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

October 29, 2001

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 50-302/01-03

Dear Mr. Young:

On September 29, 2001, the NRC completed an inspection at your Crystal River Unit 3. The enclosed report documents the inspection findings which were discussed on October 12, 2001, 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, no findings of significance were identified.

Since September 11, 2001, your staff has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to you and your staff. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

John D. Monninger, Acting Chief
Reactor Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 50-302/01-03

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-302
License No:	DPR-72
Report No:	50-302/01-03
Licensee:	Florida Power Corporation (FPC)
Facility:	Crystal River Unit 3
Location:	15760 West Power Line Road Crystal River, FL 34428-6708
Dates:	July 1- September 29, 2001
Inspectors:	S. Stewart, Senior Resident Inspector S. Sanchez, Resident Inspector G. Kuzo, Senior Radiation Protection Specialist (Sections 2OS1, 2OS2, 2OS3, 4OA1) J. Kreh, Radiation Protection Specialist
Approved by:	J. Monninger, Acting Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000302-01-03, on 07/01-09/29/2001, Florida Power Corporation, Crystal River Unit 3, Resident Inspector Integrated Inspection Report.

The inspection was conducted by the resident inspectors and two regional radiation protection specialists. No findings of significance were identified by NRC inspectors during this inspection. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

None

B. Licensee Identified Violations

A 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 appeared reasonable. This violation is listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Crystal River Unit 3 operated at or near full power until September 29, 2001, when the plant was shutdown for the planned 12R refueling outage.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors conducted walkdowns of the plant external and internal areas such as the auxiliary building and emergency diesel generator areas to verify that the severe weather features described in the Updated Final Safety Analysis Report such as water tight doors and various seals were either in place or capable of being restored. The inspectors verified that the licensee maintained the spent fuel missile shields consistent with the Updated Final Safety Analysis Report Chapter 9. Abnormal Procedure AP-770, Flooding, and Emergency Management Procedures EM-202, Duties of the Emergency Coordinator, and EM-220, Violent Weather, were checked by the inspectors for consistency with site emergency response procedures and plant design. The inspectors verified the procedure readiness, such as having specified essential systems and necessity of emergency declarations as a postulated storm progressed.

On September 13, the inspectors assessed licensee preparations and activities for the pending Tropical Storm Gabrielle using the criteria included in EM-220. The inspectors attended a violent weather committee meeting, observed implementation of the adverse weather procedures, and observed licensee tracking of the storm. The inspectors verified flood protection features were either in place or ready and observed licensee activities to ready mitigating systems for potential adverse weather affects.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted partial alignment walkdowns of risk important systems to evaluate the readiness of the redundant trains or backup systems while one train was out of service for maintenance. The walkdowns included selected switch and valve position checks looking for discrepancies with operating procedures in effect, and verification of electrical power to critical components. The inspector reviewed sections of the Final Safety Analysis Report as applicable to each walkdown. During the walkdown of the vital inverters, the inspectors verified that a normal electric plant lineup existed by comparing the main control board electrical lineup to Crystal River 3 electrical diagram EC-206-011, Electrical One Line Composite. Nuclear condition reports and operator logs for the seven day period prior to maintenance on inverter VBIT-1E were reviewed by the inspectors for other electric plant problems. During an emergency diesel generator walkdown, the inspectors verified that control room switches and diesel support equipment were aligned as specified in licensee procedure OP-707, Operation of the ES Emergency Diesel Generators. During a decay heat cooling walkdown, the inspectors verified that the decay heat removal system valve and electrical lineups were consistent with flow drawings FD-302-641, Decay Heat Removal, Sheets 1 and 3, and the valve positions were as specified in OP-404, Decay Heat Removal. The inspectors checked that the decay heat pump was operable with lubrication oil in the oilers and that the borated water storage tank volume and boron concentration were consistent with Technical Specification requirements. The specific systems walked down were:

- Vital Inverters 1A, 1B, 1C, 1D, when VBIT-1E was out of service for (NCR 45119) preventive maintenance and the subsequent failure of an inverting rectifier

- Emergency diesel generator A and applicable switchgear during preventive maintenance and relay checks on emergency diesel generator B

- Decay heat closed cycle cooling, train A, with Train B out of service for preventive maintenance

The inspectors conducted a complete walkdown of the reactor building spray system (BS) pumps (BSP-1A and BSP-1B), and associated piping located outside the reactor building. The inspectors verified risk significant valve positions and piping runs were consistent with reactor building spray drawings FD-302-711 and FD-302-712; hanger drawings PI-305-810, sheets 1 and 2; and Final Safety Analysis Report, Chapter 6.2. The inspectors verified inservice test data for the pumps and several valves were within the limits specified in the licensee inservice testing program. The inspectors verified field and control room valve positions were consistent with operating procedure OP-405, Reactor Building Spray System. Nuclear condition reports (NCRs) 47447 and 41421 concerning building spray pump oil level problems were reviewed by the inspectors to verify that issues were being identified and addressed consistent with 10 CFR Part 50, Criterion XVI, Corrective Action. The inspectors reviewed open maintenance items for the building spray system to assess the cumulative degradation and repair priority of deficiencies.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours of risk significant plant areas to assure controls for transient combustibles and ignition sources were consistent with the licensee's Fire Protection Plan (FPP) 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 operational status and material condition of fire barriers used to contain fire damage using the standards of the Fire Protection Plan, 10 CFR Part 50, Appendix R, the Florida Power Corporation Analysis of Safe Shutdown Equipment, and the Final Safety Analysis Report. The inspectors reviewed sections of Administrative Instruction AI-2200, Guidelines for Handling, Use, and Control of Transient Combustibles; Surveillance Procedure SP-190A, Operability of Auxiliary Building Fire Detection System; SP-607, Fire Damper Inspection; SP-800, Monthly Fire Extinguisher Inspection; and SP-802, Fire Hose Hydro Test and Hose Reel Inspection, for consistency with the Fire Protection Plan and to check for proper field implementation.

The inspectors reviewed nuclear condition report (NCR) 44588, written for a spurious alarm of the auxiliary building fire detection system to assure that the licensee was identifying and correcting deficiencies using the corrective action program as specified by licensee procedure NGGC-200, Corrective Action Program. The inspectors verified that in response to the spurious alarm, the licensee initiated the hourly fire patrol specified in the Fire Protection Plan until a failed fire detector was identified and replaced. The inspectors reviewed NCR 48700 which was written after a spurious actuation of a charcoal filter deluge valve. The inspectors verified that the licensee initiated a continuous fire watch as specified in the Fire Protection Plan and secured the reactor building purge. The plant areas toured by the inspectors included:

- Makeup/High Pressure Injection Pump Rooms
- Train of Decay Heat Removal and Building Spray
- Battery Charger Rooms
- 1E Battery Rooms
- Emergency Diesel Generator Rooms
- Main Control Room

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed the licensee's inspection and cleaning of the seawater side of the 1A service water heat exchanger on September 4, 2001. The inspectors reviewed Technical Manual 00030, Nuclear Service Heat Exchanger, and Florida Power Corporation Calculation M97-0133, Service Water Heat Loads Following Large Break Loss of Coolant Accident, to assure that the acceptance criteria specified in the operations curve appropriately considered differences between design conditions and testing conditions.

The inspectors evaluated heat exchanger maintenance and the operational monitoring program using 10 CFR Part 50, Appendix A Criteria 44, 45, 46; Technical Specifications; Updated Final Safety Analysis Report Section 9.5; licensee Design Basis Summary Documents; and applicable parts of NRC Generic Letter 89-13, Service Water System

Problems Affecting Safety Related Equipment. The inspectors discussed with the responsible engineering and operations personnel service water system maintenance and monitoring and observed routine heat exchanger inspections.

Consistent with the licensee response to Generic Letter 89-13, the inspectors confirmed that the licensee routinely opened one heat exchanger for evaluation each week and verified that when the acceptance criterion for blockage was exceeded, the occurrence was documented in the licensee's corrective action program and a second heat exchanger would be opened for inspection as specified in licensee procedure OP-103B. The inspectors verified that the tube plugging for the four service water heat exchangers was consistent with the licensee's design basis for heat removal capability stated in calculation M97-0133.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed operator performance in the plant simulator. The licensee administered scenario included a few minor transients (e.g., radiation spikes and makeup pump oil leak) before progressing to a reactor coolant pump seal leak resulting in licensee operators performing a rapid reactor downpower, a reactor coolant pump trip resulting in a plant runback, and concluding with a steam generator tube rupture. The inspectors evaluated the operating crews conduct during abnormal and emergency operations, including abnormal procedure AP-510 (Rapid Power Reduction), AP-545 (Plant Runback), and emergency operating procedure EOP-6 (Steam Generator Tube Rupture). The inspectors observed the crew's ability to perform timely actions prescribed by the procedures and observed that oversight and direction provided by crew supervisors was consistent with licensee emergency procedures. The inspectors verified that crew emergency plan classifications and notifications were consistent with Crystal River Unit 3 Radiological Emergency Response Plan and 10 CFR 50.72. The inspectors also verified that the licensee evaluators assessed crew performance using the licensee operating standards and that the simulator facility closely matched the actual operating facility.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

.1 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed nuclear condition report 44374 concerning the stroke time of reactor coolant system valve RCV-11 (pressurizer pilot operated relief valve), and nuclear condition report 44603 concerning an emergency diesel generator B start problem. The inspectors reviewed maintenance rule scoping in accordance with 10 CFR 50.65 and characterization of the specified system or component problem in the licensees corrective action program. The inspector reviewed the licensee's maintenance rule program (a)(1) or (a)(2) classifications for RCV-11 and the 1B emergency diesel generator to assure consistency with licensee compliance procedure CP-153B, Monitoring the Performance of Structures, Systems, and Components Under the Maintenance Rule and for consistency with 10 CFR 50.65 requirements. The inspector reviewed other documents for consistency including applicable portions of the Final Safety Analysis Report; Technical Specifications; the licensee's Maintenance Rule Scoping Report; and the First and Second Quarter Year 2001 System Health Reports.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed daily maintenance schedules and observed work controls to evaluate risk before maintenance was conducted. The inspectors employed standards for operability of equipment such as those found in Technical Specifications, the Final Safety Analysis Report, licensee procedures, and regulatory information such as NRC Generic Letter 91-18, Revision 1, Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded And Nonconforming Conditions. The inspectors also reviewed maintenance schedules to assure that overall risk was minimized through preservation of safety functions such as decay heat removal capability, reactor coolant system inventory control, electric power availability, reactivity control, and primary containment control. The inspectors verified that licensee personnel were managing risk by assuring that key safety functions were preserved and that upon identification of an unplanned situation, the resulting emergent work was evaluated for risk and controlled as described in Technical Specifications, licensee Compliance Procedure CP-253, Power Operations Risk Assessment and Management, and Operations Instruction 7, Control of Equipment and System Status. The inspectors verified that emergent work was documented in the corrective action program. The inspectors evaluated risk controls associated with nuclear condition report 47318 which was written when raw water pump 1A had a higher than specified discharge pressure due to service water heat exchanger blockage. The inspector verified that the licensee took actions specified in Operations Procedure OP-103B, Operations Curves.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the technical adequacy of nuclear condition report 46091 to verify that operability of makeup pump MUP-1B was consistent with Technical Specifications, the Final Safety Analysis Report, 10CFR Part 50 requirements, and NRC Generic Letter 91-18, Revision 1, Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded And Nonconforming Conditions. The condition report was written following identification of an oil leak on the makeup pump gearbox lubricating system piping. The inspectors monitored licensee activities to verify that operability issues were being identified at an appropriate threshold, consistent with 10 CFR 50, Appendix B requirements, and licensee procedure NGGC-200, Corrective Action Program, and that risk was assessed when plant problems were identified.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors evaluated the following post-maintenance testing activities for risk significant systems to assess 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 Technical Specifications, the Final Safety Analysis Report, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications.

The specific post-maintenance activities evaluated included:

- PM-130C for testing vital inverter VBIT-1E following maintenance
- Surveillance Procedure SP-108, Reactor Trip and Control Rod Drive Trip Functional Test, following replacement of reactor trip circuit breakers
- SP-354B, Monthly Functional Test of Emergency Diesel Generator EGDG-1B, following replacement of a fuel injector pump and associated valves using Work Request Number 370549
- Preventative Maintenance Procedure PM-101, 4.16 kiloVolt and 6.9 kiloVolt Switchgear, following breaker maintenance using Work Request Number 363429

- Surveillance Procedure SP-340B, Decay Heat Pump DHP-1A, Building Spray Pump BSP-1A, and Valve Alignment, following installation of a dynamic vibration absorber on Decay Heat Pump 1A using Work Request Number 370998

b. Findings

No findings of significance were identified.

1R20 Outage Activities

a. Inspection Scope

During the week of September 23, 2001, the inspectors checked the licensee's risk control plans for refueling outage 12R, which commenced on September 29, 2001. In assessing risk management, the inspectors used applicable Technical Specifications, the Final Safety Analysis Report, 10 CFR Part 50 requirements, and guidance described in NRC Generic Letter 87-12, Loss of Residual Heat Removal While the Reactor Coolant System is Partially Filled; Generic Letter 88-17, Loss of Decay Heat Removal; Generic Letter 98-02, Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions while in a Shutdown Condition; and NRC Manual Chapter 0609, Appendix G, Shutdown Mitigation Capability. Additionally, the Region 2 Senior Reactor Analyst reviewed major outage tasks for specific risk configurations to be avoided. The inspectors reviewed the outage summary schedule, including FPC Administrative Instruction AI-504, Guidelines for Cold Shutdown and Refueling; Abnormal Procedure AP-404, Loss of Decay Heat Removal; and Temporary Instruction TI-OP-301-02, 12R Reactor Coolant System Drain and Fill Operations for consistency with the noted standards. The inspectors verified that the outage schedule and procedures established equipment redundancy in decay heat removal capability, inventory control, reactivity control, instrumentation, and electrical power distribution. The outage schedule also provided means to establish containment, if necessary, for the various shutdown conditions. The inspectors also discussed outage planning and conduct with licensee outage and operations management.

On September 29, 2001, the inspectors observed portions of the licensee's shutdown operations during the Technical Specification Mode 4 cooldown. The inspectors verified that the cooldown was conducted consistent with Technical Specifications and licensee procedure TI-OP-209-06, 12R Plant Shutdown and Cooldown.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed surveillance testing (SPs) or reviewed test data for risk-significant systems or components, to assess compliance with Technical Specifications, 10 CFR Part 50, Appendix B, and licensee procedure requirements. The testing was also evaluated for consistency with the Final Safety Analysis Report, NRC Bulletin 88-04, Potential Safety Related Pump Loss, NRC Generic Letter 89-04, Guidance on Developing Acceptable Inservice Testing Programs, and NUREG-1482, Guidelines for Inservice Testing at Nuclear Power Plants. The inspectors verified that the testing demonstrated that the systems were ready to perform their intended safety functions. During the inspections, consistent with 10 CFR Part 50, Appendix B, Criterion XVI, and licensee procedure CAP-NGGC-200, Corrective Action Program requirements, the inspectors verified that licensee personnel were documenting surveillance problems in the corrective action program. For example, nuclear condition report number 42310, documented that a service water valve, SWV-12, air solenoid had not been time response tested during the 11R refueling outage in 1999 as planned by the licensee. The inspector checked that corrective actions for the issue had either been completed or were scheduled and that no Technical Specification requirements were compromised.

Inservice test (IST) activities were reviewed to ensure testing methods, acceptance criteria, and required 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. The specific surveillance activities assessed included:

- SP-108, Reactor Trip Module and Control Rod Drive Trip Functional Test
- SP-340E, Decay Heat Pump DHP-1B, Building Spray Pump BSP-1B, and Valve Surveillance (IST)
- SP-112R, Reactor Protection System Reactor Building Pressure Trip Calibration
- SP-349B, Emergency Feedwater Pump EFP-2 and Valve Surveillance
- SP-110D, “D” Channel Reactor Protection System Functional Testing
- SP-340B, DHP-1A, BSP-1A, and Valve Alignment

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

During the week of August 20, 2001, the inspectors evaluated administrative and engineering controls for high radiation areas entered for maintenance activities conducted in accordance with Radiation Work Permit Number 010063, WDP-11 Repair Activities. The evaluation included attendance at pre-job briefings, and observations of work-in-progress and Health Physics (HP) technician job coverage. Conduct of selected radiation and contamination surveys was observed and results discussed. Electronic dosimetry setpoints were assessed and personnel exposure results were reviewed. In addition, during tours of auxiliary building and radioactive waste radiological significant areas, the inspectors observed and evaluated administrative and engineering controls for access to high radiation, locked-high radiation, and very high radiation areas.

The inspectors reviewed the activities against Final Safety Analysis Report (FSAR), Technical Specification (TS), and 10 CFR Part 20 requirements.

b. Findings

No findings of significance were identified.

2OS2 "As Low As Reasonably Achievable" Program Planning and Controls

a. Inspection Scope

The inspectors evaluated the licensee's "As Low As Reasonably Achievable" (ALARA) program implementation and dose expenditure results. Annual dose expenditure data histories were reviewed and discussed. Specific task details, in-progress ALARA evaluations, and the projected and final recorded dose expenditures for Decay Heat Valve-3 Canopy Seal Repair and Welding conducted in May and June 2001, were reviewed and discussed. The inspectors assessed dose rate and dose expenditure data trends associated with selected systems and equipment during past refueling outages; reviewed calendar year 2001 ALARA Review Committee meeting minutes, general dose reduction initiatives, and details of specific ALARA work plans (AWPs) developed for upcoming refueling outage 12 (RFO 12) activities; and examined selected Plant Equipment Equivalency Replacement Evaluations for recent valve replacements. General initiatives reviewed and discussed included RFO 12 shutdown chemistry and decontamination, cobalt reduction program status, and worker dose tracking. Knowledge of ALARA program guidance and staff proficiency in program implementation were appraised during discussions of selected AWPs with staff and management. The following AWPs associated with RFO 12 activities were reviewed by the inspectors and discussed with the responsible department manager or technical representative:

- Steam Generator Manways and Handholds
- Steam Generator Nozzle Dam Installation and Removal
- Steam Generator Eddy Current, Tube Plugging, Rolling
- Reactor Head Removal, Maintenance, Replacement
- Reactor Head Nozzle Inspection, Cleaning, and Repair
- Refueling Activities
- Reactor Building Scaffolding and Insulation
- Health Physics Refuel 12 Outage Activities

The inspectors reviewed the program guidance and implementation against the facility's CY 2000 ALARA goals, FSAR, 10 CFR Part 20 requirements, TS, and the following procedures:

- Chemistry Radiation Protection Instruction, ALARA Implementation Programs
- Health Physics Procedure (HPP) - 106A, Radiation Work Permit Procedure
- Radiation Safety Procedure - 600, ALARA Planning

- Administrative Instruction -1602, ALARA Committee Activities/Responsibilities

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation

.1 Radiation Monitoring

a. Inspection Scope

The inspectors evaluated the Operability and availabilty of Area Gamma Radiation Monitor (RM-G) systems and portable radiation monitoring instrumentation.

The inspectors directly observed equipment installation and material condition for selected RM-G systems, where accessible; compared local, remote, and control room RM-G indicator readouts; assessed current calibration methodology and results; reviewed selected system warning and alarm set-points; and observed functional testing of RM-G 13. The most recent calibration and/or functional test data were reviewed for the following RM-G systems:

- RM-G 01, Control Room
- RM-G 06, Makeup Tank Area
- RM-G-05, Waste Gas Decay Tank
- RM-G-09, Personnel Hatch
- RM-G-11, Deborating - Demineralizer Area
- RM-G-12, Spent Resin Storage Tank Area
- RM-G-13, Decontamination Pit
- RM-G-16, Reactor Building Fuel Handling Bridge
- RM-G-17, Reactor Building Personnel Hatch
- RM-G-18, Reactor Building Incore Instrumentation

The inspectors reviewed the "as installed" control room monitor (RM-G1) against applicable design base documents and drawings. Area monitor calibration, functional checks, and set-point data were evaluated against applicable sections of the FSAR, TS, and the following procedures:

- HPP-404, Area Radiation Monitoring System Calibration
- SP - 335A, Radiation Monitoring Instrumentation Functional Test

The inspectors evaluated the availability and operability of portable radiation monitoring instruments used to monitor gamma and neutron radiation during entry into selected high dose rate radiation fields between June 1, 2000, and January 16, 2001. Calibration data were reviewed for 10 portable radiation monitoring instruments including neutron and ion chamber instrumentation. Health Physics Technician proficiency and knowledge of selected portable instruments use was evaluated through direct observations of applicable staff performance during tasks conducted within the auxiliary building radiologically controlled area. In addition, onsite gamma calibration activities for portable radiation instruments were reviewed and discussed. Licensee activities were evaluated against applicable sections of the FSAR, TS, 10 CFR Part 20.

b. Findings

No findings of significance were identified.

.2 Respiratory Protective Equipment

a. Inspection Scope

The inspectors evaluated the licensee's respiratory protection program for operations, maintenance, and health physics staff members who potentially may be required to use self-contained breathing apparatus (SCBA) equipment. The inspectors examined the SCBA charging station facilities and associated equipment, and assessed availability and readiness of equipment at specified storage locations, including the Control Room. Records of training, fit testing, and medical qualifications for 10 individuals were reviewed for compliance with applicable requirements.

The inspectors evaluated the reviewed program activities against Regulatory Guide 8.15, Acceptable Programs for Respiratory Protection, Rev 1; 10 CFR 20.1703; Section 11.5.6 of the UFSAR; and the following licensee procedures:

- RSP-500, Respiratory Protection Program
- HPP-502, Respirator Inspection and Maintenance
- HPP-515, IAP-2 Operation and Maintenance

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

.1 Occupational Radiation Safety Performance Indicator Verification

a. Inspection Scope

The inspectors verified the Occupational Exposure Control Effectiveness performance indicator for the Occupational Radiation Safety Cornerstone from January 1, 2001, through June 30, 2001. The inspectors reviewed data reported to the NRC, and sampled and evaluated applicable corrective action program records, audits, and selected Health Physics Program records. Reviewed radiation protection program records included Health Physics Logs, personnel contamination events, skin and internal contamination evaluations, health physics surveys, and electronic device incident forms.

b. Findings

No findings of significance were identified.

.2 Public Radiation Safety Performance Indicator Verification

a. Inspection Scope

The inspectors verified the Radiological Control Effluent Release Occurrences performance indicator for the Public Radiation Safety Cornerstone from January 1, 2001, through June 30, 2001. The inspectors reviewed and discussed data reported to the NRC and evaluated applicable corrective action program records, control room logs, process radiation monitor operational data, abnormal effluent release results, and effluent records associated with routine liquid and gaseous effluent releases.

b. Findings

No findings of significance were identified.

.3 Barrier Integrity Performance Indicator Verification

a. Inspection Scope

The inspectors verified the accuracy of the performance indicators for reactor coolant system leakage and activity. Reactor coolant system leakage performance indicator data submitted in June 2001, was compared to data obtained through the review of control room logs from November 2000 through June 2001. Reactor coolant activity data from July 2000 through June 2001 and reported in June 2001, were reviewed by the inspectors to ensure the values reported were congruent with that recorded on plant chemistry logs and operator logs. The inspectors also reviewed the licensee corrective action program for relevant issues related to the collection of performance indicator data.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

(Closed) Licensee Event Report (LER) 50-302/01-001: Installation Error Results In Containment Isolation Valve Inoperable Longer Than Allowed By Technical Specifications

On May 14, 2001, the licensee identified that a feedwater check valve (FWV-46) was inoperable and that the associated technical specification limiting condition for operation had not been entered. Specifically, from May 13 to 14 for about 25 hours, FWV-46 was inoperable and the penetration was not isolated as required by Technical Specification 3.6.3 Condition C. The licensee determined that the check valve failure was due to the check valve disc separating from the hinge due to lack of procedural guidance to ensure proper installation of the check valve disk. The issues described in this LER have been previously discussed in NRC Inspection Report 50-302/01-02, Sections 1R12 and 1R17, and were entered into the licensee corrective action program as nuclear condition report number CR-42306. Corrective actions included replacing the containment isolation function of FWV-46 with three other feedwater block valves, repairing FWV-46, and revising Maintenance Procedure MP-120, "Maintenance of Pressure Seal Gate, Globe, and Check Valves," to ensure that a sufficient weld was made to prevent the retaining pins from backing out of position. This finding is more than minor because it had a credible impact on safety because if a steam generator tube ruptured, then backleakage through FWV-46 could result in an unmonitored release of radioactive reactor coolant through the feedwater piping to the environment. This finding affects the Barrier Integrity Cornerstone and was considered to have very low safety significance (green) because the likelihood of an accident leading to core damage was not affected, the probability of early primary containment failure was negligible, and the feedwater piping and steam generator tubes remained intact during this event. This licensee identified non-cited violation is discussed in Section 4OA7. This LER is closed.

4OA6 Meetings

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 October 12, 2001. 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.

4OA7 Licensee Identified Violations

The following finding of very low significance was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as NCVs.

NCV Tracking Number	Requirement Licensee Failed To Meet
NCV 50-302/01-03-01	Technical Specification 3.6.3 Condition C requires that a containment penetration flow path be isolated within 4 hours, if the associated isolation valve is not operable. Contrary to this from May 13 to 14, a feedwater check valve (FWV-46) was not operable, and the penetration was not isolated. This was identified in the licensee's corrective action program as CR-42306. This finding is only of very low significance because it only affects the barrier integrity cornerstone and all other mitigating systems were functional. (Green)

PARTIAL LIST OF PERSONS CONTACTED

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D. Roderick, Plant General Manager
J. Stephenson, Supervisor, Emergency Preparedness
J. Terry, Manager, Engineering
R. Warden, Manager, Nuclear Assessment
D. Young, Vice President, Crystal River Nuclear Plant

NRC

K. Barr, Chief, Plant Support Branch, NRC Region 2
M. Lesser, Chief, Engineering Branch 2, NRC Region 2

ITEMS OPENED AND CLOSED

50-302/2001-001	LER	Installation Error Results in Containment Isolation Valve Inoperable Longer than Allowed by Technical Specifications (Section 4OA3)
50-302/01-03-01	NCV	Installation Error Results in Containment Isolation Valve Inoperable Longer than Allowed by Technical Specifications (Section 4OA7)