspection

Temporary System Alterations (TSA)
03-02-041-07
03-02-030-12
03-02-041-04
03-02-075-08

Section 1R22: Surveillance Testing

4-OSP-072.5	Main Steam Safety Valve Setpoint Verification Test (As found March 13, 1999) (As left April 6, 1999) (As found March 21, 2002) (As left October 21, 2000)
3-OSP-072.5	Main Steam Safety Valve Setpoint Verification Test (As found February 26, 2000) (As left March 24, 2000) (As found September 9, 2001) (As left October 26, 2001)

Section 4OA5: Other Activities

Site Requirements for Reactor Vessel Head CRDM Nozzle Inspection and Repair at Turkey Point Unit 3

Framatome ANP Nondestructive Examination Procedure 54-ISI-367-03, Procedure for Visual Examination for Leakage of Reactor Head Penetrations, Revision 3

Framatome NDE 108.0, Task Lesson Plan Bare Head Inspection, Revision 0

Framatome ANP Nondestructive Examination Procedure 54-PT-6-07, Visible Solvent Removable Liquid Penetrant Examination Procedure, Revision 07

Framatome ANP Nondestructive Examination Procedure VP 03-010, Reactor Head Nozzle Penetration Remote Visual Inspection Plan for Turkey Point Unit 3, Revision 02

Framatome ANP Nondestructive Examination Procedure 54-ISI-100-09, Remote Ultrasonic Examination of Reactor Head Penetrations, Revision 02

Framatome ANP Nondestructive Examination Procedure 54-ISI-137-01, Remote Ultrasonic Examination of Reactor Vessel Head Vent Line Penetrations, Revision 01

Framatome ANP Nondestructive Examination Procedure 54-ISI-25-27, Written Practice for Personnel Qualification Visual Method, Revision 27

PTN-ENG-SESJ-01-0058, Engineering Evaluation for Response to NRC Bulletin 2001-01 for Turkey Point Units 3 and 4, Revision 0

PTN-ENG-SESJ-020041, Engineering Evaluation for Response to NRC Bulletin 2002-02 for Turkey Point Units 3 and 4, Revision 0

PTN-ENG-SEFJ-021, Engineering Evaluation Input for the Reactor Vessel Temperature Analysis, Revision 1

Westinghouse WCAP-13493, Reactor Vessel Closure Head Penetration Key Parameters Comparison

Westinghouse Letter FPL-01-131, Florida Power & Light Company Turkey Point Unit 3 Upper Head Fluid Temperature Evaluation

Resolution of Indication for Turkey Point Nozzle 23, March 2003 Outage, CDRM J-Groove UT Examination; Dated March 12, 2003.

Framatome ANP Nondestructive Examination Procedure No. 54 Change Authorization No. FRA-03-004, Revision 1 dated February 26, 2003

Framatome ANP Nondestructive Examination Procedure No. 54 Change Authorization No. FRA-03-005, Revision 1 dated February 26, 2003

Framatome Liquid Dye Penetrant Examination Report for Reactor Vessel Head Vent, dated March 16, 2003

FPL Letter No. L-2003-067; Turkey Point U3 Order EA-03-009 Relaxation Request Examination Coverage of RPV Penetration Nozzles," dated March 11, 2003

FPL Letter No. L-2003-068; "Turkey Point U3 Order EA-03-009 Relaxation Request Examination Coverage of RPV Penetration Nozzles - Supplemental," dated March 14, 2003.

Personnel Certification Records for Framatome Inspection Personnel, including:

- Personnel Training Report Release # 03-040 dated 3/5/2003
- Bare Head Inspection Training Matrix
- Individual Examiner Certification, Training, and Eye Test Records

Framatome Equipment Certification Records for the following Inspection Equipment:

- µTOMOSCAN Pulser-Receivers VH-8167 and VH7969
- UT Blade-Probes S0510CN, S0531CN, S0532CN, S0533CN, S0534CN, S0535CN, and S0536CN

Liquid Penetrant Materials - Penetrant Batches 02K01K and 02C033, Developer Batches 02C009 and 02L02K, Cleaner Batch 02H006

Reactor Head Nozzle Penetration Remote Visual Inspection Plan for Turkey Point Unit 3