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

January 26, 2004

EA 03-227

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing &
Regulatory Programs
15760 West Power Line Street
Crystal River, FL 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000302/2003006 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Young:

On December 27, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Crystal River Unit 3. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 12, 2004, with you and 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.

The report closes one issue involving a failure to meet Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.12.a requirements. Although the issue constitutes a violation of NRC requirements, we have concluded that Progress Energy - Florida Power Corporation's actions did not contribute to the degraded condition and, thus, the condition was a matter beyond your control. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing an enforcement action for the violation. Evaluations were performed and we have determined that this issue was of very low safety significance.

In addition, this report documents one inspector identified finding and one self-revealing finding, both of very low safety significance (Green). The findings were determined to involve violations of NRC requirements. However, because of the very low safety significance of the issues, and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited violations (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you wish to contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-001; with copies to the

Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Crystal River Unit 3.

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 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/ (Leonard Wert) for

Victor M. McCree, Director
Division of Reactor Projects

Docket No.: 50-302
License No.: DPR-72

Enclosure: Inspection Report 05000302/2003006
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-302
License No.:	DPR-72
Report No.:	05000302/2003006
Licensee:	Florida Power Corporation
Facility:	Crystal River Unit 3
Location:	15760 West Power Line Street Crystal River, FL 34428-6708
Dates:	September 28, 2003 - December 27, 2003
Inspectors:	S. Stewart, Senior Resident Inspector R. Reyes, Resident Inspector B. Crowley, Senior Reactor Inspector (Contractor) (4OA5.2,.3) W. Bearden, Senior Resident Inspector, Browns Ferry Unit 1 (1RO8.1) G. Kuzo, Senior Radiation Inspector (1OS1, 2OS2, 2PS2, 4OA5.4) A. Nielsen, Health Physics Inspector (1OS1, 2OS2, 2PS2, 4OA5.4) R. Chou, Engineering Specialist (1RO8.2) M. Scott, Senior Reactor Inspector (1R02, 1R17) P. Van Doorn, Senior Reactor Inspector (1R02, 1R17) J. Fuller, Reactor Inspector (1R02, 1R17) N. Staples, Reactor Inspector (1R02, 1R17)
Approved by:	Joel T. Munday, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000302/2003-006; 09/28/2003 - 12/27/2003; Crystal River Unit 3; Inservice Inspection Activities and Event Followup.

The report covered a three month period of inspection by resident inspectors and announced inspections by region based engineering and health physics inspectors. Two Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC -Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green: The inspector identified a non-cited violation of Technical Specification 5.6.1.1 for failure to follow procedural requirements involving incorrect calibration of a magnetic particle testing (MT) yoke. This finding could have inhibited the identification of indications or flaws on American Society of Mechanical Engineers (ASME) Class 2 Safety-Related Feed Water to Once Through Steam Generator (OTSG) “A” piping.

This finding is more than minor because if left uncorrected, it could result in a more significant safety concern. Failure to correctly perform the calibration could reduce the ability to discover indications or flaws which could lead to pipe breaks. The issue was determined to be of very low safety significance because the likelihood of a loss of coolant accident (LOCA) initiator was not affected, the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not have available, and the finding did not increase the likelihood of a fire or flood. (Section 1R08)

Cornerstone: Barrier Integrity

Green: A self-revealing non-cited violation of Technical Specification 3.4.12.a was identified. Small cracks in the pressurizer upper level instrument tap nozzles resulted in pressure boundary leakage since late 2000.

The finding was greater than minor because the breach in the reactor coolant system (RCS) affected the RCS barrier performance attribute of the Barrier Integrity Cornerstone objective. However, the cracks were very small, were axial in direction, and therefore, were not expected to grow large enough to challenge the structural stability of the nozzle. A Phase 3 analysis was performed and because the likelihood of a LOCA initiator was not affected, the finding was determined to be of very low safety significance. (Section 4OA3.3)

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Crystal River 3 operated at full power until October 3, when the plant was shutdown for refueling outage 13. Power operation resumed on November 5, however a reactor trip occurred that day due to a malfunctioning feedwater pump controller. Power operation resumed on November 6, but the plant was shutdown on November 7, to repair a control rod drive stator. Power operation resumed on the same day and the plant reached full power on November 8. On December 9, power was reduced to 9 percent and the main turbine was taken off-line for repair of an electro-hydraulic controls system fluid leak. The plant returned to power operation on December 10 where it remained through the end of this report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity [Reactor-R]

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee's plans for mitigating cold weather to assure that vital systems and components were protected from freezing in accordance with the licensee's Administrative Instruction (AI) -513, Seasonal Weather Preparations. The inspectors walked down portions of the emergency feedwater pump building, the intake area, and the boric acid storage tank area to check for any unidentified susceptibilities. Nuclear condition reports were reviewed to verify that the licensee was identifying and correcting cold weather protection issues. During periods when outdoor temperature fell below 40 degrees Fahrenheit (F), the inspectors verified that the licensee implemented the administrative instructions to mitigate any effects from cold weather.

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, Updated Final Safety Analysis Report (UFSAR), or procedures may be made, and tests conducted, without prior NRC approval. The inspectors reviewed evaluations for five changes and additional information, such as calculations, supporting analyses, the 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 samples of design changes, UFSAR changes, commercial grade dedication packages, a procedure change, and a Technical Specification Bases 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 10CFR50.59.

The inspectors also reviewed a recent audit of the 10CFR50.59 process and selected Action Requests forms and work orders to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated. The evaluations and documents reviewed are in the List of Documents Reviewed.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

Partial Equipment Walkdowns

a. Inspection Scope

The inspectors reviewed the alignment of the selected risk-significant systems to evaluate readiness while the redundant train was out of service for maintenance. The inspectors checked switch and valve positions using the alignments specified in the licensee's operating procedures and checked for proper electrical power alignment to critical components. The inspectors reviewed applicable sections of the Crystal River 3 UFSAR to obtain design and operating requirements. Nuclear condition reports were reviewed to verify that the licensee was identifying and correcting component alignment issues. The inspectors performed the following partial system walkdowns during this inspection period:

- 120 Volt AC Vital Distribution using Operating Procedure OP-700D, Operation Of The 120 Volt AC Vital Buses, when inverter VBIT-1A was out of service for trouble shooting on September 29, 2003

- Emergency Diesel Generator, EDG - 1A using Operating Procedure OP-707, Operation Of The ES Emergency Diesel Generators, when EDG - 1B was out of service for testing on November 19, 2003

- B Train of Reactor Building Spray using Operating Procedure OP-405, Reactor Building Spray, when A Train was out of service for repair of flow instrument DC-73-FIS using work order WO 479038-01 on December 12, 2003.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors walked down the following risk-significant plant areas to verify that control of transient combustibles and ignition sources were consistent with the licensee's Fire Protection Plan and 10 CFR Part 50, Appendix R. The inspectors also evaluated the material condition, operational lineup, and operational effectiveness of fire protection systems and assessed material condition of fire barriers used to contain fire damage. The inspections were completed using the standards of the Crystal River Fire Protection Plan; 10 CFR Part 50, Appendix R; the Florida Power Corporation Analysis of Safe Shutdown Equipment; and the UFSAR. The inspectors monitored surveillance procedure SP-800, Monthly Fire Extinguisher Inspection, to assure the operational condition of fire protection equipment. When applicable, the inspectors checked that compensatory measures for fire system problems were implemented. The inspectors observed performance of fire alarm checks done in accordance with surveillance procedure SP-323, Evacuation and Fire Alarm Demonstration.

- Boric Acid Storage Tank Area
- Reactor Building, all elevations during Refuel Outage 13
- Spent Fuel Pool Room
- Emergency Feed Pump EFP-3 Building
- Fire Pump Building
- Main Control Room
- B Decay Heat and Building Spray Vault

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

.1 Steam Generator Tube Inspections

a. Inspection Scope

The inspectors observed activities and reviewed selected inspection records for the eddy current examination (ET) of the once through steam generators (OTSGs). The records were compared to the Technical Specifications (TS), License Amendments and applicable industry established performance criteria to verify compliance. Qualification and certification records for examiners, equipment and procedures for the eddy current examination activities were reviewed. Approximately twelve examples of bobbin and rotating coil inspection ET data were reviewed to evaluate the adequacy of completed data analysis. Additionally, the inspectors reviewed three action requests (ARs) associated with OTSG examinations.

b. Findings

No findings of significance were identified.

.2 Other Inservice Inspection Activities

a. Inspection Scope

The inspectors observed in-process inservice inspection (ISI) work activities and reviewed selected ISI records. The observations and records were compared to the Technical Specifications (TS) and the applicable Code (ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition with no addenda) to verify compliance.

Portions of the following Unit 3 ISIs were observed:

Ultrasonic (UT)	-	Pipe Weld C2.1.130 (FW-328), Drawing CR3-P-SK-105.2, Rev.1, Feed Water to OTSG "A"
	-	Pipe Weld C2.1.857 (FW-95D), Drawing CR3-P-SK-105.3, Rev. 1, Feed Water to OTSG "A"
Magnetic Particle (MT)	-	Pipe Weld C2.1.130 (FW-328), Drawing CR3-P-SK-105.2, Rev.1, Feed Water to OTSG "A"
	-	Pipe Weld C2.1.857 (FW-95D), Drawing CR3-P-SK-105.3, Rev. 1, Feed Water to OTSG "A"
Visual (VT)	-	Pipe Lug Weld D2.5.19 (MSH-208), Drawing CR3-P-SKH-217.1, Rev.1, 6" Main Steam to Emergency Feed Water Pump
	-	Pipe Lug Weld D2.5.22 (MSH-212), Drawing CR3-P-SKH-217.2, Rev. 1, 6" Main Steam to Emergency Feed Water Pump

Qualification and certification records for examiners and nondestructive examination (NDE) procedures for the above ISI examination activities were reviewed. Work Orders and examination documents were reviewed.

The inspectors discussed Pressurizer Water Level Sensing and Sampling Nozzle leaks with the licensee's engineers and reviewed and observed the contingency weld repairs for an ASME Section XI code repair with a relief request to use the temper bead repair process.

The inspectors reviewed the licensee's responses to NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity. The inspectors selected samples from the pressurizer nozzle leaks identified by the licensee during this Unit 3 outage and independently observed the components to assess the significance. The inspectors also independently performed a general walkdown inside the containment to search for leaks.

The inspectors reviewed the licensee's implementation of NRC Regulatory Issue Summary (RIS) 2003-01, Examination of Dissimilar Metal Welds, Supplement 10 to Appendix VIII of Section XI of the ASME Code. The inspectors discussed the issue with the licensee's engineers.

b. Findings

Introduction: A Green non-cited violation (NCV) of Technical Specification 5.6.1.1.d was identified for failure to correctly calibrate the MT yoke prior to performing the MT on the ASME Class 2 Safety-Related pipe weld C2.1.857 (FW-95D) for the feed water line to Once Through Steam Generator A as required by the procedure.

Description: On October 15, 2003, while observing an alternating current and adjustable MT yoke calibration, the inspectors identified that the licensee examiner performed an inadequate calibration of the MT yoke. The examiner extended two legs or poles of the yoke to a distance of 10 inches between the centerlines of the legs. This resulted in a distance of approximately 9 inches between the inside edges of the two legs. Licensee procedure NDEP-0301, Rev. 14, Dry Powder Magnetic Particle Examination, Section 7.2.4, requires that alternating current magnetic yokes shall have a lifting power of at least ten pounds for a fixed pole yoke and an adjustable pole yoke with a ten inch spacing with the measurement taken between the inside edges of the yoke poles. The inspectors pointed out to the examiner that the yoke leg spacing measurement should be taken between the inside edges of the yoke legs instead of the centerlines. The examiner immediately re-calibrated the yoke with the spacing based on the procedural requirements.

The incorrect calibration of the MT yoke could affect or reduce the ability to identify flaws or indications. The licensee issued a Nuclear Condition Report (NCR) 107868 for the problem identified by the inspectors. The licensee indicated in the preliminary resolution for the NCR 107868 that this procedure was adopted from the corporate procedure and was used for the first time at Crystal River Unit 3. The equipment was re-calibrated based on the correct spacing of 10 inches between the inside edges of the two legs. The pipe weld was then examined with no indications or flaws found.

Analysis: The inspectors determined that the failure to follow the procedure was a performance deficiency. The finding was greater than minor because if left uncorrected, it could become a more significant safety concern, that being an increased likelihood of a pipe break which would affect the Initiating Events cornerstone. This incorrect non-destructive examination calibration could inhibit the identification of piping flaws in the feed water piping. The issue was evaluated using the Significance Determination Process, Phase 1 Screening Worksheet, using the Initiating Events Column and because 1) the likelihood of a primary or secondary LOCA initiator was not affected; 2) the likelihood of a reactor trip with mitigating equipment or functions not being available was not affected; or 3) the likelihood of a fire or internal/external flood was not affected, this finding was determined to be of very low safety significance (Green) and no Phase 2 evaluation was required.

Enforcement: Technical Specification 5.6.1.1.d, requires that written procedures shall be established, implemented, and maintained covering the programs in Specification 5.6.2, which includes the Inservice Inspection Program. Procedure NDEP-0301, Dry Powder Magnetic Particle Examination, specified that during calibration the required distance between magnetic poles be measured from the inside edges of the yoke legs. Contrary to above, on October 15, 2003, while calibrating the magnetic yoke, the distance between magnetic poles was incorrectly measured between the centerlines of the yoke legs. Because the failure to correctly perform the MT calibration for the MT yoke was of very low safety significance and the licensee documented this condition in Nuclear Condition Report 107868, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000302/2003006-01, Failure to Correctly Perform the Magnetic Particle Calibration. The licensee subsequently re-examined the pipe with equipment properly calibrated and determined that the welds contained no indications or flaws.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

On December 3, 2003, the inspectors observed licensed operator actions on the plant specific simulator during Licensed Operator Continuing Training Exercise SES-35, Loss of Coolant Accident (LOCA) and Anticipated Transient Without Scram (ATWS). The session involved crew use of plant procedures to safely mitigate the effects of a dropped control rod, a stuck control rod, a plant runback, a loss of pressure due to a loss of coolant accident, and a failure of the reactor to trip. The inspectors routinely monitored control room operations to check that communication skills were being used during plant operations. The inspectors specifically evaluated the following attributes related to operating crew performance in the simulator:

- Clarity and formality of communication including crew briefings
- Ability to take timely action to safely control the unit including a response to a loss of subcooling margin
- Prioritization, interpretation, and verification of alarms including a loss of decay heat removal pump alarm
- Correct use and implementation of abnormal and emergency procedures, AP-545, Plant Runback; AP-490, Reactor Coolant System Boration; AP-520, Loss of Reactor Pressure due to Reactor Coolant Leak; EOP-2, Vital Systems Status Verification; and EOP-3, Inadequate Subcooling Margin
- Control board operation and manipulation, including operator actions such as establishing decay heat removal system operation from a two reactor coolant pump operation configuration
- Oversight and direction provided by supervision, including ability to identify and implement appropriate technical specification actions
- Effectiveness of the training oversight and critique

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the planned maintenance activity listed below to evaluate the licensee's implementation of the maintenance rule (10CFR50.65). The inspectors checked that licensee personnel monitored unavailability of equipment important to safety and trended key performance parameters. For the equipment issue described in the nuclear condition report (NCR) listed below, the inspectors reviewed the licensee's implementation of the Maintenance Rule (10CFR50.65) with respect to the characterization of failures, the appropriateness of the associated a(2) classification, and the appropriateness of the associated a(2) performance criteria. The inspectors checked if the licensee maintained safety functions when equipment important to safety was out of service for maintenance. The inspectors also periodically reviewed the licensee's implementation of 10 CFR 50, Appendix B and technical specification requirements regarding safety system problems. The inspectors routinely checked that the licensee promptly entered problems with plant equipment into the corrective action program or the corrective maintenance program and evaluated if maintenance preventable functional failures or other maintenance rule issues existed that the licensee had not identified. The inspectors checked work practices and as appropriate, verified the licensee documented work problems in the corrective action program.

- NCR 113186, Instrument Air Pump, IAP-4, Tripped Twice during SP-306 Weekly Surveillance Run

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's assessments of the risk impacts of removing from service those components associated with emergent work items. The inspectors evaluated the selected high risk configuration listed below for: (1) the effectiveness of the risk assessment performed before maintenance activities were conducted; (2) the management of risk; (3) that, upon identification of an unforseen situation, necessary steps were planned to assure safety; and (4) that emergent work problems were adequately identified and resolved. The inspectors evaluated the licensee's work prioritization and risk determination to assess whether risk management actions were properly planned, controlled, and executed for the work activity discussed below.

- Spent Fuel Pump -1B Power Availability After Core Off-Load, documented in 13R High Risk Evolution Contingency Plan, dated September 2, 2003,

During power operation, the inspectors reviewed daily maintenance schedules and observed work controls to check risk management while maintenance was conducted. During Refuel outage 13, the inspectors reviewed plant configuration and work activities to check that overall risk was minimized through preservation of safety functions such as decay heat removal capability, inventory control, electric power availability, reactivity control, and containment control. On October 29, the inspectors reviewed the emergent maintenance work relating to lowering the reactor coolant system (RCS) level to129 feet, 7 inches, to replace the High Pressure Injection Thermal Sleeve following non destructive inspection. Additionally, the inspector reviewed the associated documentation, "13R High Risk Evolution Contingency Plan, dated September 28, 2003, which described the risks and contingencies imposed as a result of the equipment hatch closure time being greater than the time for the RCS temperature to increase to 200 degrees F.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions and Events

a. Inspection Scope

For the non-routine events described below, the inspectors observed portions of the activity, and reviewed operator logs and plant computer data to determine that the evolution was conducted safely and in accordance with plant procedures:

- Reactor trip response and return to power operation on November 5 and 6

- Reactor power increase after repair of control rod drive stator on November 7

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following degraded conditions to determine if operability of systems or components important to safety was consistent with technical specifications, the Final Safety Analysis Report, 10 CFR Part 50 requirements, and when applicable, NRC Generic Letter 91-18, Revision 1, Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions. The inspectors monitored licensee nuclear condition reports (NCRs), work schedules, and engineering documents to check if operability issues were being identified at an appropriate threshold and documented in the corrective action program, consistent with 10 CFR 50, Appendix B requirements, and licensee procedure CAP-NGGC-200, Corrective Action Program. When problems were not immediately repaired, the inspectors checked the technical adequacy of operability evaluations, whether operability was properly justified such that no unrecognized increase in risk occurred, and whether other existing degraded conditions were considered. The following issues, including the related nuclear condition reports (NCRs), were specifically examined:

- NCR 108372, Emergency Diesel Generator ‘3B’ Slow Start Time
- NCR 108245, Tap And Electrical Connector Fell Into Fuel Transfer Canal
- NCR 110179, Rod 2-1 Stator Temperature Rise

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated design change packages for eleven modifications, in the Barrier Integrity area, Initiating Events, and Mitigating Systems cornerstone areas, to evaluate the modifications for adverse effects on system availability, reliability, and functional capability. Attributes reviewed included; adequacy of analyses; material composition, pressure/temperature rating, code requirements satisfied; applicable testing requirements satisfied; environmental and seismic qualifications satisfied; Installation requirements, including welding, satisfied; verification of conformance to design basis; and, appropriate licensee documents updated. The modifications reviewed are identified below.

- Engineering Change (EC)51801, Repair of Pressurizer Level Taps (Barrier Integrity)
- EC 54707, Replace HPI Thermal Sleeve Downstream of MUV-43 (Barrier Integrity and Mitigating System)
- EC 50290 [EC51971, EC51988, EC51991], 13R Support / Piping Modifications to Correct Non-Conformances Identified by the LBPP FA. (Outside Containment) (Mitigating System)
- EC 50656, Snubber Reduction (Initiating Event and Mitigating System)
- EC 48755, Fuel Enhancements (Barrier Integrity and Mitigating System)
- EC 53186, RWV-24 Replacement and Piping Modification (Mitigating System)
- EC 51706, 49279, 51705 ACDP-1,-2 Panel and Breaker Replacement (Initiating Event and Mitigating System)
- EC 51906, 52011 Provide Temporary Power in Place of ACDP-1,-2 for Step-Up Transformers [MTTR-3A, 3B, 3C] Rev. 0 (Initiating Event and Mitigating System)
- EC 50696, 500KV Relay Replacement Evaluation (Initiating Event)

- EC 54637, PT-341, MUP-1B, PT-360, RWP-2A, RF-13 KW Load Testing Evaluation (Mitigating System and Initiating Event)
- EC 51055, RWP2A, PT-360 KW Testing Impact on EDG 1A Loading, Rev. 0 (Initiating Event and Mitigating System)

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

The inspectors also reviewed selected corrective action documents associated with modifications 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.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

The inspectors observed outage activities for Refuel Outage 13 conducted October 3 to November 5, 2003, to confirm that the licensee minimized risk and implemented planned strategies to mitigate potential losses of key safety functions. During the outage, outage self-assessment results and similar nuclear condition reports were checked to assure that the licensee was identifying problems at a proper threshold and providing timely corrective actions. The inspectors attended outage status meetings, toured plant areas, observed maintenance, and interviewed station personnel to assure that outage plans were being implemented as planned. Some of the specific activities checked were:

Outage Risk

Prior to the start of the refueling outage the inspectors reviewed the outage risk assessment and outage plan with the licensee. During the outage the inspectors checked that the equipment configuration as described in the risk assessment and outage plan were consistent. Contingency plans for higher risk evolutions and periods were specifically reviewed to check that adequate redundancy and safety margins were maintained. The following contingency plans were reviewed: 500kV Switchyard Work During Reduced Inventory; Spent Fuel Pump SFP-1B Power Availability after Core Off-load; Equipment Hatch Closure Time Greater than Time to 200 Degrees F.

Monitoring of Shutdown and Decay Heat Removal Activities

The inspectors monitored the plant shutdown and cooldown to Mode 5 operations, including checks that the cooldown rate was within technical specification requirements. The inspectors checked the licensee's compliance with the Mode 6 refueling technical specifications including adequate coolant boron concentration, adequate source range neutron flux instrumentation, and greater than minimum refueling canal water level. Additionally, in Mode 5, the inspectors reviewed the valve line up for the operable decay heat removal train and verified greater than minimum loop flow rate using the control room instrumentation. The inspectors performed a system walk down and verified the correct valve line up for the operable 'B' decay heat closed cycle cooling system and availability of an emergency diesel generator as a backup power source.

Vessel Closure Head Replacement

The inspectors observed the reactor vessel closure head removal and placement of the head onto a temporary stand. Prior to the head movement, the inspectors reviewed with the senior reactor operator in the reactor building the controls in place for the movement and the established communications with the control room. Once the head was lifted, the inspectors checked the licensee moved the reactor head over the approved load path. The inspectors reviewed and checked implementation of licensee procedure FP-409, Reactor Vessel Closure Head Removal, to verify the licensee maintained planned controls of the head lift. The inspectors monitored movement of the replacement reactor vessel closure head from the temporary storage building into the reactor building to assure that preplanned load paths were followed and that cleanliness and security standards were maintained.

Reduced Inventory and Midloop Conditions

During reactor coolant system draining to the midloop conditions, the inspectors checked that operations were conducted in accordance with licensee procedure OP-301A, Refueling Outage Reactor Coolant System Draining. Specific safety functions, such as availability of redundant sources of inventory and backup electric power were checked. The inspectors checked that redundant trains of decay heat removal remained available and monitored various level instrumentation for agreement and proper lineup. Flexible tygon tubing used for level indication was walked down to assure proper configuration. The inspectors checked that incore thermocouples were available, that operators had been briefed on the infrequently performed evolution, and that plans were in place to restore decay heat removal using Abnormal Procedure (AP)-404, Loss of Decay Heat Removal, should an interruption occur.

Containment

The inspectors reviewed the licensee procedures associated with closing the reactor building equipment hatch, Maintenance Procedure (MP)-114 Removal And Installation Of RB Equipment Hatch; and Compliance Procedure (CP)-341, Containment Penetration Control, to verify the licensee could implement the procedure requirements and contingency plan within the required time to reinstall the equipment hatch if there was a loss of decay heat removal. The inspectors interviewed Operations, Maintenance and Health Physics personnel in the field who had responsibility in implementing the relevant containment closure procedures. Additionally, the inspectors performed a field walk down at the equipment hatch to verify the required equipment was available for containment closure as described in the procedures and there were no obstructions to prevent timely closure.

The inspectors conducted several walkdowns of containment during the refueling outage to check cleanliness, foreign material control, radiation protection, and industrial safety. Maintenance activities were checked to assure implementation of the outage plan and risk management strategies. Prior to plant restart, a final walkdown of containment was conducted while the unit was at operating temperature and pressure to inspect for reactor coolant system leaks and debris that could affect operation of the reactor building sump.

Clearance (Tagging) Activities

The inspectors performed random checks of clearance activities during the outage to verify that activities were in accordance with licensee compliance procedure CP-15B, "Personal Danger Tags, Caution Tags, and Test Tags." During plant tours, the inspectors checked proper implementation of the licensee's foreign material exclusion program.

On October 6, 7, and 8, the inspectors reviewed and walked down the following clearances:

- 54748 Makeup/High Pressure Injection System
- 55291 Decay Heat Removal Outlet at Reactor Coolant piping, DHV-3/4/41
- 55289 Reactor Building Spray Pump 1B
- 55287 Reactor Building Spray Pump 1A
- 55278 Decay Heat Purification

On October 30, the inspectors reviewed and walked down the following clearances:

- 54748 Makeup/High Pressure Injection System
- 54751 Building Spray
- 54752 Decay Heat Suction
- 54753 Pressurizer Heaters
- 54746 Core Flood Tank

Fuel Handling and Spent Fuel Cooling Activities

The inspectors observed fuel off load activities from the reactor building and from the control room. Fuel handling controls and communications between the reactor building operators moving fuel and the reactor engineers in the control room were reviewed to determine if the licensee was following the process described in the fuel handling procedure (FP)-203, Defueling and Refueling Operations. The inspectors observed fuel reload activities in the control room, reactor building, and spent fuel pool areas to verify activities were in accordance with technical specifications and plant procedures. 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.

Reactor Coolant System Instrumentation

During the outage, the inspectors monitored system pressures and level indications for proper operation and agreement. Extra checks were made during plant draining and periods of reduced inventory to ensure adequate core cooling margins were maintained.

Electrical Power

The inspectors 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. Checks that redundant sources of electrical power were available during the reduced inventory periods were conducted.

Heatup and Startup

The inspectors 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. Checks were made that the licensee restored the plant to operating condition in accordance with operating procedure OP-202A, Refueling Outage Plant Heatup and Startup.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modifications listed below to ensure that they did not adversely affect the operation of a system that was altered. The inspectors screened temporary plant modifications for systems that were ranked high in risk for departures from design basis and for inadvertent changes that could challenge the systems to fulfill their safety function. The inspectors toured plant areas and discussed system status with engineering and operations personnel to check for the existence of temporary modifications that had not been appropriately identified and evaluated.

- Temporary Engineering Change 03-00-00-12 Patch Installed Upstream of Raw Water Valve RWV-131

- Temporary Engineering Change 03-00-00-11 Patch Installed Upstream of Raw Water Valve RWV-130

b. Findings

No findings of significance were identified.

1RST Post-Maintenance and Surveillance Testing - Pilot

This pilot inspection procedure combines both post-maintenance and surveillance testing activities.

a. Inspection Scope

The inspectors observed or reviewed the following post-maintenance and surveillance testing activities for risk significant systems to check the following (as applicable): (1) the effect of testing on the plant had been adequately addressed; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and demonstrated operational readiness; (4) test instrumentation was appropriate; (5) tests were performed as written; and (6) equipment was returned to its operational status following testing. The inspectors evaluated the licensee activities against the TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications. The inspectors routinely checked that post maintenance testing and surveillance testing issues were resolved and documented in the licensee's corrective action program.

Inservice test (IST) activities were reviewed to ensure testing methods, acceptance criteria, and corrective actions were in accordance with the ASME Code, Section XI, and Florida Power Corporation ASME Section XI, Ten Year Inservice Testing Program, dated May 4, 1998.

Post- Maintenance Testing

- WO 216627-02 — Decay Heat Valve, DHV-4, Perform Leak Test Per SP-204, Class 1 System, System Leakage Test For In-service Inspection following replacement of the seal ring

- WO 477502 — High Pressure Injection Thermal Sleeve, Perform Leak Test Per SP-204, Class 1 System Leakage Test For Inservice Inspection, Perform Preservice PT and UT Of All New Or Replaced Welds following repair

•	WO 469645-18	Perform Penetrant, Radiograph, and Ultrasonic Inspection on Pressurizer Upper Level instrument Nozzles following repair
•	WO 459614	Retest of Industrial Cooling Valve CIV-40 following repair after failed local leak rate test (Tested with SP-179C, Local Leak Rate Testing)
•	WO 496329-02	Verify no leakage upon completion of pick and clean of service water heat exchanger, SWHE-1D
•	WR125352	Run Instrument Air Pump, IAP-4, after trouble shooting for engine shutoff

<u>Surveillance Testing</u>

•	PT - 520	Installation And Functional Testing Of Audio Source Range Indicator During Refueling Operations
•	SP-354B	Monthly Functional Test Of The Emergency Diesel Generator EGDG-1B
•	SP-175	Containment Sump Level And Flood Monitoring System Calibration
•	SP-417	Refueling Interval Integrated Plant Response To An Engineered Safeguards Actuation, 'A' Train
•	SP-311	DFP-1A Diesel Fuel Transfer Pump Surveillance (IST)
•	SP-179A,B	Local Leak Rate Testing, Type B and C Leakage (Containment Valve Testing)
•	SP-340E	Decay Heat Pump, DHP-1B, Building Spray Pump, BSP-1B, And Valve Surveillance (IST)

b. <u>Findings</u>

No findings of significance were identified.

Cornerstone: Emergency Preparedness (EP)

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors checked the licensee's emergency response performance on December 3, 2003, during a licensed operator simulator evaluated session that included an Alert declaration and simulated notification of state officials. The inspectors assessed whether licensee personnel correctly classified the simulated events, then made the required state notification in accordance with the Emergency Management Procedure EM-202, Duties of the Emergency Coordinator, 10 CFR Part 50.72, and 10 CFR Part 50, Appendix E. The inspectors attended the post-scenario critique to check that the licensee evaluated performance in accordance with the Radiological Emergency Response Plan. The inspectors also reviewed the licensee evaluation of emergency classification and reporting, then assessed whether conduct of emergency operations and evaluations were in accordance with licensee procedures for each of the six evaluated operating crews.

a. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstones: Occupational Radiation Safety (OS) and Public Radiation Safety (PS)

2OS1 Access Controls To Radiologically Significant Areas

a. Inspection Scope

Access Controls. Licensee Radiation Protection (RP) program activities for monitoring workers and controlling their access to radiologically significant areas and tasks were evaluated. The inspectors assessed procedural guidance; observed and evaluated implementation of administrative and physical controls; and reviewed resultant worker exposures to radiation fields and radioactive material. Radiation worker and Health Physics Technician (HPT) proficiency in implementing RP program activities were appraised.

During the onsite inspection, implementation and effectiveness of radiological controls associated with Refueling Outage Cycle 13 (RFO 13) activities were assessed and discussed with licensee representatives. Specific outage tasks assessed in detail included lower reactor pressure vessel examinations, steam generator (S/G) maintenance and eddy current testing (ECT), in-service inspections, refueling, maintenance, and scaffolding activities; and the reactor pressure vessel head replacement project. The evaluations included, as applicable, Radiation Work Permit (RWP) details and conduct of pre-job briefings; use and placement of dosimetry to monitor occupational exposures; electronic dosimetry (ED) set-points and use in loud noise areas; use of engineering controls to minimize exposure to, and air sampling to accurately monitor potential airborne radionuclides. For the reactor pressure vessel head replacement project, the inspectors assessed RP activities associated with the old reactor head disassembly, lifting, and subsequent removal from the reactor building (RB); transfer to the onsite storage building; and reviewed details of reported occupational exposures. Physical and administrative controls and their implementation for locked-high radiation area (LHRA) and Very High Radiation Area (VHRA) entries were evaluated through field observations of selected tasks and interviews of radiation workers, HPTs, and supervisory staff. The inspectors directly observed refueling activities within the RB and the Spent Fuel Pool area, in-core detector change-out; S/G maintenance and ECT, and selected tasks within auxiliary building HRA, LHRA, and VHRA locations, and the radioactive waste (Radwaste) processing and storage facilities. For the reviewed tasks or general areas having constant or transient elevated dose rates exceeding 100 millirem per hour (mrem/hr), the effectiveness of established controls were assessed against area radiation and contamination survey results, established postings, and occupational doses received.

Occupational workers' adherence to selected RWPs and HPT proficiency in providing job and standby rescue personal coverage were evaluated through direct observations of on-going tasks, review of selected exposure records and investigations, and field interviews with licensee staff. Selected occupational exposure data associated with direct radiation, potential radioactive material intakes, and from discrete radioactive particle (DRP) or dispersed skin contamination events identified from October 1, 2002, through October 24, 2003, were reviewed and assessed independently.

Radiological postings and physical controls for access to designated HRA or LHRA locations within the auxiliary building locations, and waste processing and storage areas were examined during facility tours. In addition, the inspectors independently measured radiation dose rates and evaluated established posting and access controls for RB 95 foot Elevation S/G 'A' and "B" Platform Boundaries, the RB 95 foot Elevation High Level Trash Storage Area; and the Auxiliary Building 95 foot elevation, and Radwaste storage and processing facilities and areas.

Radiation protection program activities and their implementation were evaluated against Title 10 Code of Federal Regulations (10 CFR) 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J; Final Safety Analysis Report (FSAR) Section 11, Radioactive Waste and Radiation Protection; Improved Technical Specification (ITS) Sections 5.6.1, Procedures, Programs and Manuals, and 5.8.1, High Radiation Area; and approved licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in Section 2OS1 of the report Attachment.

<u>Problem Identification and Resolution.</u> Issues identified through department self-assessments, Nuclear Assessment Section (NAS) audits, and Corrective Action Program (CAP) documents associated with radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with responsible licensee representatives. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with Corrective Action Program

(CAP) - Nuclear Generating Group Corporate (NGGC) Standard Procedure - 006, Corrective Action Program, Rev. 9. Specific assessments, audits, and Action Request (AR) / Nuclear Condition Report (NCR) documents reviewed and evaluated in detail for this inspection area are identified in Section 2OS1 of the report Attachment.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

As Low As Reasonably Achievable (ALARA). The inspectors evaluated ALARA program guidance and its implementation for ongoing RFO 13 job tasks. The inspectors reviewed, and discussed with licensee staff, ALARA work plan (AWP) documents including dose estimates and prescribed ALARA controls for selected outage work activities expected to incur significant collective doses. The inspectors reviewed the implementation of dose-reduction initiatives for high person-rem-expenditure tasks and assessment of the effectiveness of source-term reduction efforts. These elements of the ALARA program were evaluated for consistency with the methods and practices delineated in applicable licensee procedures.

The implementation and effectiveness of ALARA planning and program initiatives during work in progress were evaluated. The inspectors made direct field or closed-circuit-video observations of work activities involving the old reactor head disassembly and lift, S/G maintenance and ECT; in-service inspections; fuel movement; maintenance; and scaffolding activities. The inspectors interviewed radiation workers and HPT staff regarding understanding of dose reduction initiatives and their current and expected final accumulated occupational doses at completion of lower pressure vessel examinations, S/G maintenance, refueling, and ECT activities. The inspectors reviewed individual and collective doses; equipment dose rates, and potential exposures to personnel within the owner controlled area for the old reactor head disassembly, lift, removal from the RB, and transfer to the onsite storage facility.

Projected dose expenditure estimates detailed in AWP documents were compared to actual dose expenditures, and noted differences were discussed with cognizant ALARA staff. Changes to dose budgets relative to changes in job scope also were discussed. The inspectors attended pre-job briefings and evaluated the communication of ALARA goals, RWP requirements, and industry lessons-learned to job crew personnel. In addition, the inspectors reviewed internal dosimetry assessments for adequacy of respiratory protection and engineering controls.

Implementation and effectiveness of selected program initiatives with respect to source-term reduction were evaluated. Shutdown chemistry program actions and cleanup initiatives, and their resultant effect on reactor building and auxiliary building area dose rates were compared to previous refueling outage data. The effectiveness of selected shielding packages installed for the current outage was assessed through reviews of selected survey records and comparison to expected planning data. Cobalt reduction initiatives for reactor building check valve replacement activities were reviewed and discussed. In addition, the inspectors reviewed design documents and discussed incorporation of ALARA initiatives affiliated with the new reactor vessel head and control rod drive mechanism (CRDM) service structure.

The plant collective exposure history for calendar year (CY) 2001 and CY 2002, based on the data reported to the NRC pursuant to 10 CFR 20.2206 (c), was reviewed and discussed with licensee staff, as were established goals for reducing collective exposure. The inspectors reviewed guidance and examined dose records of declared pregnant workers during CY 2003 to evaluate current gestation dose. The applicable RP procedure was reviewed to assess licensee controls for declared pregnant workers.

RP program activities and their implementation were evaluated against 10 CFR 19.12; 10 CFR Part 20, Subparts B, C, F, G, H, and J; and approved licensee procedures. In addition, licensee performance was evaluated against Regulatory Guide (RG) 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable, and RG 8.13, Instruction Concerning Prenatal Radiation Exposure. Procedures and records reviewed within this inspection area are listed in Sections 2OS1 and 2OS2 of the report Attachment.

Problem Identification and Resolution. Licensee CAP documents associated with ALARA activities were reviewed and assessed. The inspectors evaluated the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with CAP-NGGC-006, Corrective Action Program, Rev. 9. Specific assessments, audits, and AR / NCR documents reviewed and evaluated in detail for this inspection area are identified in Section 2OS2 of the report Attachment.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization. During the weeks of October 6 and October 20, 2003, the inspectors evaluated licensee methods for processing and characterizing radwaste materials. Inspection activities included direct observation of processing equipment for solid and liquid radwaste and evaluation of waste stream characterization data.

Solid and liquid radwaste equipment was inspected for material condition and for configuration compliance with the Final Safety Analysis Report (FSAR) and Process Control Program (PCP). Inspected equipment included spent resin storage tanks; resin transfer piping; resin and filter packaging components; and abandoned waste evaporators. The inspectors discussed system changes, component function, and equipment operability with licensee staff. In addition, procedural guidance for resin transfer was evaluated and compared to PCP requirements.

Licensee radionuclide characterizations for each major waste stream were reviewed and discussed with radwaste staff and count room technicians. For Spent Fuel Pool Filters, RCS Filters, Primary Resin, and Dry Active Waste (DAW) the inspectors evaluated analyses for hard-to-detect nuclides and appropriate use of scaling factors. Comparison results between licensee waste stream characterization data and outside laboratory data were reviewed for the period 2002-2003. For selected shipment records, waste classification calculations were independently performed and the methodology used for waste stream mixing and concentration averaging was evaluated. The inspectors also reviewed the licensee's Failed Fuel Action Plan that accounts for changing operational parameters.

Radwaste processing activities were reviewed for compliance with 10 CFR Part 50.59 and consistency with the licensee 's PCP, Rev. 5, and FSAR, Rev. 26.2, Chapter 11. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 61.55 and guidance provided in the Branch Technical Position (BTP) on Waste Classification and Waste Form. Reviewed guidance documents and records are listed in Section 2PS2 of the report Attachment.

<u>Transportation.</u> During the weeks of October 6, and October 20, 2003, the inspectors evaluated the licensee's activities related to transportation of radioactive material. The evaluation included direct observation of shipment preparation activities and review of shipping related documents.

The inspectors directly observed the preparation of a DAW shipment and interviewed technicians regarding dose limits, vehicle placarding, and other shipping regulations. The inspectors also performed independent dose rate measurements of the shipping package and compared the results to Department of Transportation (DOT) limits.

Seven shipping records were reviewed for consistency with licensee procedures and compliance with NRC and DOT regulations. The licensee's procedure for opening and closing their Type B shipping cask was compared to recommended vendor protocols and Certificate of Compliance (CoC) requirements. In addition, training records for all individuals currently qualified to ship radioactive material were checked for completeness. The inspector reviewed and evaluated the adequacy of the computer-based training curriculum provided to these workers.

Transportation program implementation was reviewed against regulations detailed in 10 CFR Parts 20 and 71, 49 CFR Parts 170-189; as well as the guidance provided in NUREG-1608. Training activities were assessed against 49 CFR Part 172 Subpart H. Documents reviewed during the inspection are listed in Section 2PS2 of the report Attachment.

Problem Identification and Resolution. Selected NCRs associated with radwaste processing and transportation were reviewed. Three NCRs, one NAS report, and one Self-Assessment were reviewed in detail and discussed with HP supervision. The inspectors assessed the licensee's ability to characterize, prioritize, and resolve the identified issues in accordance with licensee procedure CAP-NGGC-0200, Corrective Action Program, Rev. 9.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Reactor Safety

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the period of October 2002 through September 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, " Regulatory Assessment Performance Indicator Guideline, " Rev. 2, were used to verify the basis in reporting for each data element.

- High Pressure Injection System Unavailability
- Safety System Functional Failures

The inspectors reviewed operator logs, daily plant status reports, nuclear condition reports, licensee event reports, and performance indicator data sheets to verify that the licensee had adequately identified the submitted performance indicator data. The inspectors interviewed licensee personnel associated with the performance indicator data collection, evaluation, and distribution. During plant tours, the inspectors periodically checked that high radiation areas were secured as specified in the licensee's procedures and that there were no unmonitored release paths.

b. Findings

No findings of significance were identified.

.2 Radiation Safety

a. Inspection Scope

The inspectors sampled licensee data for the PIs listed below for the period from August 1, 2002, through August 31, 2003. 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 report for each data element.

Occupational Radiation Safety Cornerstone

- Technical Specification (TS) High Radiation Area (> 1 Rem/hour) Occurrences
- Very High Radiation Occurrences
- Unintended Exposure Occurrences

The inspectors reviewed Occupational Radiation Safety PI data collected from August 1, 2002, through August 31, 2003 for the Occupational Radiation Safety Cornerstone. For the reviewed period, the inspectors assessed CAP records to determine whether HRA, VHRA, or unplanned exposures, resulting in TS or 10 CFR 20 non-conformances, had occurred during the review period. In addition, the inspectors reviewed selected personnel contamination event data, internal dose assessments, and individual Radiation Control Area (RCA) exit transaction dose records exceeding 100 mrem against RWP dose limits. Reviewed documents relative to this PI are listed in Sections 2OS1, 2OS2, and 4OA1 of the report Attachment.

Public Radiation Safety Cornerstone

- Radiological Effluent Technical Specification (RETS) / Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences PI

The inspectors reviewed the RETS/ODCM Radiological Effluent Release Occurrences PI results for the Public Radiation Safety Cornerstone from August 1, 2002, through August 31, 2003. For the review period, the inspectors reviewed data reported to the NRC, procedural guidance for reporting PI information, and four NCRs documented in Section 4OA1 of the report Attachment. In addition, the inspectors reviewed monthly PI reports, out-of-service effluent monitor logs, and selected effluent release permits.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

Routine Problem Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems", and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed screenings of all items entered into the licensee's corrective action program. This review was accomplished by attending the daily management meeting where issues were discussed and by accessing and reviewing the licensee's computerized database.

The inspectors selected the following nuclear condition reports (NCR) for detailed review and discussion with the licensee. These reports were examined to verify whether problem identification was complete and timely; that safety concerns were properly classified and prioritized for resolution; and that technical issues were evaluated and dispositioned to address operability and reportability. As applicable, root cause or apparent cause determinations were checked for thoroughness; extent of condition, generic implications, common causes, and previous history were adequately considered; and that appropriate corrective actions were implemented or planned in a manner consistent with safety and technical specification compliance. The inspectors evaluated the issues using the requirements of the licensee’s corrective action program in Administrative Procedures CAP-NGGC-0200, “Corrective Action Program” and 10 CFR 50, Appendix B.

- Nuclear Condition Report 108070, Shoehorn Rope Dropped onto Core Support Plate During Refueling

- Nuclear Condition Report 108482, Trend Identified in the Area of Foreign Materials Control During Refueling Outage 13

b. Findings and Observations

There were no significant licensee performance issues or NRC violations identified by the inspectors regarding these condition reports. The inspectors verified that the apparent cause evaluation and initial corrective actions were appropriate in relation to the safety significance of the problems. Long term corrective actions were appropriately planned.

4OA3 Event Followup

.1 Automatic Reactor Trip

a. Inspection Scope

On November 5, 2003, the Unit 3 reactor automatically tripped from 37% power due to high reactor coolant system pressure caused by low Once Through Steam Generator water level. The problem was initiated by a malfunctioning feedwater pump control circuit. The inspectors responded to the control room and verified the unit was stable in Mode 3, and confirmed that all safety-related mitigating systems had operated properly. The inspectors examined operator and plant response by reviewing plant parameters, strip charts, operator logs, and discussing the event with operations personnel and members of the licensee’s Event Review Team. The inspectors verified that appropriate notifications were made in accordance with 10 CFR 50.72. Furthermore, the inspectors reviewed the post-trip report and attended the Plant Nuclear Safety Committee meeting for restart.

b. Findings

No finding of significance were identified

.2 (Closed) Licensee Event Report (LER) 05000302/2001-003-00, Small Pressure Boundary Leakage Found in Reactor Coolant Pump Heat Exchanger

a. Inspection Scope:

On October 4, 2001, while Crystal River Unit 3 was in Mode 5, Cold Shutdown, for a refueling outage the licensee identified a small tube leak in the drilled-hole heat exchanger on reactor coolant pump RCP-1B while investigating the source of elevated radionuclide activity in the nuclear services closed cycle cooling system (SW). (The drilled hole heat exchanger is part of the reactor coolant system pressure boundary.) The root cause of the leak and failure mechanism was unknown although a very small crack due to stress corrosion cracking was suspected. The leak was determined to be less than 0.02 gpm and was repaired by plugging the heat exchanger, thus removing it from service. Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.12.a requires that reactor coolant system leakage shall be limited to “No Pressure Boundary Leakage” when in Modes 1, 2, 3 and 4. Because this leak was a pressure boundary leak that occurred during operation, it constituted a violation of this TS requirement. The inspectors reviewed the LER and nuclear condition report 43024 and determined that the violation was not associated with a performance deficiency. The inspectors concluded that the licensee could not have reasonably determined that pressure boundary leakage existed while the plant was operating and could not have reasonably been expected to identify the source as a crack in the reactor coolant pump drilled-hole heat exchanger. However, a significance determination process was used to determine the risk significance level of this issue. Because the issue involved the RCS barrier, a Phase 2 analysis was required. However, Phase 2 worksheets were not applied to this finding and Phase 3 analysis was required in accordance MC 0609. A Phase 3 analysis was performed and because the likelihood of a LOCA initiator was not affected, the finding was determined to be of very low safety significance. The inspectors determined that the cracks were very small and were not expected to grow large enough to challenge the structural integrity of RCP cover thermal barrier. Therefore, the NRC has decided to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy. This LER is closed.

3. (Closed) LER 05000302/2003-003-00, Reactor Coolant System Pressure Boundary Leakage Limit Exceeded Due to Pressurizer Instrument Tap Nozzle Cracks.

a. Inspection Scope

The inspectors reviewed the LER and licensee’s program for leak detection to determine if current leak detection practices should have identified the pressure boundary leakage experienced on the unit during plant operation. The review included:

- descriptions of the leak detection practices and requirements provided in the TS

- recorded leakage data
- discussions with cognizant plant personnel

b. Findings

Introduction: A Green self-revealing non-cited violation of TS 3.4.12.a was identified for having a small pressure boundary leak in the pressurizer upper level instrument tap nozzles since late 2000.

Description: In late 2000, an increase in Reactor Building (RB) airborne activity was noted when pressurizer spray was initiated. During Refueling Outage 12 in October 2001, the pressurizer spray valve was repacked and the insulated pressurizer spray line received a focused walk down, in addition to the routine boric acid and reactor coolant system (RCS) leakage walk downs. However, after return to power operations, the RB airborne activity still increased when pressurizer spray was initiated. The licensee formed a team to address this issue and concluded that the most likely source of the leak was the pressurizer steam space. During the next refueling outage on October 4, 2003, while performing a visual inspection of the three pressurizer upper level instrument tap nozzles, evidence of small RCS pressure boundary leaks were found. The most likely failure mechanism was primary water stress corrosion cracking. The three pressurizer upper level instrument tap nozzles were repaired and other pressurizer nozzles were inspected. No further evidence of leakage was identified.

Analysis: The inspectors determined that although the licensee's corrective actions completed in 2001 seemed reasonable given the indication of leakage at the time the problem reoccurred when the unit was restarted. During the refuel outage of 2003 the licensee performed a more comprehensive inspection using more sophisticated leak detection equipment and identified the pressure boundary leak to be in the pressurizer upper level instrument tap nozzles. The finding was greater than minor because a breach in the RCS barrier affected the RCS barrier performance attribute of the Barrier Integrity Cornerstone objective. Because the finding involved the RCS barrier, a Phase 2 analysis was required. However, Phase 2 worksheets were not applied to this finding and Phase 3 analysis was required in accordance MC 0609. The inspectors determined that the cracks were very small, were axial in direction, and therefore, were not expected to grow large enough to challenge the structural stability of the nozzles. Because the likelihood of a Loss of Coolant Accident (LOCA) initiator was not affected the finding was determined to be of very low safety significance (Green).

Enforcement: Technical Specification Limiting Condition for Operation (LCO) 3.4.12 requires that reactor coolant system leakage shall be limited to "No Pressure Boundary Leakage" when in Modes 1, 2, 3 and 4. Contrary to this requirement, on October 21, 2003, the licensee identified a small pressure boundary leak in the pressurizer upper level instrument tap nozzles which had existed since 2000. The cracks could have been identified in 2001. Because the failure to identify and correct a small pressure boundary leak was of very low safety significance and had been entered into the licensee's corrective action program as NCR 106443, this violation is being treated as an NCV-

Cited Violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000302/2003006-02, Failure to Identify and Correct a Small Pressure Boundary Leak in The Pressurizer Upper Level Instrument Tap Nozzles. The LER is closed.

4OA5 Other Activities

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

a. Inspection Scope

The inspectors reviewed the Crystal River 3 visual inspection program for reactor vessel head penetrations as discussed in the licensee's response to NRC Bulletin 2003-02. The inspectors observed the licensee conduct a portion of the examination and discussed the examination with the licensee's technical staff. The inspectors independently examined a number of lower head penetrations to assess the material condition of the lower reactor vessel area. The inspection guidelines were provided in TI 2515/152, dated September 5, 2003.

b. Findings

There were no findings of significance. The results of the inspection are described:

a.1) Verification that visual examination was performed by qualified and knowledgeable personnel:

The inspectors verified that Crystal River Unit 3 examination personnel had received specialized training on the visual expectations for leakage from lower reactor penetrations and on the site specific procedures to be used for the examinations. The inspectors interviewed the examination personnel and noted that they were knowledgeable of the specialized qualification criteria. The inspector verified that both examination personnel were certified as Level II or III, VT-2 examiners, and had completed a recent visual acuity evaluation.

a.2) Verification that visual examination was performed in accordance with demonstrated procedures:

The inspectors verified that the examination was done in accordance with Progress Energy Procedure, NDEP-0612, VT-2 Visual Examination of Nuclear Power Plant Components. The inspectors observed that the examination was documented in Visual Examination for Boric Acid Detection, Report Number VT-03-048. The inspectors verified by direct observation and in discussions with examination personnel that the approved acceptance criteria and/or critical parameters for penetration leakage were applied in accordance with the procedures.

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

The inspectors verified that the licensee's inspection plan provided penetration indexing and drawings with adequate guidance to ensure that the visual examinations included 100% circumferential coverage of each penetration. The inspectors verified that the examination result for each penetration was individually documented. No evidence of boric acid or other reactor coolant system leakage was identified.

a.4) Verification that the licensee was capable of identifying the pressure boundary leakage as described in the bulletin and/or reactor pressure vessel lower head corrosion:

The inspectors visually observed about 40 percent of the lower vessel head during the licensee's examination; observed the licensee conduct a portion of the examination; discussed the examination with the licensee examiners prior to, during, and following the examination; reviewed a video of the examination, checked examination documentation, and verified the qualification of the licensee examination personnel. The inspectors concluded that the licensee conducted an effective visual inspection to identify potential leakage resulting from lower vessel penetrations. No evidence of boric acid or other reactor coolant system leakage was identified.

b) Evaluate condition of the reactor pressure vessel lower head (debris, insulation, dirt, boric acid deposits from other sources, physical layout, viewing obstructions). Did it appear that there were any boric acid deposits at the interface between the vessel and the penetrations?

The inspectors observed no significant examples of insulation, leakage sources, debris, dirt, or other physical impediments that prevented a thorough visual examination. The licensee was able to adequately view each of the 52 penetrations, bare metal, during the examinations. Hand held lighting provided adequate illumination for the examinations. No evidence of boric acid or other reactor coolant system leakage was identified. The inspectors noted that the lower head had been coated with a silver paint, that the paint had peeled in most areas, and that the lower head had rusted. The licensee documented the condition in the corrective action program (NCR 107571) and completed a pressure washing of the area following the visual examination to remove debris and establish a baseline condition for future examinations.

c) How was the visual examination conducted and how complete was the coverage?

The licensee's examination was done by direct visual check of each penetration using hand held illumination. Each penetration was examined through the 360 degree circumference of each penetration through the lower vessel and the overall lower vessel was checked for evidence of leakage from any source. A hand held camera was used to provide a backup to the direct inspection.

d) Could small boric acid deposits representing reactor coolant system leakage, as described in the Bulletin 2003-02, be identified and characterized, if present?

The licensee was able to adequately observe and characterize each penetration by direct visual examination to assure that no leakage was indicated. The inspectors noted no significant items that could impede the visual examination process.

e) What material deficiencies were identified that required repair?

No material deficiencies were found that required repair. The inspectors noted that the lower head had been coated with a silver paint, that the paint had peeled in most areas, and that the lower head had rusted. The licensee completed a pressure washing of the area following the visual examination to remove debris and establish a baseline condition for future examinations.

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

No inspection impediments were identified. Lower head insulation was removed for the examination and radiation levels were not prohibitive. Each penetration was examined by direct visual technique.

g) Did the licensee perform appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components above the reactor pressure vessel lower head?

Not applicable, no boric acid residue was identified on the lower head.

h) Did the licensee take any chemical samples of the deposits? What type of analysis was performed? What were the licensee's criteria for determining that boric acid deposits were not from reactor coolant system leakage?

Not applicable, no boric acid residue was identified on the lower head.

i) Is the licensee planning to do any cleaning of the lower head?

The licensee did a pressure wash of the lower head to establish a baseline condition for any future examinations.

j) What are the licensee's conclusions regarding the origin of any deposits present and what is the licensee's rationale for the conclusions? Does the inspector believe the licensee's conclusions are reasonable?

Not applicable, no boric acid residue or other deposits were identified on the lower head.

.2 Reactor Vessel Closure Head (RVCH) Replacement

a. Inspection Scope

The inspectors observed/reviewed the activities detailed below for the replacement RVCH to verify compliance with applicable Codes (ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition with no Addenda and Section III 1989 Edition with no Addenda) as defined in Engineering Change (EC) 50220R2, RVCH [Reactor Vessel Closure Head] and Service Structure Replacement.

Observation and Inspection of RVCH

The inspectors observed the RVCH after installation of the new Control Rod Drive Mechanisms (CRDM) and rotation of the head 90 degrees while on the installation platform ready for movement to the containment. In addition, at the “W” axis, the top of the head, nozzle penetrations and CRDM Flanges were inspected through one of the inspection ports and a cooling fan opening. Specifically, CRDM flange to nozzle welds and nozzle to head interface areas were visually examined for penetration numbers 44, 45, 61, 68, and 69.

RVCH and CRDM Fabrication Welds

The inspectors visually reviewed the following RVCH fabricator (Framatome ANP) shop welds:

- Production Weld Data Sheets for CRDM housing (nozzle to CRDM flange) welds (Weld S/C001)
- Radiographic (RT) film for nozzle to CRDM housing welds (Weld S/C001) CRDHC 26.26, 26.27, 26.28, 26.29, 30.35, 38.42, and 46.48
- Liquid Penetrant Report for inside and outside of CRDM housing welds (Weld S/C001)
- Production Weld Data Sheets for nozzle to RVCH J-Groove butter welds (Weld B/D001)
- Production Weld Data Sheets, Final Liquid Penetrant Examination Report, and PT-white Liquid Penetrant Examination Report for nozzle to RVCH J-Groove welds (Weld S/P001)
- Production Weld Data Sheets, Liquid Penetrant Examination Report, and Ultrasonic Examination Report for RVCH clad welds (Welds R/D001, R/D002, R/D006, and R/D007)

- Certified Material Test Reports for the following heats/lots of welding material used for the above welds: 3/32" Inconel 152 - Lot WC72F1, 1/8" Inconel 152 - Lot WC72FO, 5/32" Inconel 152 - Lot WC77E3, 5/32" Inconel 152 - Lot WC61F8, 0.035" Inconel 52 - Heat NX1576JK, 0.035" Inconel 52 - Heat NX3167 JK, 309L wire/flux combination - Lots 36519-204989, and 308L wire/flux combination 36140-00828

- Heat Treatment Report, including time-temperature strip chart for final stress relief of the RVCH

- RVCH Hydrotest Report CC/CR 001, including Examination Report

- Certified Material Test Report, including the Japan Steel Works Quality Plan for the RVCH forging (Heat 02W10-1-1), Heat Analysis, Record of Ultrasonic Examination, Mechanical Test Results, and Record of Magnetic Particle Inspection

Preservice Inspection (PSI) and Baseline Inspections

Relative to Preservice Inspection of the replacement RVCH, the inspectors discussed with licensee personnel the completed inspections as well as planned inspections, which consisted of: (1) fabrication Non-Destructive Examination (NDE) inspections for the Category BO CRDM Motor Tube welds, (2) recently completed (by Crystal River NDE examination personnel) liquid penetrant (PT) examination of peripheral Category BO CRDM flange to nozzle welds (3) visual (VT) examination of Category B-G-2 CRDM bolting, and (4) VT examination of the Bolted Mechanical Connection between the RVCH and the Service Support Structure, and (4) VT examination for leakage of Category BE welds (J-Groove partial penetration welds) during hydrostatic testing after fabrication and VT inservice leakage examination during power ascension after installation of the head. The inspectors reviewed: (1) VT Examination Reports VT-03-001 and VT-03-028 for examination of the CRDM bolting, (2) PT Examination Reports (including certification records for penetrant materials) for CRDM flange to nozzle welds CC/CR001-CRDH-46, 47, 48 and 69, (3) a vendor certification of PT results and a sample of the Code Data Reports for fabrication of the CRDM Motor Tube welds, (4) VT examination report for the Bolted Connection between the RVCH and the Service Support Structure, and (5) RVCH Hydrotest Report CC/CR 001, including Examination Report.

The licensee discussed with the inspectors the scope of baseline inspections of the CRDM nozzle and J-Groove weld area, which included: bare metal VT of the top of the head, ultrasonic examination (UT) of the nozzle volume from a minimum of 2 inches above the J-Groove weld to the bottom of the nozzle, and PT-white of the surface of the J-Groove welds (including 2 inches on both sides of the weld). The inspectors reviewed the completed Examination Summary Report for these UT and VT examinations, including review of the UT inspection data for penetrations 1, 7, 13, 18, 28, 45, and 67. In addition, the examination report for the PT-white examination of the J-Groove welds was reviewed as part of the review of fabrication records detailed above.

b. Findings

No findings of significance were identified.

.3 Review of 10 CFR 50.59 Evaluations for the Replacement RVCH

a. Inspection Scope

The inspectors reviewed Engineering Change 50220R2, RVCH and Service Structure Replacement, including the associated 10 CFR 50.59 evaluation to verify that changes between the original RVCH and the replacement RVCH, and modifications resulting from installation of the replacement RVCH were properly evaluated in accordance with 10 CFR 50.59.

b. Findings

No findings of significance were identified.

.4 Reactor Vessel Head Replacement Radiation Protection Inspection

a. Inspection Scope

The inspectors reviewed and evaluated radiation protection planning and preparation, established radiological controls and their implementation for the reactor pressure vessel head replacement activities. Specifically, the inspectors reviewed and evaluated ALARA planning; dose estimates and dose tracking; exposure controls including temporary shielding, contamination and airborne radioactivity controls, radioactive material management; AWP documents, emergency contingencies; and project staffing and training plans.

ALARA work plan details for the reactor head replacements were reviewed. The radiation, contamination, and airborne radioactivity surveys in the packages were reviewed for radiological work conditions and the adequacy of prescribed postings and surveys. The inspectors reviewed RWPs to determine projected exposure, expected conditions, electronic dosimeter dose and dose rate alarm settings, dosimetry requirements, protective clothing/equipment, worker instructions and HPT instructions. The ALARA exposure estimates were reviewed and evaluated against changing work scope/radiological conditions. In addition, the inspectors reviewed internal dosimetry assessments for adequacy of respiratory protection and engineering controls. Corrective action documentation was reviewed for significant trends or recurring problems with work practices and controls. The expected and measured source term and ALARA initiatives implemented during the tasks evolutions were discussed.

The inspectors discussed contingency plans associated with movement of the reactor head from the RB to the established on-site storage location. In addition, dose rate data and resultant cumulative dose to workers within the owner control area were reviewed and discussed in detail. The inspectors reviewed radiological controls, established monitoring programs, and toured and observed independent surveys of the storage facility environs.

Licensee program activities and their implementation were evaluated against 10 CFR 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J; and approved licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in Sections 2OS1 and 2OS2 of the report Attachment.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The resident inspectors presented the inspection results to Mr. Young and other members of licensee management at the conclusion of the inspection on January 12, 2003. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. The licensee did not identify any proprietary information.

The inspectors presented the inspection results regarding the Reactor Vessel Closure head Replacement inspection to Mr. Young, and other members of the licensee's staff on October 9, 2003. Proprietary documents were reviewed during the inspections, but proprietary information is not included in this report. (4OA5.3)

A preliminary exit was conducted on October 17, 2003, to discuss the details of this regional based inspection of the inservice inspection program. The licensee did not identify any proprietary information. (1R08, 1R17)

On October 24, 2003, results of the onsite radiation protection inspection were discussed with Mr. D. Young, Site Vice President. The inspectors noted that proprietary information was reviewed during the course of the inspection but would not be included in the documented report. The inspectors discussed the areas reviewed and noted that findings associated with an October 4, 2003, RB LHRA control event would be considered unresolved pending review of additional let-down heat exchanger dose rate data during RFO 13 shutdown activities. In addition, the significance and disposition of a finding associated with failure to document and properly mark a container of low level waste shipped to a licensed burial site with the proper classification would be determined following discussions with NRC Headquarters staff.

A second telephone exit was held on December 18, 2003, regarding the October 4, 2003, LHRA finding and failure to document and classify waste. The inspectors noted that from further review of data provided and discussions with NRC NRR staff, the issues would be considered minor.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
R. Lemberger, Project Engineer
H. Oates, Superintendent Major Projects
J. Huegel, Manager, Operations
S. Bernhoft, Supervisor, System Engineering
W. Brewer, Manager, Outage
R. Davis, Manager, Training
J. Franke, Plant General Manager
J. Kreuhm, Manager, Maintenance
D. Roderick, Director Site Operations
S. Glenn, Supervisor, Corrective Actions Program
S. Powell, Supervisor, Licensing
M. Rigsby, Radiation Protection Manager
J. Stephenson, Supervisor, Emergency Preparedness
M. Annacone, Manager, Engineering
J. Terry, Manager, Special Projects
R. Warden, Manager, Nuclear Assessment
D. Young, Vice President, Crystal River Nuclear Plant
S. Young, Security Manager

NRC personnel:
J. Munday. Chief, Reactor Projects Branch 3, NRC Region II
J. Riveria-Ortiz, NRC Intern
C. Casto, Director, Division of Reactor Safety, NRC Region II
Espiranza Espana, NRC Visitor
Ramon Cortez, NRC Intern
L. Reyes, NRC Region II Administrator

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000302/2003006-01	NCV	Failure to Correctly Perform the Magnetic Particle Calibration (Section 1R08)
05000302/2003006-02	NCV	Failure to Identify and Correct a Small Pressure Boundary Leak in The Pressurizer Upper Level Instrument Tap Nozzles (Section 4OA3.3)

Closed

05000302/2001-003-00	LER	Small Pressure Boundary Leakage Found in Reactor Coolant Pump Heat Exchanger (Section 4OA3.2)
05000302/2003-003-00	LER	Reactor Coolant System Pressure Boundary Leakage Limit Exceeded Due to Pressurizer Instrument Tap Nozzle Cracks (Section 4OA3.3)
2515/152 (Docket 50-302)	TI	Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02) (Section 4OA5.1)

Discussed

None

LIST OF DOCUMENTS REVIEWED

1R02: Evaluations of Changes, Tests or Experiments and 1R17: Permanent Plant Modifications

Procedure Changes and Letters
02-0070, Procedure change for alternate flow path for RCS makeup, Rev. 3
03-0546, Technical Specification Basis change for reload core design and safety analysis, Rev. 2
02-0188, CP-140 Evolution Order for Isolating CI RB Cavity Cooling, in support of replacing SWV-151, Rev. 0
03-0002, LAR 270 EC 49696, Document Procedures for Power Level Upgrade at CR3, Rev. 0
03-0588, MP-147, AHV-1A Control Circuit,
02-0090, Procedure change for alternate method to perform isolation of AHHE-10A/10B, Rev. 0
02-0151, Procedure change to reflect alternate means of determining RCP seal leakage, Rev. 0
02-0239, Technical Specification Basis change to remove containment isolation valve indications, Rev.0
02-0275, Procedure change to add verification of valve sealing mechanisms, Rev. 0
03-0674, EC53186, RWV-24 Replacement and Piping Modification, Rev. 1
03-0200, SP-340B, Rev. 0
02-0523, EOP-04, Rev.8, Rev. 0 [screen revision]
02-0097, CP-140, Operations Evolution Order 02-02-05, Rev. 0
02-0397, HPP-219, Rev. 2, Rev.0 [screen revision]
02-0069, MAR 01-09-03-01 Setpoints for CIP-3A and 3B Suction Pressure Alarms
02-0057, Work Order [WO] 371038, EFP-3 pump Bearing Replacement Instructions
02-0081, EC48494, Install Carbon Steel Anodes in the Inlet of SWHE-1B, SWHE-1C, SWHE-1D, Rev. 0
02-0083, NCR 55714, RW Leak Identified at Downstream Weld of RWV-131, Rev. 0
02-0498, FSAR change for required volumes in the BAST and BWST, Rev. 0
03-0153, EC 51706, 49279, 51705 ACDP-1,-2 Panel and Breaker Replacement, Rev. 1
03-0165, EC 51906, 52011 Provide Temporary Power in Place of ACDP-1,-2 for Step-Up Transformers

02-0516, EC 50696, 500KV Relay Replacement Evaluation, Rev. 2
02-0582, EC 51055, RWP2A, PT-360 KW Testing Impact on EGDG1A Loading, Rev. 0

Crystal River Unit 3 - Issuance of Amend Regarding Technical Specification Change Request for Use of M5 Advanced Alloy Fuel Cladding (TAC NO. MB6590), dated October 1, 2003 [NRC to Site Vice President]

Interoffice Correspondence, SE03-0090, Low Flush Water Flow to RWP-3A during SP-344A, dated 10/16/03

Crystal River Unit 3 - Issuance of Amendment Regarding Technical Specification Change Request for Departure from Nucleate Boiling Correlation (TAC NO. MB7035), date October 16, 2003 [NRC to Site Vice President]

Safety Evaluation of Framatome Technologies Topical Report BAW-10164P Revision 4, "RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analyses" (TAC NOS. MA8465 and MA8468) NRC to Framatome ANP, Richland, Inc. - Mr. Mallay, Dated April 9, 2002

File No. NF-405.1607, Serial No. NF-03A-0121 Approval of DRs 6297, 6299, and 6306 and Crs 10596, 10632, and 10720 [Progress Energy to Framatome ANP, Inc., dated August 6, 2003]

Procedures
EGR-NGGC-0005, Engineering Change, Rev. 19
MP-122, Disassembly and Re-assembly of Flanged Connections, Rev. 31
Vendor Manual 2303, Vol. 1A for EFP-3 Pump
SP-344A, RWP-2A, SWP-1A and Valve Surveillance, Rev. 44 [WO 453267 completed, 9-18-03]
MCP-NGGC-0401, Material Acquisition, Rev. 15
CP-213, Administrative Procedure for 10CFR50.59 Reviews, Rev. 10

Self Assessment Documents
CV-ES-03-01, Engineering Section Assessment, dated 3/24/2003
Assessment Number 71230, Project Management of Reactor Vessel Head Replacement Project, dated 10/11/02
Assessment Number 51351, Effectiveness of Project/Modification Selection Process, dated 2/11/02
Assessment Number 58939, REG-NGGC 10CFR50.59 Reviews and Administrative Processes, CP-213, dated 1/24/03

Action Requests, AR and Work Orders
AR 85798, NAS Engineering Assessment C-ES-03-01 findings
AR 71174, Engineering Assessment findings
AR 88795, SAST 58939, Screen Justifications
AR 58939, Self Assessment for 10CRF50.59 Implementation
WO 462318-01, RWP-2A Surveillance
AR 107787, Bearing Lubrication [flush water flow]
AR 00076622, NCR, Automatic Reactor Trip Occurred at 1439 11/07/02

Drawings
FD-302-611, Sheet 1 of 4, Nuclear Services and Decay Heat Sea Water, Rev. 96
B-208-005 AHV-35, Containment Purge Valves 3A & 3D, Rev. 20
SS-201-061-AC-1, 480V AC Main Transformer Dist. Panel (Normal), Rev. 12
SS-201-061-AC-2, 480V AC Main Transformer Dist. Panel (Alternate), Rev. 10
SS-201-061-AC-1A, 480V AC Main Transformer Dist. Panel (Normal), Rev. 5
SS-201-061-AC-2A, 480V AC Main Transformer Dist. Panel (Alternate), Rev. 3

1R08: Inservice Inspection Activities
Partial List of Documents Reviewed
Crystal River Unit 3 ASME Section XI Inservice Inspection Program Internal 3 and Ten Year (NDE) Program, Dated August 12, 2003
Nondestructive Examination Procedure NDEP-0437, Rev. 0, Ultrasonic Examination Procedure for Ferritic Pipe Welds (PDI)
Nondestructive Examination Procedure NDEP-0301, Rev. 14, Dry Powder Magnetic Particle Examination Plant Operating Manual SP-208, Rev. 5, Visual Examination of Component Supports
Nondestructive Examination Procedure NDEP-0612, Rev. 17, VT-2 Visual Examination of Nuclear Power Plant Components
Standard Procedure EGR-NGGC-0207, Rev. 0, Boric Acid Corrosion Control
Ultrasonic Calibration/Examination Report No. UT-03-116 on October 15, 2003 for Pipe Weld Mark C2.1.857, Drawing CR3-P-SK-105.3, Rev. 1, 10" Diameter Feed Water Line to OTSG “A”
Ultrasonic Calibration/Examination Report No. UT-03-120 on October 16, 2003 for Pipe Weld Mark C2.1.130, Drawing CR3-P-SK-105.2, Rev. 1, 18" Diameter Feed Water Line to OTSG “A”
Magnetic Particle Examination Report Nos. MT-03-002 & 003 on October 15, 2003 for Pipe Weld Mark C2.1.857, Drawing CR3-P-SK-105.3, Rev. 1, 10" Diameter Feed Water Line to OTSG “A”
Magnetic Particle Examination Report No. MT-03-005 on October 16, 2003 for Pipe Weld Mark C2.1.130, Drawing CR3-P-SK-105.2, Rev. 1, 18" Diameter Feed Water Line to OTSG “A”
Visual Examination of Pipe Hanger, Support or Restraint (VT-1) Report No. VT-03-163 on October 16, 2003 for Pipe Lugs Mark No. D2.5.19 , Drawing CR3-P-SKH-217.1, Rev. 1, 6" Main Steam to Emergency Feed Water Pump
Visual Examination of Pipe Hanger, Support or Restraint (VT-1) Report No. VT-03-164 on October 16, 2003 for Pipe Lugs Mark No. D2.5.22 , Drawing CR3-P-SKH-217.2, Rev. 1, 6" Main Steam to Emergency Feed Water Pump
Drawing CR3-P-SK-105.2, Rev. 1, Feed Water to OTSG “A”
Drawing CR3-P-SK-105.3, Rev. 1, Feed Water to OTSG “A”
Drawing CR3-P-SKH-217.1, Rev. 1, 6" Main Steam to Emergency Feed Water Pump
Drawing CR3-P-SKH-217.2, Rev. 1, 6" Main Steam to Emergency Feed Water Pump
Pressurizer Nozzle Contingency Repair Brief
Framatome ANP Drawing 5028916 D, Rev. 3, Crystal River -3 Pressurizer Level Sensing and Water Sampling Nozzle Repair
Framatome ANP Process Traveler Pressurizer Nozzle Modification
R13 Mode 3 Reactor Building Walkdowns for Early Identification of Maintenance Deficiency (Boric Leaks)
Boric Acid Corrosion Training for Mode 3 Walkdowns
Visual Examination for Boric Acid Detection Reports VT-03-030 & 105
Nuclear Condition Report (NCR) No. 107868

2OS1: Access Control To Radiologically Significant Areas

Procedures, Instructions, Manuals, Guides

Administrative Instruction (AI) - 1507, Conduct of the Radiation Protection Subunit, Rev. 4
Health Physics Procedure (HPP) -106A, Radiation Work Permit Procedure, Revision (Rev.) 11
HPP-202A, Radiological Surveys and Inspections, Rev. 21
HPP-213A, Area and Equipment Postings, Rev. 11
HPP-219, Radiation Protection Failed Fuel Action Plan, Rev. 03
HPP-214, Very High Radiation Area Controls, Rev. 4
HPP-320, Whole Body Counting System Operation, Rev. 15
HPP-333, Dose Calculations for Members of the Public and Unmonitored Occupational Individuals, Rev. 1
HPP-510, Setup and Removal of Breathing Air Service Equipment for Supplied Air Respirator Use, Rev. 6,
Operations Procedure (OP-417) Containment Operating Procedure, Rev. 92
Fuel Procedure (FP)-409, Reactor Vessel Closure Head Removal, Rev. 20
Nuclear Operations Training, Radiation Control Contractor Technician Training, Plant Specific Training, Rev. 8
Nuclear Operations Training, Special Technical Training, Respirator User Hands-On Training, GN6C11C (Initial) and GN7C11C (Refresher), Rev. 0
Dosimetry (DOS) - Nuclear Generating Group Corporate (NGGC) Standard Procedure - 002, Dosimetry Issuance, Rev. 19
DOS - NGGC - 006, Personnel Exposure Investigations, Rev. 9
Corrective Action Program (CAP) - NGGC Standard Procedure - 006, Corrective Action Program, Rev. 9
Health Physics Surveillance (HPS) - NGGC Standard Procedure - 0013, Personnel Contamination Monitoring, Decontamination, and Reporting, Rev. 2

Radiation Work Permits (RWPs)

RWP-00000809, Refueling Outage (R)13 Inspections Under the Reactor Vessel, Task 00433433 07 02, ISI Under the Reactor Vessel Inspection, Rev. 06
RWP-00000809, R13 Inspections Under the Reactor Vessel, Task 00433433 07 20, Insulation Under the Reactor Vessel, Rev. 06
RWP-00000815, R13 Refueling Activities Moderate Radiological Risk, Rev. 0
RWP-00000822, R13 Old Reactor Head Movement to Storage Building, Rev. 0
RWP-00000827, R13 Steam Generator Nozzle Dam Installation and Removal, Task 00411306 01 0, Tasks Requiring Multi-Badge, Rev. 0
RWP 00000829, R13 Steam Generator Eddy Current and Associated Activities, Task 00411329 01 01, Tasks Requiring Multi-Badge, Rev. 0

Records, Data, and Drawings

Unit 3 Refueling Outage Daily Dose Reports, October 6, 2003 through October 24, 2003
Health Physics Shift Logs, Selected Entries for the Months of January 1 - 31, 2003; March 1 - 31, 2003; and October 6, 2003 through October 24, 2003
Contamination Occurrence Logs, August 2002, through October 24, 2003

Personnel Contamination Event Records and Supporting Dose Assessment Data, for Selected Evaluations Conducted from August 1, 2002, through October 24, 2003
Initial Intake Assessment Data Sheets documented from August 1, 2003, through October 24, 2003
Health Physics Survey Record 03-10-0139, Inside “D” Rings at 95 foot Elevation, Shielding Letdown Line Insie D Ring, 10/05/03
Health Physics Survey Record 03-10-0151, Upper D Ring, ‘A’ Steam Generator (S/G) Manway; and 03-10-0152, ‘B’ S/G Manway, 160 foot Elevation, 10/06/03
Health Physics Survey Record 03-10-0198, Lower ‘B’ S/G Manway; and 03-10-0199, ‘A’ S/G Lower Manway, 95 Elevation, 10/08/03
Health Physics Survey Record 03-10-0121 and 03-10-0155, Under Vessel - 95 foot Elevation Reactor Building (RB), conducted 10/06/03 and 10/07/03
Health Physics Survey Record 03-10-0207, RB Cavity 134foot and 160 foot Elevations RB, 10/08/03
Health Physics Survey Record 03-10-0569, -0615, RB New Letdown Cooler Room, 95 foot Elevation,10/19/03
Crystal River Unit 3 - 2001 Occupational Radiation Exposure Report, 04/12/02
Crystal River Unit 3 - 2002 Occupational Radiation Exposure Report, 04/17/03

Corrective Action Program Records

Crystal River (C) Nuclear Assessment Section (NAS), Radiation Protection Assessment, Report File Number (No.) C-RP-03-01, dated July 15, 2003
CNAS, Environmental and Chemistry Assessment, Report File Number (No.) C-RP-03-01, dated September 2, 2003
Self-Assessment Report No. 79558, Radiation Protection Annual Review, conducted March 31 - April 4, 2003
Self-Assessment Report No. 79942, Radiation Protection Refuel 13 Readiness, conducted April 28 - May 09, 2003
Significant Adverse Condition Investigation Report, Action Request (AR) Number (No.) 106453, Locked High Radiation Area (LHRA) Controls were not Implemented Properly During the October 4, 2003 Reactor Building Mode 3 Boron Corrosion Inspection Walk Downs, 10/04/03
Nuclear Condition Report (NCR) 79125, Locked High Radiation Area (LHRA) Flashing Light Failed, 12/09/02
NCR 87625, Lock on LHRA Gate Degraded, 03/12/03
NCR 89030, Worker Entered RCA without Electronic Dosimeter, 03/28/03
NCR 98425, Key to Administratively Locked Door Not Returned to Health Physics Department, 07/08/03

2OS2: As Low As Reasonably Achievable

Procedures, Manuals, and Guidance Documents

Administrative (ADM) - NGGC Standard Procedure - 0105, ALARA Planning, Rev. 6
AI - 1602, ALARA Committee, Rev. 4
Chemistry (CH) Procedure - 442, Reactor Coolant System Chemistry During Plant Shutdown, Rev. 0

OP - 209A, Refueling Outage Plant Shutdown and Cool-down, Rev. 1
Engineering (EGR) -NGGC-0005, Engineering Change, Rev. 19
EGR-NGGC-0201, Incorporation of ALARA for Design and Engineering Work, Rev. 2
EGR-NGGC- 0204, Evaluation and Selection of Materials for Plant Components, Rev. 5

Records, Worksheets, and Drawings

ALARA Work Plan (AWP) 03-0005, Reactor Head Disassembly, Maintenance, and Reassembly, Rev. 0
AWP 03-0006, Control Rod Drive Removal / Replacement, Rev. 0
AWP 03-0007, Bag, Transport, and Store Old Reactor Head, Rev. 0
AWP 03-0009, OTSG Manways & Handholes, RWP 826, Rev. 0
AWP 03-0010, Nozzle Dams, RWP 827, Rev. 0
AWP 03-0011, Eddy Current Activities, RWP 829, Rev. 0
AWP 03-0026, Reactor Building Scaffold Activities, RWP 813, 814, and 833, Rev1
AWP 03-0027, Maintenance Activities, RWP 797, 802, 805, 806, 815, 824, & 838, Rev. 0
In-Progress ALARA Evaluation, 03-0011, Eddy Current, Conducted 10/20/03
Refueling Outage (RFO) -13, Reactor Coolant System (RCS) CRUD Burst Cleanup Curve, 10/04/03
RCS Total Activity Versus Cobalt (Co) - 58, 10/02/03 through 10/07/03
RFO - 13 Crud Burst Dose Rates 10/04/03 through 10/07/03
Reactor Head Dose Rate Comparison for Refueling Outage (RFO)12 and RF 13
Once Through Steam Generator (OTSG) Tube Sheet Dose Rates RFO 8 through RFO 13
Crystal River Unit 3 Operator Logs: 10/05/03 through 10/06/03-
Drawing P - 35300 - 40 Piston Check Valve Nuclear Class, Rev. G
Drawing P1 - 75108-K-1, Welded Cover Pistron Check Valve, Rev. C
50.59 Evaluation, Regulatory (REG) - NGGC - 0010, Engineering Change 50220 Replaces the Reactor Vessel Closure Head and Covers the New Control Rod Drive Mechanism Structure
Health Physics Survey Record 03-10-0544, Reactor Head Storage Building,10/17/03
13 R High Risk Evolution Contingency Plan, Reactor Vessel Head Transfer Through the Equipment Hatch, 10/02/03

Corrective Action Program Documents

NCR-70046, Operations Department Exceeded 2002 Week 33 Dose Budget, 08/27/02
NCR-71386, Craft Personnel at Radiological Control Area (RCA) Access Area Failed to Understand ALARA Task Associated with Work Order, 09/19/02
NCR-74040, NAS Assessment of Maintenance Identified Inconsistent or Poor Radiation Work Practices, 09/02
NCR-80472, Elevated Dose Rates Found During Routine Survey, 12/30/02
NCR-91987, Inaccurate Dose Estimates, 04/30/03

2PS2: Radioactive Material Processing and Transportation

Procedures, Manuals, and Guidance Documents

HPS-NGGC-0001, Radioactive Material Receipt and Shipping Procedure, Rev. 16
HPS-NGGC-0002, Instructions for Utilizing the Chemnuclear Systems (CNS) 8-120B Cask, Rev. 12
HPP-219, RP Failed Fuel Action Plan, Rev. 3
HPP-112, Hard to Detect Radionuclide Analyses, Rev. 0
Waste Procedure (WP) -205A, Disposal of Spent Resin from Waste Decay Tank - 6, Rev. 3
Duratek Procedure TR-OP-035, Handling Procedure for Transport Cask CNS 8-120B, Certificate of Compliance Number 9168, Rev. 15
CAP-NGGC-0200, Corrective Action Program, Rev. 9

Shipping Records and Radwaste Data

03-055, Low Specific Activity, Dry Active Waste (DAW) Trash, 10/08/03
03-046, Low Specific Activity, Primary Filters, 08/26/03
02-046, Non-Department of Transportation (DOT) Regulated, Condensate Resin, 10/21/02
02-047, Non-DOT Regulated, Condensate Resin, 10/22/02
02-026, Type B, Primary Resin, 05/30/02
02-022, Type B, Primary Resin, 05/09/02
01-075, Type B, Specimen Capsules, 10/16/01
Results of Radiochemistry Crosscheck Program, 2nd Quarter, 2003
10 CFR Part 61 Analysis Reports, 2002 and 2003
Certificate of Compliance no. 9168, CNS 8-120B Shipping Cask, Rev. 12
50.59 Screen/Evaluation, ID No. 02-0499, Add Isolation Valve, WDV-1376, Downstream of WDV-39, -41, -43

Corrective Action Program Documents

Action Request (AR) 00087899, Received Shipment from Framatome with Higher than Expected Levels of Contamination, 03/17/03
AR 00095680, Shipments of Secondary Resin Sent to Envirocare for Disposal Without Including a Waste Class on NRC Form 541, 06/09/03
AR 00098703, Transuranic Activity in the DAW Waste Stream for 2002 and 2003 Was Below MDA, Even Though Analyses Prior to 2002 Identified Activity above MDA Levels, 07/14/03
Nuclear Assessment Section (NAS) Report C-RP-03-01, Radiation Protection Assessment, 07/15/03
Self-Assessment Report Number 79946, Radioactive Material Processing and Transportation, 06/16/03 - 06/20/03

4OA1 Performance Indicator Verification (71151)

Procedures, Manuals, and Guides

REG-NGGC-0009, NRC Performance Indicators, Rev. 2
CP-217, NRC Performance Indicator (PI) Program, Rev. 6

Records and Data

Monthly Occupational Exposure Control Effectiveness Report Data. August, 1, 2002 through August 31. 2003
Personnel Contamination Logs, August 01, 2002, through October 24, 2003
Electronic Dosimeter Alarm Evaluations, August 01, 2002, through October 24, 2003
Whole-Body Count Data, Routine and Investigational, July 1, 2002, through October 24, 2003
Health Physics Logs, January 01 - 30, 2003; March 01-28, 2003; October 01 - 24, 2003
Monthly RETS/ODCM PI reports, April 2003 - September 2003
Out-of-service Effluent Monitor Report and Selected Compensatory Sampling Records, August 2002 - August 2003
Liquid Radioactive Waste Release Permits 20160.006.211.L (12/27/02) and 30145.002.278.L (08/03/03)
Gaseous Radioactive Waste Release Permits 20071.020.162.G (12/31/02) and 30045.020.194.G (08/07/03)

Corrective Action Program Documents

AR 00090625, Incorrect Identification of Sr-85 in Effluent Samples, 04/16/03
AR 00091076, RM-A12 High Volt Settings Differ from OPs Log Book, 04/22/03
AR 00098147, Liquid Release Radioactivity Higher than Expected in June, 07/08/03
AR 00101599, Primary-to-secondary Leak Rate Increasing According to RM-A12, 08/11/03

Partial List of Documents Reviewed (Section 4OA5.3)
Engineering Change 50220R2, RVCH and Service Structure Replacement
Framatome Document 51-5023338-01, CR3 Replacement RV Closure Head Reconciliation
Framatome ANP Crystal River 3 Reactor Vessel Closure Head Replacement Quality Plan
Framatome Document 51-5032373-00, Crystal River Unit 3 Replacement RVCH NDE Final Report
A Sample of Framatome ANP Production Weld Data Sheets, NDE Records, RT Film, and Material Certification Records for RVCH Fabrication Welds S/C001, B/D001, S/P001, R/D001, R/D002, R/D006, and R/D007
Final Stress Relief Heat Treatment Report, including time-temperature strip chart for the RVCH
Hydrotest Report CC/CR 001, including Examination Report
Certified Material Test Report, including the Japan Steel Works Quality Plan for the Closure Head Forging (Heat 02W10-1-1), Heat Analysis, Record of Ultrasonic Examination, Mechanical Test Results, and Record of Magnetic Particle Inspection

LIST OF ACRONYMS

ADM	Administrative
AI	Administrative Instruction
ALARA	As Low As Reasonably Achievable
AP	Abnormal Procedure
ATWS	Anticipated Transient Without Scram
AR	Action Request
AWP	As Low As Reasonably Achievable Work Plan
BTP	Branch Technical Position
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CH	Chemistry
CNS	ChemNuclear Systems
Co	Cobalt
CoC	Certificate of Compliance
CP	Compliance Procedure
CRDM	Control Rod Drive Mechanisms
CY	Calendar Year
DAW	Dry Active Waste
DOT	Department of Transportation
DRP	Discrete Radioactive Particle
EC	Engineering Change
ECT	Eddy Current Testing
ED	Electronic Dosimeter
EDG	Emergency Diesel Generator
EGR	Engineering
EOP	Emergency Operating Procedure
EP	Emergency Preparedness
ET	Eddy Current Examination
FP	Fuel Handling Procedure
FFD	Fitness for Duty
FPC	Florida Power Corporation
FSAR	Final Safety Analysis Report
FW	Feedwater
HP	Health Physics
HPP	Health Physics Procedure
HPT	Health Physics Technician
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
IST	Inservice Testing
ITS	Improved Technical Specifications
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LOCA	Loss of Coolant Accident
MT	Magnetic Particle Test
MP	Maintenance Procedure

MPFF	Maintenance Preventable Functional Failure
NAS	Nuclear Assessment Section
NCR	Nuclear Condition Report
NCV	Non-cited Violation
NGGC	Nuclear Generating Group Corporate
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
OP	Operating Procedure
OS	Occupational Radiation Safety
OTSG	Once-Through Steam Generator
PI	Performance Indicator
PMT	Post Maintenance Testing
PCP	Process Control Program
PS	Public Radiation Safety
PSB	Plant Support Branch
PSI	Preservice Inspection
PWSCC	Primary Water Stress Corrosion
Radwaste	Radioactive Waste
RB	Reactor Building
RCA	Radiation Control Area
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REG	Regulatory
RETS	Radiological Environmental Technical Specification
Rev.	Revision
RFO	Refueling Outage
RG	Regulatory Guide
RIS	Regulatory Summary Issue
RP	Radiation Protection
RVCH	Reactor Vessel Closure Head
RWP	Radiation Work Permit
SFP	Spent Fuel Pump
S/G	Steam Generator
SSC	Safety System Component
SW	Nuclear Services Closed Cycle Cooling System
TI	Temporary Instruction
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	NRC Unresolved Item
UT	Ultrasonic Test
VHRA	Very High Radiation Area
VT	Visual Test
WDT	Waste Decay Tank
WO	Work Order
WP	Waste Procedure