UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report No.: 50-302/88-10

Licensee: Florida Power Corporation
3201 34th Street, South
St. Petersburg, FL 33733

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

Facility Name: Crystal River 3

Inspection Conducted: March 21-25, 1988

Inspector: S. E. Sparks 4-19-88
S. E. Sparks Date Signed

Accompanying Personnel: J. Zeiler

Approved by: Frank Jape 4/19/88
F. Jape, Chief, Test Programs Section Date Signed
Division of Reactor Safety

SUMMARY

Scope: This routine, unannounced inspection was in the areas of review of post-refueling startup testing, reactor coolant system leakage determination, review of local leak rate testing, and verification of containment integrity.

Results: No violations or deviations were identified.

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REPORT DETAILS

1. Persons Contacted

Licensee Employees

*J. Alberdo, Assistant Director, Nuclear Plant Operations
*P. D. Breedlove, Supervisor, Records Management
*M. Collins, Nuclear Safety and Reliability Superintendent
*J. Cooper, Nuclear Technical Support Superintendent
*G. H. Halnon, Nuclear Operations Tech.
W. G. Neuman, Supervisor of ISI
*W. L. Rossfeld, Manager, Nuclear Compliance
*M. S. Williams, Nuclear Regulatory Specialist
*E. Welch, Manager, Nuclear Electrical Engineering Services
*C. A. Woody, Reactor Engineer

Other licensee employees contacted included engineers, technicians, operators, mechanics, and office personnel.

NRC Resident Inspectors

*J. Tedrow, Resident Inspector
T. Stetka, Senior Resident Inspector

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on March 25, 1988, with those persons indicated in paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection findings. No dissenting comments were received from the licensee. The following new items were identified during this inspection:

Inspector Follow-up Item (IFI) 302/88-10-01: Verify licensee's revision to PT-114, Moderator Temperature Coefficient Determination.

IFI 302/88-10-02: Review licensee's evaluation as to the need to local leak rate test containment electrical penetrations.

The license did identify some material as proprietary during this inspection but, this material is not included in this inspection report.

3. Licensee Action on Previous Enforcement Matters

This subject was not addressed in the inspection.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Reactor Coolant System Leak Rate Measurement - Unit 3 (61728)

The licensee performed reactor coolant leakage measurements in accordance with Surveillance Procedure SP-317, RC System Water Inventory Balance, on March 23, 1988. The procedure requires that only starting and ending values be taken over an eight hour interval. The inspector reviewed the procedure for technical adequacy, and verified the correction factors used for Reactor Coolant System (RCS) temperature and pressure changes. The inspector observed plant operators during the performance of SP-317. Measurement sources for the various plant parameters were taken from recall points of the plant computer, and all leakages were determined by hand calculations and a four function calculator. The operators seemed knowledgeable about the procedure.

The measurements described above address the measurement of total leakage from the RCS. Leakage that is collected in the reactor coolant drain tank from reactor coolant pump seal leakoff, as well as that from measured leaks such as valve packing, is defined as identified leakage. The allowable leakage of one gallon per minute is the difference between the measured total leakage and the identified leakage.

The microcomputer program RCSLK9 was used by the inspector for verification of reactor coolant system leakage measurements, and is described in NUREG-1107. The parameter list is given in Attachment 1. A version of RCSLK9 had to be modified due to the unique configuration of the plant. The modification involved compensation for the addition of 0.2 gallons per minute flow rate into the reactor drain tank due to the reactor coolant pump seal standpipe flush water. The purpose of the standpipe flush is to preclude precipitation of boron around the standpipe and seals. The results of RCSLK9 are provided in Attachment 2, and demonstrate conformance to the Technical Specification limit of less than 1 GPM unidentified leakage. The RCSLK9 results also agree well with licensee results.

No violations or deviations were identified.

6. Post-Refueling Startup Testing - Unit 3 (72700, 61708, 61710)

a. OP-210, Reactor Startup, performed January 8, 1988

Reactor criticality was accomplished in an orderly fashion by diluting to a specified boron concentration, after which control rods were withdrawn. Deboration of the RCS commenced to a precritical concentration of 1955 ppmB. As required per procedure, two Estimated Critical Position calculations were performed and were in close agreement prior to reactor startup. Safety Rod groups were withdrawn to their upper limits. Regulating Rods were withdrawn until

criticality was achieved. Estimated and actual critical rod positions were as follow:

Rod Group	Estimated (% withdrawn)	Actual (% withdrawn)
1-4	100	100
5	100	100
6	100	100
7	34	42
8	25	25

The actual boron concentration at the time of criticality was 1955 ppmB.

b. PT-112, Hot Zero Power Regulating Rod Group Worth and Differential Boron Worth Measurement, performed January 9, 1988

The inspector reviewed procedure PT-112 to determine conformance with acceptance criteria. Predicted and measured rod group worths were as follows:

Rod Group	Predicted Worth	Measured Worth
5	1355 pcm	1401 pcm
6	927 pcm	832 pcm
7	881 pcm	838 pcm
5-7	3163 pcm	3071 pcm

The acceptance criteria of less than or equal to 15% for groups 5, 6, and 7 individually, and less than or equal to 10% for groups 5-7 were satisfied.

The differential boron worth of control rod groups 5-7 was measured to be 7.8 pcm/ppmB. This agrees well with the predicted value of 7.6 pcm/ppmB.

c. PT-114, Moderator and Temperature Coefficients Determination at Hot Zero Power, performed January 9, 1988.

Procedure PT-114 calls for the determination of the Moderator Temperature Coefficient (MTC) following a 5 degree heatup, a 10 degree cooldown, and again for a 5 degree heatup. The reported MTC for the cooldown was 3.76 pcm/F, which satisfies Technical Specification (TS) 3.1.1.3 requirement of less positive than 9.0 pcm/F whenever thermal power is less than 95% of rated thermal power.

Procedure PT-114, Revision 11, had been substantially revised per Interim Change I.C. #PT-114-5 in accordance with proper station administration procedures. Most revisions were interim-to-become permanent changes and were made to reflect TS changes, increase

clarity, and provide more flexibility. However, the inspector did not agree with the following revisions to PT-114:

(1) Acceptance criteria 10.5 was deleted. This criteria requires that MTC's calculated for the two heatups be within plus or minus 2.0 pcm/F of the MTC calculated for the cooldown. The purpose of this acceptance criteria is to provide additional technical validity for the reported MTC calculated during the cooldown. The MTC for the cooldown is the final reported value, and is usually more accurate because it is determined over a larger temperature change.

(2) Also deleted from the latest version of PT-114 was acceptance criteria 10.6, which required independent verification and sign-off of the test results.

(3) In addition, the inspector noticed that the procedure did not require the notification of the Nuclear Shift Supervisor if the acceptance criteria associated with TS's were not met. The failure to satisfy some of the stated acceptance criteria of PT-114 would result in a TS action statement.

The inspector discussed the above items with licensee personnel, who agreed that they should be re-incorporated into PT-114. At the exit interview, management made a commitment to revise procedure PT-114 to re-incorporate acceptance criteria 10.5 and 10.6, as well as require the notification of the Nuclear Shift Supervisor and reference to the appropriate TS if acceptance criteria were not met (IFI 302/88-10-01, Verify licensee's revision to procedure PT-114, Moderator Temperature Coefficient Determination).

d. PT-111, Hot Zero Power, All Rods Out Critical Boron Test, performed January 9, 1988

The inspector reviewed test results for the licensee's test to determine the critical RCS boron concentration at hot zero power. All necessary limitations, precautions, and prerequisites were met. The measured value of 2033 ppmB was within plus or minus 50 ppmB of the predicted value, and thus satisfied the acceptance criteria.

e. SP-102, Control Rod Drop Time Tests, performed January 7, 1988

The inspector reviewed all safety and regulating rod drop times from the fully withdrawn position and subsequent power interruption to 3/4 insertion. All drop times satisfied TS 3.1.3.4 requirement of less than or equal to 1.66 seconds.

No violations or deviations were identified.

7. Determination of Reactor Shutdown Margin - Unit 3 (61707)

The inspector reviewed surveillance procedure SP-421, Reactivity Balance Calculations, for technical adequacy, compliance with procedural requirements, and compliance with TS. The reported shutdown margin of -3300 pcm satisfies the TS requirement of -1000 pcm. The inspector verified that reactivity worth curves used in the determination of reactor shutdown margin were properly interpreted and were controlled documents.

No violations or deviations were identified.

8. Containment Local Leak Rate Program (61720)

a. Documents Reviewed

- SP-179, "Containment Leakage Test - Type "B" and "C", Revision 25"

- SP-181, "Containment Air-Lock Test, Revision 19"

b. Scope of Review

The inspector reviewed the overall local leakage rate program to verify that procedures have been developed and implemented consistent with regulatory requirements. The inspector reviewed the documents listed above for technical adequacy, for compliance with the regulatory requirements of Appendix J to 10 CFR 50, the Technical Specifications, and with applicable industry standards. The inspector also held discussions with the licensee regarding the documentation of test results, the repair and retesting following failed tests, and the relationship of these items to the "As-Found" and "As-Left" conditions of containment as applied to Containment Integrated Leak Rate Test (CILRT) results.

c. Findings

Based on the review of portions of the above procedures, the inspector concluded that the licensee had developed and implemented procedures which address the essential elements of the local leak rate test regulatory requirements. However, the inspector noted that the licensee was not leak rate testing any containment electrical penetrations and nor did the licensee's Technical Specifications require electrical penetrations to be subject to Type B leak rate tests. Paragraph II.G. of Appendix J to 10 CFR 50 require that Type B tests be performed to measure leakage across each pressure-containing or leakage limiting boundary of the primary reactor containment. The inspector expressed this concern of not leak rate testing electrical penetrations and requested information pertaining to the design and installation of all types of containment electrical penetrations found in the plant. The licensee is currently evaluating the need to local leak rate test containment eletrical

penetrations. At the exit interview, the licensee agreed to supply the requested information regarding containment electrical penetrations to the NRC for review. This matter was identified as:

IFI 302/88-10-02: Review licensee's evaluation as to the need to local leak rate test containment electrical penetrations.

The inspector reviewed the local leak rate test results summary and discussed analysis of test results and the status of repairs and retests with the licensee. The inspector was satisfied with the licensee's understanding of the application of these results to the "As-Found" and "As-Left" conditions of containment. The licensee acknowledged the application of the results to the technical specification overall leakage limits and to CILRT failure criteria. The total allowable leakage rate for type B and C tests is 248,656 SCCM (0.6 La). The results of type B and C tests conducted October through December 1987, during the last refueling outage, showed that the "As-Found" leakage exceeded the 0.6 La value. The "As-Left" value was 25,611.1 SCCM.

Witnessing of local leak rate test performance was performed by the Resident Inspector and is documented in Inspection Report 50-302/87-30.

Within the areas inspected, no violations or deviations were identified.

9. Verification of Containment Integrity (61715)

a. Records Reviewed

- SP-341, "Monthly Containment Integrity Check, Revision 20"
- SP-181, "Containment Air Lock Test, Revision 19"
- SP-300, "Operating Daily Surveillance Log, Revision 97"
- SP-347, "ECCS and Boration Systems Flow Path, Revision 35"

b. Scope of Review

The inspector reviewed all records from the above TS required surveillance procedures since the plant started up from the past refueling outage in early January 1988. Records from SP-341 were reviewed to verify the proper alignment of isolation valves and blind flanges for all primary containment penetrations per TS requirement 4.6.1.1.a. Records from SP-181 were reviewed to verify the operability of each containment air lock per TS requirement 4.6.1.3. Records from SP-300 were reviewed to verify the licensee's operation within the required containment pressure and temperature limits per

TS requirement 4.6.1.4 and 4.6.1.5. Records from SP-347 were reviewed to verify the proper alignment of each valve in the flow path of the containment spray system per TS requirement 4.6.2.1.a. The inspector also accompanied the Resident Inspector in a walkdown of the Reactor Building Spray System to verify that the component lineup was in accordance with license requirements for system operability and that the system drawing correctly reflect "as-built" plant conditions.

c. Findings

The inspector's review of the above records found no instances where required Technical Specification surveillances were not performed at the appropriate interval. Also, no concerns were identified as a result of the safety system walkdown with the Resident Inspector.

In summary, within the scope of this inspection, the inspector found reasonable assurance that the licensee was maintaining and implementing procedures to ensure containment integrity.

Within the areas inspected, no violation or deviations were identified.

Attachments:
1. Parameter List
2. RCS Leak Rate

ATTACHMENT 1

PARAMETER LIST

Unit Identification:	
Plant Name	CRYSTAL RIVER
Unit Number	3
Docket Number	50-302
Nuclear Steam System Supplier	Babcock & Wilcox
Vessel and Piping:	
Volume	10579 cubic feet
Pressurizer:	
Level Units	inches
Temperature Compensated	yes
Nominal Level	180 inches
Volume Below Nominal Level	800 cubic feet
Volume Above Nominal Level	702 cubic feet
Calibration Curve	
Slope	3.207 cubic feet per inch
Upper Level Limit	320 inches
Lower Level Limit	0 inches
Relief	Drain Tank
Makeup Tank:	
Level Units	inches
Calibration Curve	
Slope	256.6 pounds per inch
Upper Level Limit	120 inches
Lower Level Limit	0 inches
Geometric Method Available	No
Drain Tank:	
Level Units	inches
Calibration Curve	
Slope	288.5 pounds per inch
Upper Level Limit	140 inches
Lower Level Limit	75 inches
Geometric Method Available	No

ATTACHMENT 2

NRC
INDEPENDENT MEASUREMENTS PROGRAM
REACTOR COOLING SYSTEM LEAK RATES

STATION: CRYSTAL RIVER
UNIT: 3
DOCKET: 50-302

TEST DATE: MARCH 23, 1988
START TIME: 0300
DURATION: 8 hours

TEST DATA

System Parameters	Initial	Final
Pressure, psia	2157	2162
T Ave, degrees F	579.2	579.6
Water Levels		
Pressurizer, inches	204	204.4
Makeup Tank, inches	62.48	62.5
Drain Tank, inches	120.03	124.6

Water Charged = 205 gal Water Drained = 0 gal

TEST RESULTS

Change in Water Inventory in pounds:

Vessel & Piping	-274	Drain Tank (1)	1318
Pressurizer	20		
Makeup Tank (1)	5		
Less: Water Charged	1706		
Plus: Water Drained	0		
Cooling System	-1954		

Leak Rates in gpm (3):

Gross	0.49
*Identified	0.13
Unidentified	0.36

(1) Determined from tank calibration curve.
(2) Determined form tank dimensions.
(3) The density used for converting inventory change to leak rate was 62.31 pounds/cubic foot based on standard conditions
* Identified leakage compensated for 0.2 GPM standpipe flush water.