UNITED STATES

NUCLEAR REGULATORY COMMISSION

REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

January 27, 2006

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 05000302/2005005

Dear Mr. Young:

On December 31, 2005, 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 9, 2006, 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.

Based on the results of this inspection, there was one finding of very low safety significance (Green). Also, a licensee identified violation, which was of very low safety significance is listed in Section 4OA7 of this report.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000302/2005005
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-302
License No.:	DPR-72
Report No:	05000302/2005005
Licensee:	Progress Energy Florida (Florida Power Corporation)
Facility:	Crystal River Unit 3
Location:	15760 West Power Line Street Crystal River, FL 34428-6708
Dates:	October 1, 2005 - December 31, 2005
Inspectors:	T. Morrissey, Senior Resident Inspector R. Reyes, Resident Inspector F. Wright, Senior Health Physicist (Section 2OS2) R. Hamilton, Senior Health Physicist (Sections 2OS1, 4OA1, and 4OA7) J. Diaz, Health Physicist (Sections 2PS2 and 4OA5) S. Vias, Senior Reactor Inspector (Sections 1R02 and 1R17) A. Vargas, Reactor Inspector (Sections 1R08 and 4OA5) W. Bearden, Senior Resident Inspector, Brown Ferry Unit 1, (Section 1R08) C. Fong, Reactor Inspector (Sections 1R02 and 1R17) K. Harper, Reactor Inspector (Sections 1R02 and 1R17) J. Blake, NRC Consultant (Section 1R08) C. Julian, Senior Program Manager (Sections 1R02 and 1R17) B. Schin, Senior Reactor Inspector (Section 4OA5)
Approved by:	Joel T. Munday, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000302/2005-005; 10/01/2005 - 12/31/2005; Crystal River Unit 3; Event Followup.

The report covered a three month period of inspection by the resident inspectors and inspections by region based engineering and health physics inspectors. One Green self-revealing finding was 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

Cornerstones: Initiating Events and Mitigating Systems

Green. A self-revealing finding was identified for failure to provide adequate condensate system procedural guidance to preclude condensate pump operation at critical speed. As a result, prolonged operation at critical speed caused the condensate pump to fail and subsequently, the reactor was manually tripped in anticipation of a loss of the normal heat sink. The licensee entered this issue into the licensee's corrective action program as nuclear condition reports (NCRs) 174440 and 174442.

This finding is more than minor because it affected the procedure quality attribute of the Initiating Events cornerstone and resulted in an event that upset plant stability and challenged critical safety functions. This finding also affected the equipment reliability attribute of the Mitigating Systems Cornerstone objective and resulted in a loss of the normal heat sink. Because two Cornerstones were affected, a Phase 2 analysis was required. The consequences of the finding were assessed through the Significance Determination Process (SDP) Phase 2, and although the likelihood of a unit trip was increased and resulted in a loss of the normal heat sink, the exposure time for this condition was less than 3 days and all other mitigation capabilities described on the Phase 2, SDP worksheet for transient (reactor trip) core damage sequences were maintained. Consequently, the finding was determined to be of very low safety significance (Green). The finding was associated with non-safety related equipment and therefore, no violation of regulatory requirements occurred (Section 4OA3.2).

B. Licensee-identified Violations

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

REPORT DETAILS

Summary of Plant Status:

Crystal River 3 operated at full power until October 9, when power coastdown for a refueling outage commenced. On October 29, during the plant shutdown from 82% power for refueling outage 14, the unit was manually tripped from approximately 16% power when condensate flow was lost. After the refueling outage, power operation resumed on December 9, and the plant reached full power on December 12. On December 24, the unit was reduced to 80% power to investigate increasing combustible gas concentrations in two of the three main stepup transformers. On December 28, power was reduced to 10% and the main generator was removed from service to inspect the two transformers. On December 31, the unit was shutdown to repair/replace two main stepup transformers.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection

Impending Adverse Weather: Hurricane Wilma

a. Inspection Scope

Between October 18-20, 2005, the inspectors reviewed the licensee's hurricane preparations for Hurricane Wilma which had entered the Gulf of Mexico. The licensee used the checklists in Emergency Management Procedure EM-220, Violent Weather, to plan actions should the storm approach. The inspectors verified that the licensee's violent weather committee had been established and that preparations were made for possible tropical storm conditions. The nuclear condition report (NCR) database was reviewed to verify that the licensee was identifying and correcting adverse weather protection issues.

b. Findings

No findings of significance were identified. The storm did not approach the site and violent weather conditions did not occur.

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 five evaluations reviewed are listed in the List of Documents Reviewed.

The inspectors also reviewed samples of changes 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 twenty one "screened out" changes reviewed are listed in the List of Documents Reviewed.

The inspectors also reviewed corrective action reports to confirm that problems were identified at an appropriate threshold, were entered into the corrective action process, and appropriate corrective actions had been initiated.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

Partial System Walkdowns

a. Inspection Scope

The inspectors verified the critical portions of equipment alignments for selected trains that remained operable while the redundant trains were inoperable. In addition, the inspectors verified containment configuration during core alterations and ensured provisions for achieving timely containment closure were available. The inspectors reviewed plant documents to determine the correct system and power alignments, and the required positions of select valves and breakers. The inspectors verified that the licensee had properly identified and resolved equipment alignment or containment barrier issues that could cause initiating events, impact mitigating system availability, or affect the ability to provided containment integrity. The inspectors verified the following two partial system alignments in system walkdowns using the listed documents:

- October 11, Emergency Core Cooling System (ECCS) Train 'B' (Decay Heat, and Building Spray) using procedures OP-404, Decay Heat Removal System, and OP-405, Reactor Building Spray System, during an ECCS Train 'A' outage.

- On November 8-9, containment configuration using procedures: CP-341, Containment Penetration Control; MP-114 Removal And Installation Of RB Equipment Hatches; and 14R High-Risk Evolution Contingency Plan, Time to 200F less than 2.5 Hours.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

Fire Protection Walkdowns

a. Inspection Scope

The inspectors walked down accessible portions of the plant to assess the licensee's implementation of their fire protection program. The inspectors checked that safety equipment was free of transient combustible material and other ignition sources. Also, fire detection and suppression capabilities, fire barriers, and compensatory measures for fire protection problems were verified. The inspectors checked fire suppression and detection equipment to determine whether conditions or deficiencies existed which could impair the function of that equipment. The inspectors selected the areas based on a review of the licensee's probabilistic risk assessment. Documents reviewed are listed in the attachment. The inspectors toured the following nine areas important to plant safety:

- Auxiliary Building 162' Spent Fuel Pool Area
- Fire Service Pump Building
- Make-up Pumps and Valve Alley Areas
- Cable Spreading Room
- 'B' ES Transformer, Start-Up Transformer, Unit 3 Step-Up Transformers, and the Off-Site Power Transformer Areas
- Auxiliary Building 95' Sea Water Pump Room
- Decay Heat / Building Spray Pump 'A' Cubicle
- Emergency Feedwater Initiation and Control (EFIC) Rooms
- Emergency Feedwater Pump (EFP)-3 Building

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures - Internal Flooding

a. Inspection Scope

The inspectors reviewed the Crystal River Unit 3, Final Safety Analysis Report (FSAR), Chapter 2.4.2.4, Facilities Required for Flood Protection, Design Basis Documents, and System Descriptions, that depicted protection for areas containing safety-related equipment to identify areas that may be affected by internal flooding. A walkdown of the EFP-3 building and the Decay Heat / Building Spray Pump 'A' Cubicle were conducted to ensure that flood protection measures were in accordance with design specifications. Specific plant attributes that were checked included structural integrity, sealing of penetrations, operability of sump systems, and adequacy of watertight doors between flood areas.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

On October 11, 2005, the inspectors observed maintenance personnel perform a heat exchanger inspection and operability assessment as part of Work Order 687545, Decay Heat Closed Cycle Heat Exchanger 1A, Shoot and Clean. The inspectors verified that the inspection and cleaning was performed in accordance with preventative maintenance procedure, PM-275, General Preventative Maintenance Work, Enclosure 3, DC Heat Exchanger Shoot & Clean. The inspectors observed the as-found condition to ensure the heat exchanger was in a condition to perform its design function.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

a. Inspection Scope

ISI Activities

The inspectors observed in-process ISI work activities, reviewed ISI procedures, and reviewed selected ISI records, associated with risk significant structures, systems, and components. The observations and records were compared to the requirements specified in the Technical Specifications (TS) and the ASME Boiler and Pressure Vessel Code (1989 Edition w/no Addenda), to verify compliance and to ensure that examination results were appropriately evaluated and dispositioned.

Specifically, non-destructive examination (NDE) activities were reviewed as follows:

Ultrasonic Testing (UT):

- Reactor Head Closure Stud #25-207-21, 22, 23, 24, 29, 30, 31, 32
- Weld #: MU-23B, Tee to Pipe Weld
- Weld#: MS-26C, Elbow to Pipe Weld
- Weld #: MU-66, Valve to Pipe Weld

Visual Testing (VT):

- Component #: Incore inst. Noz. Weld in Reactor Vessel, Incore Instrumentation Nozzles (52)

Penetrant Testing (PT):

- Weld #: DH-48B, Pipe to Elbow Weld
- Weld #: MUV-43, HPI Safe-end to pipe weld (MU-00-036)
- Weld #: MUV-43, HPI Safe-end to pipe weld (MU-00-039)

Magnetic Particle Testing (MT):

- Weld#: FW-95D, Elbow to Pipe Weld

Radiograph Testing (RT):

- Weld #: MUV-43, HPI Safe-end to pipe weld (MU-00-036)
- Weld #: MUV-43, HPI Safe-end to pipe weld (MU-00-039)

Qualification and certification records for examiners, equipment and consumables, and NDE procedures for the above ISI examination activities were reviewed.

The inspectors reviewed RT films for Class 1 and 2 welds for the following welds. Inspectors also reviewed welding process and procedures used. Materials used along with welders' qualifications were also verified to ensure compliance with ASME Code.

- Weld #: MUV-43, HPI Safe-end to pipe weld (MU-00-036)
- Weld #: MUV-43, HPI Safe-end to pipe weld (MU-00-039)

Reactor Vessel Upper Head Penetration Examination

The inspectors reviewed the results of the licensee's Bare Metal Visual Examination of the Reactor Vessel Head and 69 Control Rod Drive Mechanism Penetrations.

- Component #: RCRE-1, Reactor Vessel Head and 69 CRDMs

Boric Acid Corrosion Control (BACC) Inspection

The inspectors reviewed implementation of the licensee's BACC program to verify implementation of commitments made in response to GL 88-05 and Bulletin 2002-01. The inspectors reviewed the inspection records for a sample of BACC walkdown visual examination activities to verify that the examiners were adequately identifying and documenting boric acid leakage throughout the plant. The inspectors reviewed the inspection scope of the BACC program to ensure that it included locations where boric acid could cause degradation to safety-related components. The inspectors also reviewed associated corrective action documents to evaluate the engineering bases for conclusions regarding apparent cause and severity of discovered leaks, and justification for corrective actions.

The inspectors reviewed the following Work Orders (WO) to verify dispositioning of indications and defects in accordance with ASME Code requirements or an alternative approved by the U.S. Nuclear Regulatory Commission (NRC):

- WO-00603927, MUV-303 and Pipe Cap Leak
- WO-00310812, DHV-79 leaking at cap. Catch bag is installed (replace valve)

Steam Generator (SG) Tube ISI

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 fifteen examples of bobbin and rotating coil inspection ET data were reviewed to evaluate the adequacy of completed data analysis. In-situ testing of Tube 30-03 on OTSG A was observed. Additionally, the inspectors reviewed nine action requests (ARs) associated with OTSG examinations.

The inspectors also reviewed revision 28 of SP-305, "OTSG Inservice Inspection," to evaluate the changes to the procedure delineating the increased involvement of licensee personnel in the OTSG, inspection-results, evaluation processes. The review focused on procedure changes that involved issues discussed in the August 22-25, 2005, NRC Inspection Report 50-302/2005009.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the licensee's effectiveness in performing routine maintenance activities. This review included an assessment of the licensee's practices pertaining to the identification, scope, and handling of degraded equipment conditions, as well as common cause failure evaluations and the resolution of historical equipment problems. For those systems, structures, and components within the scope of the maintenance rule per 10 CFR 50.65, the inspectors verified that reliability and unavailability were properly monitored, and that 10 CFR 50.65 (a)(1) and (a)(2) classifications were justified in light of the reviewed degraded equipment condition. The inspectors conducted this inspection for the two degraded equipment conditions associated with the items listed below.

C NCR's 158650 and 158654, MUV-24 and MUP-1A Failures Were Not Evaluated For MR Impact
C NCR's 148617 and 148396, Chilled Water System exceeded its Maintenance Rule Limit for Functional Failures

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the risk impact of removing from service those components listed below and verified the licensee's associated risk management activities. This review primarily focused on equipment determined to be risk significant within the maintenance rule. The inspectors also assessed the adequacy of the licensee's identification and resolution of problems associated with risk management including emergent work activities. The licensee's implementation of their compliance procedure CP-253, Power Operation Risk Assessment, was verified in each of the following five work week assessments.

- Work Week 05W40, Risk assessment for operations with the pressurizer block valve RCV-11 shut and raw water pump 2B out of service for maintenance activities
- Work Week 05W41, Risk assessment for operations with the pressurizer block valve RCV-11 shut and Decay Heat Pump (DHP)-1A and Building Spray Pump (BSP)-1A out of service for preventive maintenance
- Work Week 05W42, Risk assessment for operations with the pressurizer block valve RCV-11 shut and DHP-1B and BSP-1B out of service for preventive maintenance
- Work Week 05W46, Risk assessment for operations with Emergency Diesel Generator (EGDG)-1A out of service for maintenance and EGDG-1B inoperable due to a loss of its associated jacket water pump and heaters
- Work Week 05W52, Risk assessment for operations with EGDG-1A out of service for surveillance and the main turbine off line for main stepup transformer inspection.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions

a. Inspection Scope

For the four non-routine plant evolutions described below, the inspectors reviewed the operating crew's performance, operator logs, control board indications, and the plant computer data to verify that operator response was in accordance with the associated plant procedures.

C October 29, Manual reactor trip and subsequent recovery in accordance with emergency operating procedures (EOP) -2, Vital Systems Status Verification, and EOP-10, Post-Trip Stabilization

C November 14, Inadvertent Train B Engineered Safeguards Actuation and subsequent recovery in accordance with AP-340, Invalid Engineered Safeguards Actuation

C December 9, Power escalation to Mode 1 in accordance with operating procedures (OP)-210, Reactor Startup and OP-203, Plant Startup

C December 28, Power reduction from 80 percent to 15 percent and removal of the main turbine from service in accordance with OP-204, Power Operations and OP-208, Power Shutdown

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following four nuclear condition reports (NCRs) to verify that the operability of systems important to safety was properly established, that the affected components or systems remained capable of performing their intended safety function, and that no unrecognized increase in plant or public risk occurred. The inspectors determined if operability of systems or components important to safety was consistent with technical specifications, the FSAR, 10 CFR Part 50 requirements, and when applicable, NRC Inspection Manual, Part 9900, Technical Guidance, “Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety. The inspectors monitored licensee 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 NGGC-CAP-200, Corrective Action Program.

- NCR 172484, DHV-34 Failed Stroke Time After Implementing Engineering Change (EC)- 62218
- NCR 173742, EGDG-1B Tripped During Surveillance
- NCR 174141, DHP-1B Decrease In Performance
- NCR 175994, Two loose/broken core baffle bolts

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

Cumulative Effects

The inspectors performed a semi-annual evaluation of the potential cumulative effects of all outstanding operator work arounds (OWAs). The inspectors reviewed the OWAs listed on the Work Around Overview and Status report. The inspectors evaluated these OWAs along with issues on the degraded equipment log for their cumulative effects, and discussed these potential effects with control room supervisors and operators. The inspectors reviewed the equipment out-of-service logs and walked down the control room and plant areas to verify OWAs were being identified and properly entered into the corrective action program.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated design change packages for 13 modifications, in the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstone areas, to evaluate the modifications for adverse effects on system availability, reliability, and functional capability. The modifications and the associated attributes reviewed were as follows:

- MOD 61474 SWP-2A Grout Baseplate, (Mitigating Systems) (Screen Out 05-0351)
  - Materials / Replacement Components
  - Structural

- MOD 51890 10CFR21 Evaluation of Whiting #25 Hoist Unit (Gear Case) Mounting Bolts for FHCR-5, RCCR-1, TBCR-1 (Mitigating Systems) (Screen Out 03-0167)
  - Materials / Replacement Components
  - Structural
  - Operating Event (OE) 39545

- MOD 50656 Snubber Reduction (Barrier Integrity) (Screen Out 03-0420)
  - Structural
  - Pressure Boundary

MOD 49332 EH Pump Replacement (Mitigating Systems) (Screen Out 03-0129)
  - Energy Needs
  - Material / Replacement Components
  - Process Medium

- MOD 53186 RWV-24 Replacement & Piping Modifications (Mitigating Systems) (Screen Out 03-0674)
  - Materials
  - Heat Removal
  - Operations
  - Pressure Boundary

- MOD 52829 EFP-3 Barring Motor Permanent Installation (Mitigating Systems) (Screen Out 03-0322)
  - Energy Needs
  - Materials / Replacement
  - Pressure Boundary

- MOD 61028 EDDG Fuel Oil Header Modification (Mitigating Systems) (Screen Out 05-0292)
  - Energy needs

- MOD 51992 SWH-427, N-351, SWR-336, 361, 524, & 525 Modifications, Child EC to EC 50289, (Barrier Integrity) (Screen Out 03-0046)
  - Pressure Boundary
  - Functional Properties
  - Seismic Qualifications
  - ASME (American Society of Mechanical Engineers (ASME) Repair/Replacement Requirements

- MOD 51801 Repair of Pressurizer Level Tap and Sampling Nozzles, (Barrier Integrity) (Screen Out 03-0620)
  - ASME Repair/Replacement Requirements
  - Pressure Boundary
  - Functional Properties

- MOD 51111 Hanger RWH-14 Requires Rework to Address Spalled Concrete and Maintain the Hanger as a Rigging Point (Barrier Integrity) (Screen Out 03-0022) (AR 00003617 - Cracked Section of Concrete))
  - Pressure Boundary
  - Material Replacement Components
  - Seismic Qualifications
  - Functional Properties

- MOD 54359 CR3 Cycle 13 Extension, (Initiating Events) (Screen Out 03-0654)
    - Material compatibility
    - Functional requirements
    - Environmental Qualification (EQ)
    - Structural evaluation
    - Justification for lack of 50.59

- MOD 59702 Reactor Coolant Pump Runback Initiation Point with 4 Reactor Coolant Pumps (RCPs) Operating (Initiating Events) (Screen Out 04-0683)
    - Timing sequence
    - Response time
    - Control signals

- MOD 58155 EFP-3 Pump/Speed Increaser "Undercut" Bolts, (Mitigating Systems) (Screen Out 04-0490)
    - Material compatibility
    - Functional requirements
    - EQ
    - Seismic qualification
    - Structural evaluation
    - Justification for lack of 50.59
    - Failure modes

For selected modification packages, the inspectors observed the as-built configuration. Documents reviewed included procedures, engineering calculations, modification 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 Action Requests (ARs) 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.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors witnessed and/or reviewed post-maintenance testing procedures and/or test activities, as appropriate, for selected risk significant systems to verify whether: (1) testing was adequate for the maintenance performed; (2) acceptance criteria were clear, and adequately demonstrated operational readiness consistent with design and licensing basis documents; (3) test instrumentation had current calibrations, range, and accuracy consistent with the application; (4) tests were performed as written with applicable prerequisites satisfied; and (5) equipment was returned to the status required to perform its safety function. The six post-maintenance tests reviewed are listed below:

- Surveillance Procedure SP-354A, Monthly Functional Test of the EGDG-1A, performed on November 15-16, after performing maintenance on EGDG-1A per WO's 579652 and 324730

- Operating Procedure OP-405, Reactor Building Spray System, performed on October 12, after performing maintenance on the BSP 1-A, per WO 728771

- Maintenance Procedure MP-402C, Limit Torque Valve Actuators; and Test Instructions EC-56504 and EC-56505, Appendix R Circuit Reroutes For DHV-34 and 35 respectively, performed on November 15 and November 17, after performing circuit reroutes per Work Order (WO) 552839 and 552840

- Maintenance Procedure MP-402C, Limit Torque Valve Actuators, performed on November 8, after performing operator refurbishment on Main Steam Valve MSV-55, per WO 503140

- Surveillance Procedure SP-435, performed on November 24, after replacing actuator motor on Decay Heat Valve DHV-6 per WO 783902

- Surveillance Procedure SP-343, Main Steam Isolation Valves Part Stroke Exercising, performed on December 7, after replacing solenoid valve MS-413-SV3 per WO 609013-01

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

Refueling Outage (RFO14)

a. Inspection Scope

The inspectors reviewed the licensee's RFO14 Outage Risk Assessment report to confirm the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing the outage plan. During the refueling outage, the inspectors observed and monitored licensee controls over the outage activities listed below. Documents reviewed are listed in the Attachment.

- Outage related risk assessment monitoring
- Controls associated with shutdown cooling, reactivity management, reduced inventory activities, electrical power alignments, containment closure and integrity, and spent fuel pool cooling

- Implementation of equipment clearance activities
- Refueling activities
- Reactor mode changes
- Reactor heatup and repressurization
- Containment cleanup and closeout inspection
- Reactor initial startup and reactor physics testing
- Reactor power ascension and related testing

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed the surveillance tests listed below to verify that technical specification surveillance requirements were followed and that test acceptance criteria were properly specified. The inspectors verified that proper test conditions were established as specified in the procedures, that no equipment preconditioning activities occurred, and that acceptance criteria had been met. Additionally, the inspectors also verified that equipment was properly returned to service and that proper testing was specified and conducted to ensure that the equipment could perform its intended safety function following maintenance or as part of surveillance testing. The following seven activities were observed/reviewed:

In-Service Tests:

- SP-349C, Emergency Feed Pump EFP-3 And Valve Surveillance
- SP-311, Diesel Fuel Transfer Pump DFP-1B Surveillance
- SP-340C, MUP-1A, MUP-1B, and Valve Surveillance

Surveillance Tests:

- SP-178, Containment Leakage Test-Type “A” Including Liner Plate

RCS Leak Detection Test:

- SP-317, Reactor Coolant System Leak Rate Determination

Containment Isolation Valve Tests

C SP-179C, Containment Leakage Test - Type “C” (Penetration 124, CFV-17)
C SP-179C, Containment Leakage Test - Type “C” (Penetration 427, MSV-130)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors evaluated the following one temporary modification and associated 10 CFR 50.59 screenings against the system design basis documentation and FSAR to verify the modification did not adversely affect the safety functions of important safety systems. Additionally, the inspectors reviewed licensee procedure EGR-NGGC-0005, Engineering Change, to assess if the modification was properly developed and implemented.

- MP-450, Temporary Power to “A” side Battery Chargers During 480 V Bus Outage

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

Access Control Licensee activities for monitoring workers and controlling access to radiologically significant areas were inspected. The inspectors evaluated procedural guidance and directly observed implementation of administrative and physical controls; appraised radiation worker and technician knowledge of, and proficiency in implementing Radiation Protection (RP) program activities; and assessed worker exposures to radiation and radioactive material.

Radiological postings and material labeling were directly observed during tours of the containment building, auxiliary building, radwaste processing area, and spent fuel pool. The inspectors conducted independent surveys in these areas and compared the results to licensee’s documented surveys. During plant tours, control of Locked High Radiation Area (LHRA) keys and the physical status of LHRA doors were examined. In addition, the inspectors discussed radiological controls for non-fuel items stored in the spent fuel pool and alternative storage arrangements for high activity non-fuel components and waste. The inspectors also reviewed selected RP procedures and radiation work permits (RWPs), and discussed current access control program implementation with RP supervisors.

During the inspection, radiological controls for work activities in High Radiation Areas (HRA) were observed and discussed. The inspectors attended an RP pre-job planning meeting for removal of the reactor head, its movement to the head stand and observed preparatory work activities. The inspectors observed workers' adherence to RWP guidance and Radiation Protection Technician (RPT) proficiency in providing job coverage. The inspectors evaluated dose controls, contamination controls and radioactive material control in the Radiologically Controlled Area. Controls for limiting exposure to airborne radioactive material were reviewed and operation of ventilation units and positioning of air samplers were also observed in containment and auxiliary buildings. The inspectors evaluated electronic dosimeter alarm setpoints for consistency with radiological conditions in and around the containment, auxiliary building and radwaste processing areas. In addition, the inspectors interviewed workers to assess knowledge of RWP requirements.

The inspectors evaluated worker exposures through review of data associated with discrete radioactive particle and dispersed skin contamination events. The inspectors reviewed whole body count data for 2004 and 2005 to assess internal exposures. Controls used for monitoring extremity doses and the placement of dosimetry when work involved significant dose gradients were reviewed.

Radiation Protection program activities were evaluated against 10 CFR Part 20; Regulatory Guide 8.38, Control of Access to High and Very High Radiation Areas in Nuclear Power Plants; and approved licensee procedures. Licensee guidance documents, records, and data reviewed are listed in the report Attachment.

<u>Problem Identification and Resolution</u> Eleven Nuclear Condition Reports (NCRs) associated with radiological controls, personnel monitoring, and exposure assessments were reviewed and discussed with RP supervisors. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with CAP-NGGC-0200, Corrective Action Program, Revision 16 (Rev.). Specific documents reviewed are listed in the report Attachment.

The inspectors completed 21 of the required 21 samples.

b. <u>Findings</u>

One licensee identified finding is described in Section 4OA7.

2OS2 <u>ALARA Planning and Controls</u>

a. <u>Inspection Scope</u>

<u>As Low As Reasonably Achievable (ALARA)</u> Guidance and implementation of the licensee's ALARA program during the RFO 14 were observed and evaluated by the inspectors. The inspectors reviewed ALARA plans, dose estimates, and prescribed ALARA controls for outage work tasks expected to incur the maximum collective exposures. The inspectors reviewed ALARA plans for high person-rem-expenditure activities included: Health Physics (HP) activities, reactor building scaffolding and

shielding, steam generator eddy current inspection and repair work, and maintenance. The inspector reviewed shielding installations in containment for the “B” “J leg”, letdown line, and those installed to support modification of the containment sump. The incorporation of planning, established work controls, expected dose rates, and dose expenditure into the ALARA pre-job briefings, ALARA Committee Minutes and RWPs for those activities were reviewed. Projected dose estimates detailed in ALARA work plan documents and noted differences with actual does expenditures were reviewed and discussed with cognizant HP staff. Licensee tools for tracking key high collective dose activities were reviewed by the inspectors.

ALARA training and use of mockups, prior refueling outage RFO 14, were reviewed and discussed with HP staff. Occupational radiation workers and HP technicians were interviewed to evaluate their understanding of ALARA dose reduction initiatives. Applicable procedures were reviewed to assess licensee controls for declared pregnant workers. The inspectors examined the dose records of all declared pregnant workers during 2004 and 2005 to evaluate total or current gestation dose.

Selected elements of the licensee's source term reduction and control program were examined to evaluate the effectiveness of the program in supporting ALARA program goals. Shutdown chemistry program implementation and the resultant effect on containment and auxiliary building dose rate trending data were reviewed and discussed with cognizant licensee representatives. Additionally, trends in the plant’s three-year rolling average collective exposure history, outage, non-outage and total annual doses for selected years were reviewed and discussed with licensee representatives. Individual and collective personnel exposure trends at the facility were also reviewed. Records of year-to-date individual radiation exposures sorted by work groups were examined for significant variations of exposures among workers. Post job ALARA critiques for Refuel 13 Outage RFO 13 and licensees collective dose reductions for 2004 were also reviewed and discussed with radiation protection staff.

The licensee's ALARA program implementation and practices were evaluated for consistency with UFSAR Chapter 12, Radiation Protection; 10 CFR Part 20 requirements; Regulatory Guide 8.29, Instruction Concerning Risks from Occupational Radiation Exposure, February 1996; and licensee procedures. Documents reviewed during the inspection of this program area are listed in Section 2OS2 of the report Attachment.

<u>Problem Identification and Resolution.</u> The inspectors reviewed the corrective action program documents listed in Section 2OS2 of the report Attachment that were related to the licensee’s ALARA program. The inspectors assessed the licensee’s ability to identify, characterize, prioritize, and resolve the identified issues in accordance with CAP-NGGC-0200, Corrective Action Program, Rev.16 and CAP-NGGC-0201, Self-Assessment Program, Rev. 8.

The inspectors completed 15 of 15 required samples.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

Waste Processing and Characterization The inspectors evaluated licensee methods for processing and characterizing radioactive waste (radwaste). Inspection activities included direct observation of processing equipment for solid and liquid radwaste and evaluation of waste streams characterization data.

Solid and liquid radwaste equipment was inspected for material condition, configuration compliance with the UFSAR, and consistency with Process Control Program (PCP) requirements. The inspectors reviewed the status of non-operational or abandoned in place radwaste equipment. The inspectors reviewed the licensee's administrative and physical controls of non-operational or abandoned in place radwaste equipment to prevent unmonitored releases, impact to operating systems, and contributions to unnecessary personnel exposure. Inspected equipment included liquid radwaste hold-up tanks; resin transfer piping, filters, and elements of the demineralization system. The inspectors inquired about waste system changes and whether changes were made to the waste processing systems since the last inspection. The inspectors discussed waste processing component functions and equipment operability with licensee staff. In addition, procedural guidance for resin transfer was evaluated and compared with current practices.

Licensee radionuclide characterizations for selected waste streams were reviewed and discussed with radwaste staff. For condensate resin, reactor coolant system (RCS) crud filters, seal injection filters, spent fuel pool crud filters, Waste Demin - 6, 15, 16, 17, 18, 19, and 20 tanks, auxiliary building air handling charcoal, and dry active waste (DAW) the inspectors evaluated the analyses for hard-to-detect nuclides and appropriate use of scaling factors. Representative samples of licensee waste stream characterization data results processed by an outside laboratory were reviewed for the period of January 2003 to January 2005. For selected shipment records, the inspectors performed independent waste classification calculations and the methodology used for resin waste stream mixing and concentration averaging was evaluated. The inspectors also interviewed cognizant radwaste staff and reviewed procedures guidance, to evaluate the licensee's program for monitoring changing operational parameters.

Radwaste processing activities were reviewed for consistency with the licensee's PCP, Rev. 6; and UFSAR, Chapter 11 - Radioactive Waste & Radiation Protection, Rev. 28. Waste stream characterization analyses were reviewed against regulations detailed in 10 CFR Part 61.55 and guidance provided in the Branch Technical Position on Waste Classification and Waste Form, 1983. Reviewed documents are listed in Section 2PS2 of the report Attachment.

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

The inspectors directly observed transportation activities including shipment packaging, surveying, blocking and bracing, vehicle placarding, vehicle checks, emergency instructions, preparation of waste disposal manifest, and the provision of shipping papers and special instructions to drivers. Specifically, the inspectors observed one incoming shipment and one outgoing shipment. The incoming shipments contained equipment for use during the outage. The outgoing shipment contained two sea land containers loaded with DAW. The DAW shipment was shipped as exclusive use, radioactive-low specific activity.

As part of the document review, the inspectors evaluated eleven shipping records for consistency with licensee procedures and compliance with NRC and DOT regulations. In addition, training records for one individual currently qualified to ship radioactive material was checked for completeness and the training curriculum provided. Documents reviewed during the inspection are listed in Section 2PS2 of the report Attachment.

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

Problem Identification and Resolution The inspectors reviewed licensee events reports and a self assessment related to radioactive material processing and transportation areas, to determine if problems were identified and entered in the system for resolution. Specifically, the inspectors reviewed NCR reports and interviewed cognizant licensee personnel to determine if problems were identified, properly characterized, prioritized, evaluated and corrected. Documents reviewed are listed in Section 2PS2 of the report Attachment.

The inspectors completed 6 of 6 required samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors sampled licensee records to verify the accuracy of reported Performance Indicator (PI) data for the period listed below. To verify the accuracy of the reported PI elements, the reviewed data were assessed against guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Rev. 3, and the PI Frequently Asked

Questions (FAQ) list.

Occupational Radiation Safety Cornerstone The inspectors reviewed the Occupational Exposure Control Effectiveness PI results for the period of October 2004 through October 2005. For the assessment period, the inspectors interviewed several individuals and reviewed corrective action documents. Report Section 2OS1 contains additional details regarding the inspection of controls for exposure significant areas and review of related corrective action documents. Documents reviewed are listed in Sections 2OS1 and 4OA1 of the report Attachment.

Public Radiation Safety Cornerstone The inspectors reviewed the Radiological Control Effluent Release Occurrences PI results for the period of October 2004 through October 2005. For the assessment period, the inspectors reviewed cumulative and projected doses to the public. The inspectors also reviewed licensee procedural guidance for collecting and documenting PI data. Documents reviewed are listed in section 4OA1 of the report Attachment.

The inspectors completed two of the required samples for IP 71151.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

.1 Daily Screening of Items Entered Into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee’s corrective action program. This review was accomplished by attending daily plant status meetings, interviewing plant operators and applicable system engineers, and accessing the licensee’s computerized database.

b. Findings

No findings of significance were identified.

.2 Annual Sample Review

a. Inspection Scope

The inspectors selected ten nuclear condition reports relating to Post-Maintenance Tests (PMT) for a detailed review to evaluate the effectiveness of the licensee’s corrective actions. Several NCRs selected documented NRC identified issues associated with PMTs that had not been adequately specified in the WO package.

NCRs reviewed are listed in the attachment.

b. Findings and Observations

The inspectors found there has been a long standing issue relating to lack of understanding of various portions of the PMT program, which has led to several examples of incorrectly specifying PMT requirements in work order packages. These examples are listed in the attachment. Issues include lack of knowledge of both the PMT process and use of the manual, and confusion of the program ownership. NCRs dating back through 2003, including an October 2004 priority 1 NCR (QA audit), documented similar issues in the PMT program. Corrective actions have included training in the PMT process and procedure, clarification of program ownership, and revisions to the PMT procedure, CP-113D, Post Maintenance Testing. The inspectors concluded that the licensee's long term corrective actions to date have not been effective as shown by the recent NCRs associated with incorrectly specified PMT instructions. Since the inadequate post maintenance tests were identified and corrected just prior to performing the tests, there have been no cases that resulted in an inoperability of a structural, system and component (SSC) or violation of a regulatory requirement as a result of an inadequately completed PMT. On November 17, the licensee wrote NCR 143932 to document the results of a self assessment on the PMT process. The assessment was very comprehensive and corrective actions to address issues in the PMT program are scheduled for completion through 2006.

.3 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's corrective action program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in section 4OA2.1 above, plant status reviews, plant tours, and licensee trending efforts. The inspectors' review nominally considered the six month period of July 2005 through December 2005, although some examples may have expanded beyond those dates when the scope of the trend warranted. The review also included issues documented in the Equipment Performance Priority List dated December 19, 2005; the System Health Report dated August 8, 2005, various nuclear assessment section reports, and various maintenance rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's 3rd Quarter 2005, Site CAP Rollup & Trend Analysis report. Corrective actions associated with a sample of the issues identified in the licensees trend report were reviewed for adequacy.

b. Findings and Observations

No findings of significance were identified. The inspectors in reviewing licensee performance over the last six months, noted two trends, discussed below.

First, an increasing trend in the number of minor jacket water cooling systems leaks associated with the EGDGs was noted. In all cases the EGDGs remained operable. The inspectors discussed the increasing number of documented jacket water leaks with the system engineer who had also noted the same trend. Priority 1 NCR 177191, which documented exceeding the EGDG support system functional failure performance criteria, due to the failure of the 1B EGDG jacket water pump on November 14, is also addressing the apparent degrading reliability trend associated with the jacket water system.

The second trend was noted in designating incorrect post-maintenance tests in work order packages. Additional details and examples are documented above in Section 4OA2.2.

4OA3 Event Followup

.1 Manual Reactor Trip

a. Inspection Scope

On October 29, 2005, at approximately 16 percent power, a manual reactor trip was initiated in anticipation of an emergency feedwater actuation. Condensate flow was lost due to a failure of the magnetic coupling of the only operating condensate pump. The inspectors verified that all safety-related mitigating systems had operated as designed. The inspectors reviewed plant response by reviewing operating logs, operator statements, post-trip data collected as part of AI-704, Reactor Trip Review and Analysis, and completed EOPs. The inspectors reviewed the licensee's corrective action documents including the root cause report. Additional information and a finding associated with this event is documented in Section 4OA3.2 below.

b. Findings

No findings of significance were identified.

.2 (Closed) Licensee Event Report (LER) 05000302/2005-003-00, Manual Reactor Trip and Subsequent Emergency Feedwater Actuation Due to Condensate Pump Loss

a. Inspection Scope

The inspectors reviewed the LER to evaluate the licensee's assessment of the event and to identify any licensee performance deficiencies associated with the cause.

b. Findings

Introduction: A self-revealing Green finding was identified for failure to provide adequate condensate system procedural guidance to preclude condensate pump operation at critical speed. As a result, prolonged operation at critical speed caused the condensate pump to fail and subsequently, the reactor was manually tripped in anticipation of a loss of feedwater flow.

Description: On October 29, 2005, at approximately 16 percent power, a manual reactor trip was initiated in anticipation of an emergency feedwater actuation. The unit was being shutdown for a refueling outage when the only operating condensate pump tripped on a motor electric overload. The licensee determined that the magnetic coupling had failed due to prolonged operation in the critical speed range where pump vibration is at its highest. The licensee determined that operating procedure OP-209A, Refueling Outage Plant Shutdown and Cooldown, failed to contain adequate guidance to minimize condensate pump operation in the critical speed region. Safety systems responded as designed and feedwater was supplied by the emergency feedwater system. The inspectors reviewed the licensee’s corrective action documents including the root cause report.

The licensee determined that operating procedure OP-209A, Refueling Outage Plant Shutdown and Cooldown, Revision 4, failed to contain adequate guidance to minimize condensate pump operation in the critical speed region. The procedure contained the precaution: "If possible, minimize operation of the condensate pumps between 300 - 600 rpm." The precaution was originally added to OP-603, Condensate System in February 1988 to address an engineering recommendation that critical frequencies should be rapidly passed through when ramping the unit. The precaution was added to OP-209A on original issue in October 2003.

Analysis: The failure to provide adequate condensate system procedural guidance to preclude condensate pump operation at critical speed was a performance deficiency. This finding is more than minor because it affected the procedure quality attribute of the Initiating Events cornerstone and resulted in an event that upset plant stability and challenged critical safety functions. This finding also affected the equipment reliability attribute of the Mitigating Systems Cornerstone objective and resulted in a loss of the normal heat sink. Because two Cornerstones were affected, a Phase 2 analysis was required. The consequences of the finding were assessed through the Significance Determination Process (SDP) Phase 2, and although the likelihood of a unit trip was increased and resulted in a loss of the normal heat sink, the exposure time for this condition was less than 3 days and all other mitigation capabilities described on the Phase 2, SDP worksheet for transient (reactor trip) core damage sequences were maintained. Consequently, the finding was determined to be of very low safety significance (Green).

Enforcement: The inspectors determined that since the finding was associated with non-safety related equipment, no violation of regulatory requirements occurred. The licensee has revised several operating procedures, performed an extent of condition review, and initiated a high level root cause evaluation. The issue has been entered into the licensee’s corrective action program as NCRs 174440 and 174442. The finding is identified as FIN 05000302/2005005-01, Inadequate procedure guidance resulted a Loss of Condensate flow and a Manual Reactor Trip.

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000302/2004009-02: Single Failure Vulnerability of Common Electrical Protection and Metering Circuits

This URI was related to URI 05000302/2004009-01, Unprotected Post-Fire Safe Shutdown Cables and Related Non-Feasible Local Manual Operator Action. Both URIs involved the design of the common electrical protection and metering circuit for the 3A and 3B safety-related 4160V switchgear. One common protection and metering circuit served both of the switchgear, such that fire damage to the common circuit or a single failure in the common circuit could result in a loss of both the 3A and 3B 4160V switchgear.

NRC report 05000302/2005007 dated June 16, 2005, provided a preliminary significance determination of White for URI 05000302/2004009-01 (fire vulnerabilities of the common protection circuit), which bounded risk associated with URI 05000302/2004009-02 (single failure vulnerabilities of the common protection circuit). URI 05000302/2004009-01 was closed and was replaced with Apparent Violation 05000302/2005007-01.

Subsequently in NRC report 05000302/2005011, dated September 1, 2005, the apparent violation was closed and replaced with a White finding and associated Violation 05000302/2005011-01. The licensee corrected the issue by modifying the protection and metering circuit so that any fault on the circuit (due to fire damage or single failure) could only affect one train of safety-related switchgear. Because the licensee had already implemented corrective actions, that White finding and violation were closed. Because the risk associated with the condition that is the subject of URI 05000302/200409-02 has already been considered and corrective action completed, this URI is closed.

.2 (Closed) URI 05000302/2004009-04: No Cooling to Reactor Coolant Pump Seals for up to Eight Hours

The licensee's post-fire safe shutdown analysis (SSA) and procedures relied on the ability of the reactor coolant pump (RCP) seals to hold reactor coolant system pressure without any cooling for up to eight hours without failing or leaking more than normal. This issue was left open as a URI for further NRC review of the technical basis for assuming that the RCP seals had this capability. Crystal River 3 had Byron-Jackson (now Flowserve) N-9000 seal cartridges installed in the RCPs. Further NRC review of licensee and vendor information did not identify a current basis to dispute the RCP seal cooling assumptions in the licensee's SSA. Consequently, this URI is closed.

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

The inspectors reviewed the licensee's 60-day response to NRC Bulletin 2004-01, dated July 26, 2004. The inspectors verified that the licensee's examinations conducted during October 27 - November 1, 2005 were consistent with the licensee's response to

BL 2004-01.

The inspectors observed the Bare Metal Visual (BMV) examination performed on a sample of the welds that fall under the scope of BL-2004-01. BMV examinations were observed for the following welds:

- 1 vent and sample nozzle
- 1 sample nozzle (water)
- 1 thermowell
- 1 spray valve nozzle
- 3 level sensing nozzles (water)
- 2 relief valve nozzles

The inspectors reviewed the BMV examination documentation for the above welds as well as the verification of personnel qualifications of individuals performing the visual exams to ensure compliance with ASME Section XI, VT-2.

The inspectors also reviewed the licensee qualified procedure used for the BMV examination to ensure that it contained specific instructions related to the identification, disposition, and problem resolution.

The inspectors accompanied Crystal River Unit 3 personnel to the top of the pressurizer to ensure that the physical conditions of the pressurizer nozzle to safe end connections were clean and accessible for the prescribed inspections. The inspectors also verified that there were no issues with debris, insulation, dirt, boron from other sources, physical layout, or viewing obstructions which could have interfered with the identification of relevant indications.

The VT-2 BMV examinations did not result in any indications in the area of interest. The inspectors also reviewed the examination documentation to verify conformance to commitments made in response to BL 2004-01.

Reporting Requirements are as follows:

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

1. Performed by qualified and knowledgeable personnel?

   Yes. The licensee used a knowledgeable staff member certified as Level II, VT-2 examiner.

2. Performed in accordance with demonstrated procedures?

   Yes. The licensee performing the bare metal inspection of the pressurizer penetrations in accordance with the qualified procedure.

3. Able to identify, disposition, and resolve deficiencies?

Yes. The inspectors concluded that the licensee's direct visual examinations were capable of detecting leakage from cracking in pressurizer penetrations if it had existed. This conclusion was based upon the inspectors direct observations of pressurizer penetration locations, which were free of debris or deposits that could mask evidence of leakage in the areas examined. The inspectors also verified that the licensee's procedures included guidance for proper disposition and investigation of any identified deficiencies.

4. Capable of identifying the leakage in pressurizer penetration nozzle or steam space piping components, as discussed in NRC Bulletin 2004-01?

   The inspectors verified that the licensee's examination personnel were capable of identifying any leakage in pressurizer penetration nozzles.

b. What was the physical condition of the penetration nozzle and steam space piping components in the pressurizer system?

   Through observations, the inspectors verified that the metal reflective insulation had been removed. The licensee personnel performed a direct visual inspection of these pressurizer penetrations. Based on this examination, the area examined was generally clean and free of debris or deposits or other obstructions which could mask evidence of leakage.

c. How was the visual inspection conducted?

   The licensee's inspection personnel used the direct visual examination technique.

d. How complete was the coverage?

   The licensee was able to view the entire circumference, 360°, around each penetration.

e. Could small boron deposits, as described in the Bulletin 2004-01, be identified and characterized?

   The examination personnel were appropriately trained and qualified to identify small boron deposits as described in the bulletin.

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

   There were no deficiencies identified that required repair.

g. What, if any, impediments to effective examinations, for each of the applied methods, were identified?

   There were no impediments for an effective examination.

h. If volumetric or surface examination techniques were used for the augmented inspections, what process did the licensee use to evaluate and dispose any indications that may have been detected as a result of the examinations?

Not Applicable. No augmented surface or volumetric examinations were performed. In accordance with the licensee's response, only a BMV examination was conducted this outage, and there were no indications identified that required further examination.

I. Did the licensee perform appropriate follow-up examinations for indications of boric acid leaks from pressure-retaining components in the pressurizer system?

Not Applicable. There were no indications of boric acid leaks from susceptible pressure-retaining components.

Two interim exits were conducted on November 10, 2005 and November 17, 2005 to discuss the findings of this region based inspection.

.4 (Closed)Temporary Instruction 2515/161: Transportation of Reactor Control Rod Drives (CRDs) in Type A Packages

a. Inspection Scope

The inspectors interviewed cognizant personnel and reviewed shipping logs from CY 2002 to November 18, 2005, to determine if the licensee had packaged and shipped any irradiated CRD mechanisms in DOT Specification 7A Type A packages. The inspectors determined that the licensee made one shipment of CRD mechanisms. The inspectors reviewed the shipping documents and interviewed licensee personnel who performed the shipping activities. The inspectors determined that the licensee shipped the CRD mechanisms in accordance with 10 CFR Part 71, 49 CFR Parts 170-189, and specifically with the requirements specified in 49 CFR 173.412 and 49 CFR 173.415.

The inspectors completed Phase I and Phase II of NRC Temporary Instruction 2515/161.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

On January 9, 2006, the resident inspectors presented the inspection results to Mr. D. Young, Site Vice President and other members of licensee management, who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee Identified Violations

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

Contrary to Technical Specification 5.8.2 an individual entered a properly posted locked high radiation area, having a dose rate of 1000 mrem at 30 centimeters from the source

without qualified radiation protection coverage. No overexposure occurred as the individual was present in the room for a short duration. This event is captured in the licensee's corrective action program as NCR 075491.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
M. Annacone, Manager, Engineering
W. Brewer, Manager, Maintenance
R. Davis, Manager, Training
J. Franke, Plant General Manager
J. Hays, Manager, Outage and Scheduling
T. Hobbs, Manager, Nuclear Assessment
J. Holt, Manager, Operations
P. Infanger, Supervisor, Licensing
M. Rigsby, Radiation Protection Manager
D. Roderick, Director Site Operations
J. Stephenson, Principal Nuclear Emergency Preparedness Specialist
D. Young, Vice President, Crystal River Nuclear Plant

NRC personnel:
B. Desai, Acting Chief, Reactor Projects Branch 3, NRC Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000302/2005005-01	FIN	Inadequate procedure guidance resulted a Loss of Condensate flow and a Manual Reactor Trip (Section 40A3.2)

Closed

05000302/2004009-02	URI	Single Failure Vulnerability of Common Electrical Protection and Metering Circuits (Section 4OA5)
05000302/2004009-04	URI	No Cooling to Reactor Coolant Pump Seals for up to Eight Hours (Section 4OA5)
2515/160	TI	Pressurizer Penetration Nozzles and Steam Space Piping Connections in U.S. Pressurized Water Reactors (NRC Bulletin 2004-01) (Section 4OA5)
2515/161	TI	Transportation of Control Rod Drives in Type A Packages (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R02: Evaluation of Changes, Tests, or Experiments

Full Evaluations

•	05-0125	Intake Canal PT-501 Survey Frequency Change (Mitigating Systems)
•	05-0240	GL 91-18 evaluation of RCV-11 (PORV Block) Closure to Isolate Leak on PORV Flow Path
•	05-0275	Generic Screen to Eliminate the Use of Hydrogen Recombiners
•	04-0640	Reactor Building Sump Level Instrument Mods (EC 59476)
•	05-0279	Revise EOPs/APs for PORV/Block Valve Being Closed per NCR 152691

Screened Out Items

•	04-0641	Reactor Building Sumps (EC 59477)
•	04-0045	Replace Battery Charger FSP-2B Fire Service Pump (EC 55484)
•	04-0127	Replace Circuit Breaker for Replacement Battery Charger (EC 55710)
•	04-0087	Correct Drawing for Field Cables (EC 55564)
•	05-0351	SWP-2A Grout Baseplate
•	03-0167	10CFR21 Evaluation of Whiting #25 Hoist Unit (Gear Case) Mounting Bolts for FHCR-5, RCCR-1, TBCR-1
•	03-0420	Snubber Reduction
•	03-0129	EH Pump Replacement
•	03-0674	RWV-24 Replacement & Piping Modifications
•	04-0664	Revise Piping Specification to Allow Use of ASME B31.1-2001 Addenda 2002 (EC 59453)
•	03-0322	EFP-3 Barring Motor Permanent Installation
•	05-0292	EDDG Fuel Oil Header Modification
•	03-0046	SWH-427, N-351, SWR-336, 361, 524, & 525 Modifications, Child EC to EC 50289
•	03-0620	Repair of Pressurizer Level Tap and Sampling Nozzles
•	03-0022	Hanger RWH-14 Requires Rework to Address Spalled Concrete and Maintain the Hanger as a Rigging Point
•	03-0654	CR3 Cycle 13 Extension
•	04-0683	Reactor Coolant Pump Runback Initiation Point with 4 RCP(s) Operating
•	04-0490	EFP-3 Pump/Speed Increaser "Undercut" Bolts
•	04-0253	Advance Condition Investigation Form CAP-NGGS-0200-3-11, (AR 130907) Equivalency Determination

- 06830 Valve, Two-way, Butterfly Assembly ,8"-125# Flange, W/ D-3246 Actuator
- 62136 Head, Pump Discharge [for Raw Water Pump RWP-3B]

Sections 1R02 & 1R17:

Procedures

- EGR-NGGC-0005, Engineering Change, Rev. 23
- CP-213, Preparation of a Safety Assessment and Unreviewed Safety Question Determination, Rev. 9
- MCP-NGGC-0401, Material Acquisition (Procurement, Receiving, and Shipping), Rev. 18
- REG-NGGC-0010,10 CFR 50.59 and Selected Regulatory Reviews, Rev. 8
- MNT-NGGC-0005, Control of Rigging and Temporary Loads, Rev. 1
- PM-164, Raw Water Lining Inspection, Rev. 4
- SP-344A, RWP-2A, SWP-1A, and Valve Surveillance, Rev. 45

Self-Assessment Documents

- SAST 58939, 2003, Self Assessment (AR# 88795)
- CE-ES-05-01, Nuclear Assessment Engineering Assessment
- AR 152907, Root Cause Response to NAS-CE-ES-05-01
- AR 113084, Engineering Product Quality Process Effectiveness
- SAST 111423, Self Assessment Report 10CFR50.59 Evaluation Process and Program

Others

- AR 130907 Momentary Operating of PORV when Block Valve was Opened (NRC NCV 2004-004-01)
- AR 130907 Advance Condition Investigation Form CAP-NGGS-0200-3-11, (Screen Out 04-0253)
- AR125824 PORV Flow Experienced During Restoration from Leak Check
- AR 110023 Loss of Feedwater, Reactor Scram and EFC Actuation (NRC Finding 2004-003)
- AR 1200070 Unplanned RCS Cooldown During Shutdown
- LER 05-302/03-05-01 Rector Trip Caused by Loss of Feedwater While Troubleshooting Feedwater Control Problems
- AI-506 Operational Decision Making Evaluation for RCV-11 PORV/Block Valve

Section 1R05: Fire Protection

Procedures

AI-2205A, Pre Fire Plan - Control Complex
AI-2205B, Pre Fire Plan - Turbine Building
AI-2205C, Pre Fire Plan - Auxiliary Building
AI-2205F, Pre Fire Plan - Miscellaneous Buildings and Components

Section 1R08: ISI Activities and 4OA5: Other

Procedures

Crystal River Steam Generator Integrity Program, Rev. 5
Framatome Field Procedure for In-situ Pressure Testing of OTSG Tubes Using Triplex Pump, February 10, 2005
ARVEVA 51-5005589-03, Qualified Eddy Current Examination Techniques for Crystal River, Rev. 3

ARVEVA 51-5035818-03, Condition Monitoring and Operational Assessment Report, September 15, 2005
Crystal River Eddy Current Data Analysis Guidelines for the Once Through Steam Generator Inservice Inspection , Rev 3, October 6, 2005
Progress Energy Surveillance Instruction SP-305, OTSG Inservice Inspection, Rev 28
NDEP-0457, Ultrasonic Examination of Dissimilar Metal Welds (PID), Revision 0
NGGM-PM-0011, NDEP-A, Nuclear NDE Program and Process
NDEP-0457, Ultrasonic Examination of Dissimliar Metal Welds (PDI), Revision 0
EGR-NGGC-0207, CR3 Boric Acid Control Program Awareness Training
NDEP-0612, Augmented Inservice Visual Examination of Inconel Components, Revision 18

Other Documents

EPRI TR-1003138, PWR Steam Generator Examination Guidelines
Crystal River Refueling Outage 14, 2005 OTSG Inservice Inspection Plan, Rev 1
Summary of eddy current examination data
Framatome ANP eddy current NDE examiner qualification records
ANATEC, Inc eddy current NDE examiner qualification records
Action Request (AR) 174492, Fibrous material found in OTSG bowls
AR 174656, Foreign material on upper tubesheet of A OTSG
AR 174657, Bubble testing of OTSGs revealed indication of leakage
AR 175216, Four I-600 tube plugs found defective
AR 175217, Axial indications in upper tubesheets of both OTSGs were detected with RPC which had not been identified with bobbin coil invalidating bobbin coil qualification for detection of ODSCC or IGA in tubesheets
AR 175247, Use of RPC probe in tubesheet areas requires revision of degradation assessment
AR 175808, Lower I-600 plug in tube 146-25 in B OTSG requires removal for tube stabilization
AR 175809, Circ indication in tube 73-55 in B OTSG
AR 175938, Undocumented reroll of tube 71-109 in B OTSG during outage in 1999
Work Order (WO) 00562534-01, RCSG-1B, Perform Eddy Current Inspection of the OTSG tubes in accordance with SP-305
WO 00540366-01, RCSG-1A, Perform Eddy Current Inspection of the OTSG tubes in accordance with SP-305
Boric Acid Corrosion “Just In Time BAC Training for Mode 3 Boron Corrosion Personnel” Training Module

Section 1R20: Refueling and Outage Activities

Procedures

OP-203, Plant Startup
OP-210, Reactor Startup
OP-204, Power Operations
OP-209A, Refueling Outage Plant Shutdown and Cooldown
AI-504, Guidelines for Cold Shutdown and Refueling
WCP-102, Outage Risk Assessment
WCP-103, Station Readiness for Reduced Inventory, Mode 4/3 Entry, and Mode 2/1 Entry
SP-324, Containment Inspection

Section 4OA2.2: Problem Identification and Resolution Annual Sample Review

NRC Identified NCRs

172430	PMT work instructions required clarification to determine the system condition necessary to constitute an adequate leak check 10/05
164359	No PMT assigned on Service Water Pump SWP-1A, after disassembling coupling for lubrication 7/05
164361	EFP-3 PMT, the surveillance and the WO did not give specific instructions for performing an engine start on the air motor that had been replaced 7/05
164370	PMT for replacement of the ICS Rate limited Card was revised by engineering. The adequacy of the original PMT was questioned by auditors as well as the process that was used to formulate the PMT 7/05
132516	The responsible engineer was bypassed during the WO Task review and approval process. This is the third CR that relates to confusion concerning the PMT program 7/04

Licensee Identified NCRs

170958	SAST 143932, Operations knowledge of the PMT program requirements and expectations are not consistent among shifts 9/05
170961	SAST 143932, Additional training needed for planners, craft, supervisors, schedulers, and required operations and engineering personnel on PMTs 9/05
138837	NAS Assessment C-MA-04-01, Recurring problems in the PMT program have not been resolved with the appropriate technical rigor and priority 10/04
126094	PMT responsibilities are not always clearly understood by the responsible departments in accordance with CP-113D and ADM-NGGC-0104. Examples include the use of the PMT Manual 5/04
87508	CCHE Self-Assessment 51716, The CR3 Post Maintenance Testing manual has not been revised since 1998. Ownership of the PMT manual is not clear 3/03

Section 2OS1: Access Control To Radiologically Significant Areas

Procedures, Manuals, and Guides

CAP-NGGC-0200, Corrective Action Program, Rev. 16
CAP-NGGC-0201, Self Assessment Program, Rev. 8
DOS-NGGC-0001, Dosimetry Records Management, Rev. 9
DOS-NGGC-0002, Dosimetry Issuance, Rev. 22
DOS-NGGC-0003, Xe-133 Skin Dose Calculation, Rev. 4

DOS-NGGC-0004, Administrative Dose Limits, Rev. 8
DOS-NGGC-0005, Skin Dose from Contamination, Rev. 18
DOS-NGGC-0006, Personnel Exposure Investigations, Rev. 10
DOS-NGGC-0007, Internal Dose Calculations, Rev. 9
DOS-NGGC-0008, In Vitro Bioassay, Rev.7
HPP-106A,Supplemental Instructions to HPS-NGGC-0014 Radiation Work Permits, Rev. 18
HPP-112, Hard to Detect Radionuclide Analyses, Rev. 0
HPP-113, Remote Monitoring Using Plant Information (PI) System, Rev. 0
HPP-115, Steam Generator Inspection and Maintenance (RP Steam Generator Coverage), Rev. 0
HPP-202A, Supplemental Instructions to HPS-NGGC-0003: Radiological Surveys and Inspections, Rev. 26
HPP-209 Radioactive Material Storage Warehouses (RMSW), Rev. 20
HPP-215, HP Source Receipt and Control, Rev. 7
HPP-219, RP Failed Fuel Action Plan, Rev. 4
HPP-220, RP Response to OTSG Tube Leaks, Rev. 0
HPP-221, High Radiation Area, Locked High Radiation Area, and Very High Radiation Area Controls, Rev. 4
HPP-222, RP Planning PreJob Briefings and Post Job Reviews, Rev.1
HPP-320, Whole Body Counter Operations, Rev. 18
HPS-NGGC-0013, Personnel Monitoring, Rev. 3
HPS-NGGC-0014, Radiation Work Permits, Rev. 3
HPS-NGGC-0016, Access Control, Rev. 2

<u>Corrective Action Program (CAP) Documents</u>
NCR 128875, Results from NAS assessment C-RP-04-01-W3
NCR 147036, Equipment stored in clean area was found to have smearable contamination
NCR 150132, Daily Alarm Checks for SAM-9 and SAM-11 were recorded on a prior revision of form.
NCR 161446, RCA Boundary Not Intact (Posting Moved 3 feet to Accommodate EP Drill)
NCR 164478, Inconsistent Contaminated Area Postings
NCR 165696, Request for Safety Evaluation of Locked High Radiation Area Gates
NCR 167445, Radiological Posting Defaced
NCR 174320, Radiological Posting Found Mis-Positioned
NCR 174773, Radiological Posting Partially Covered with Ear Muffs
NCR 175132, Radiation Area Posting Missing
NCR 175491, Potential Unauthorized Entry Across LHRA Boundary

<u>Other Records/ Documents</u>
Whole body count result summaries for counts performed in 2004 and 2005
Radiation Worker Training Initial and Retraining Lesson Plan(GNB02N and GNR02N) discussing expected response to electronic dosimeter alarms

<u>Section 2OS2: ALARA Planning and Controls</u>
<u>Procedures, Manuals, and Guides</u>
ADM-NGGC-0105, ALARA Planning, Rev. 7
AI-1602, ALARA Committee, Rev. 5
CAP-NGGC-0200, Corrective Action Program, Rev. 16

CAP-NGGC-0201, Self Assessments, Rev. 8
DOS-NGGC-0002, Dosimetry Issuance, Rev. 22
HPP-106A, Supplemental Instructions to HPS-NGGC-0014:Radiation Work Permits, Rev.18
HPP-202A, Supplemental Instructions to HPS-NGGC-0003: Radiological Surveys and Inspections, Rev. 26
HPP-221, High Radiation Area, Locked High Radiation Area, and Very High Radiation Area Controls, Rev. 3
HPP-222, RP Planning Pre-Job Briefings and Post-Job Reviews, Rev. 1
HPS-NGGC-0014, Radiation Work Permits, Rev. 3
MNT-NGGC-0003, Radiation Shielding Use, Rev. 8

CAP Documents
NCR 139398, PM-178B Dose Estimates Were Exceeded During “A” ECCS Outage
NCR 143636, Ops Dose Projections Often Do Not Match Actual Dose Receive
NCR 149345, Questioning Posting of Inaccessible Areas Greater Than 100 mrem/hr
NCR 149768, Lead Blanket On Instrument Air Tubing at West End of Spent Fuel Pool
NCR 150472, Emergent Activity Affected Operations Dose Projection
NCR 157245, Unanticipated Dose During Work Week 05W17 (emergent work)
NCR 164235, Weekly Dose Estimates Have Been Very Conservative for Several Weeks and More Challenging Dose Estimates are Needed for CR3 to Keep Dose Reduction on Track
NCR 165851, The Waste Gas Analyzer Work Was Delayed Today Due To A Potential For Causing Elevated Dose In The Area By An Activity To Sluice Resin.
NCR 168812, Filter Canister Lid Disengaged While Loading Filters In HIC
Self Assessment 144272, CR3 - Industrial Health Program
C-RP-05-01, Crystal River Nuclear Plant Assessment Section, Radiation Protection Assessment, June 8, 2005

Other Records/ Documents
CR-3 Refuel 14 Project Plan For RCS Shut Down Chemistry Plan Including Hydrogen Peroxide Addition and CRUD Burst Clean Up, 8/10/05
Shutdown Script involving Chemistry, Operations and RP - R-14 First 3 days., 10/25/05
General Employee Training ALARA Module Computer Based Training (CBT), 10/5/05
Lesson Plan:GNQ0001C, Radiological Practical Factors Certification / ReCertification, Rev. 2
CR-3 2004 ALARA Report
CR-3 2003 ALARA Report
ALARA Committee Minutes: 05/05/05, 05/09/05, 06/08/05,and 09/27/05
ALARA Work Plan, 03-0002, Health Physics Activities
ALARA Work Plan, 03-0011, Steam Generator Eddy Current, Tube Plugging and Rolling
ALARA Work Plan, 03-0012, Refueling Activities and Equipment Modifications
ALARA Work Plan, 03-0025, Scaffolding Activities
ALARA Work Plan, 03-0027, Maintenance Activities
ALARA Work Plan, 03-0028, 14R General Operations Support
Post-Job ALARA Critique, 03-0002, Health Physics Activities
Post-Job ALARA Critique, 03-0011, Steam Generator Eddy Current, Tube Plugging and Rolling
Post-Job ALARA Critique, 03-0012, Refueling Activities and Equipment Modifications
Post-Job ALARA Critique, 03-0025, Scaffolding Activities
Post-Job ALARA Critique, 03-0027, Maintenance Activities
In-Progress ALARA Evaluation, ALARA Work Plan 03-0037, Pressurizer Nozzle Repair

ALARA Work Plan, 05-0001, HP Activities
ALARA Work Plan, 05-0003, 14R Shielding Activities
ALARA Work Plan, 05-0004, Reactor Head Disassembly, Maintenance, Reassembly
ALARA Work Plan, 05-0009, CR3 Steam Generator Eddy Current Testing And Repair
ALARA Work Plan, 05-0018, Steam Generator Replacement Walkdown.
ALARA Work Plan, 05-0020, 14R Reactor Building Scaffolding
ALARA Work Plan, 05-0021, Reactor Building Sump Modification
ALARA Work Plan, 05-0023, MWST & Auxiliary Building Sump Clean Out

Section 2PS2: Radioactive Material Processing and Transportation

Procedures/Manuals and Guides

CAP-NGGC-0201, Self-Assessment Program, Rev. 8
Process Control Program (PCP), Rev. 6
OES-16, Abandoned Equipment, Methods, Documentation and Identification, Rev. 0
WP-210, Sluicing, Loading, Filling and Venting Waste Disposal Tanks WDT-15 Through WDT-21 Demineralizer System, Rev. 7
WP-205A, WDT-6 Primary Resin Tank Operations, Rev. 8
WP-204, Condensate Resin Dewatering, Rev. 1
HPS-NGGC-0001, Radioactive Material Receipt and Shipping Procedure, Rev. 20
HPS-NGGC-0002, Vendor Cask Utilization Procedure, Rev. 13
RWT-001, Hazardous Materials Training, Rev. 8

Records, Audits and Corrective Action Documents

RADMAN Database Report, pages 1 to 159 (Undated) (includes Part 61 sample analysis [all waste streams] and scaling factors used for waste characterization)
EVC-NGGC-0026, Rev. 1, Page 15, Low-Level Radioactive Waste Analysis Data Sheet, for Samples 04R007261 (2004 DAW Smears) and 03R003845 (2003 DAW Smears)
Shipping Documents for Shipments 05-063, 05-040, 05-036, 05-035, 05-034, 05-028, 04-063 04-038, 04-037, 04-017, 04-008, 04-004, and 02-019
Shipping Logs for CY 2002, 2003, 2004, and from January 2005 to September 13, 2005
HAZMAT Employee Training record (July 2003), Dangerous Goods Regulation (IATA) Training record (January 2004), Radioactive Material Packaging and Transportation AITA Update Training record (August 2003) for individual responsible for shipping.
CR3 Key Performance Indicator V. 2.01 Report (Screen Capture Image file), Radioactive Waste Generated for CYs 2002, 2003, 2004, and from January 2005 thru August 2005
C-RP-05-01 (CNAS-2005-0033), Radiation Protection Assessment (Undated Copy, assessment conducted on May 6, 2005)
Self-Assessment No. 79946, Radioactive Material Processing and Transportation
Drawing P-304-691, Sheets 1 to 3; Drawing FD-302-681, Sheets 1 to 6; Drawing FD-302-683, Sheet 1; Drawing FD-302-683, Sheets 1 to 3; Drawing FD-302-691, Sheets 1 to 3
Drawing FD-302-696, Rev. 6, Sheet 1, Reactor Coolant Evaporator System Flow Diagram (Details boundary for equipment abandoned in place)
Drawing FD-302-690, Rev. 9, Sheets 1 to 3, Miscellaneous Waste Evaporator System (details boundary for equipment abandoned in place)
MAR99-08-02-01, Abandonment of Miscellaneous Waste and Reactor Coolant Waste Evaporator (including Document Summary and Parts 1 thru 13)

NCRs 00158359 (open), 00157687 (Open), 00146358 (Open), 00135922 (Closed), 00132629 (Closed), 00132624 (Closed), 00168812 (Open), 00160662 (Open), 00109841 (Closed), 00174274 (Closed), 00176471 (Open), and 00176476 (Open)

Section 4OA1: Performance Indicator Verification

Procedures/Guidance Documents

REG-NGGC-0009, NRC Performance Indicators And Monthly Operating Report Data, Rev. 4
CAP-NGGC-0200, Corrective Action Program, Rev. 16
CAP-NGGC-0205, Significant Adverse Condition Investigations, Rev. 4

CAP-NGGC-0206, Corrective Action Program Trending and Analysis, Rev. 1
Occupational and Public Performance Indicator portions of NEI-99-02, Rev. 3

Records

Liquid Radioactive Waste Release Permit 40145.002.347.L, 10/04
Liquid Radioactive Waste Release Permit 40161.002.352.L, 11/04
Liquid Radioactive Waste Release Permit 40180.002.357.L, 12/04
Liquid Radioactive Waste Release Permit 50012.002.359.L, 01/05
Liquid Radioactive Waste Release Permit 50029.006.392.L, 02/05
Liquid Radioactive Waste Release Permit 50037.001.376.L, 03/05
Liquid Radioactive Waste Release Permit 50055.001.380.L, 04/05
Liquid Radioactive Waste Release Permit 50134.002.392.L, 08/05
Liquid Radioactive Waste Release Permit 50160.006.448.L, 09/05
Liquid Radioactive Waste Release Permit 50192.002.410.L, 10/05
Gaseous Radioactive Waste Release Permit 40065.020.259.G 10/04
Gaseous Radioactive Waste Release Permit 40072.020.263.G 11/04
Gaseous Radioactive Waste Release Permit 40077.020.267.G 12/04
Gaseous Radioactive Waste Release Permit 50006.020.273.G 01/05
Gaseous Radioactive Waste Release Permit 50014.020.277.G 02/05
Gaseous Radioactive Waste Release Permit 50022.020.282.G 03/05
Gaseous Radioactive Waste Release Permit 50027.020.286.G 04/05
Gaseous Radioactive Waste Release Permit 50058.020.303.G 08/05
Gaseous Radioactive Waste Release Permit 50064.020.308.G 09/05
Gaseous Radioactive Waste Release Permit 50072.020.312.G 10/05
SQL Queries of PASSPORT AR System for Action Request by keyword for each month from 10/04 through 10/05
RCA Daily Status Reports for 10/04 through 10/05 (highest doses, dose rate alarms, etc.)
NCR 00175491, Potential Unauthorized Entry Across LHRA Boundary

Section 4OA5. Other Activities

Calculations and Analyses

B&W Document I.D. 51-1172516-00, N-9000 Seal Appendix R Evaluation
Crystal River Calculation M88-1026, RCP N-9000 Seal Appendix R Evaluation, Rev. 0