Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302

July 12, 1994
3F0794-01

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Subject: Status of Items of Interest
Following Completion of Refuel 9

Dear Sir:

Florida Power Corporation (FPC) recently completed a very successful Refuel 9 at Crystal River Unit 3 (CR-3) . This letter provides a brief discussion of outage related modifications and activities that are of mutual interest and the results of our outage goals. The status of the outage goals are included in Attachment A to this letter.

Once Through Steam Generator (OTSG) Inspection

The scope of the inspection included actions to meet the requirements of Technical Specification 5.6.2.10 and supplementary actions to disposition low signal-to noise indications. The inspection results confirmed the technical basis for the inspection strategy developed to identify all probable indications and to perform repairs as appropriate. This strategy and the preventative sleeving that was performed on 163 tubes in each OTSG provide us with a high degree of confidence of continued excellent steam generator performance and safe operation. Laboratory examinations of the four tubes that were removed will also provide valuable information to further understand the mechanisms that can degrade the tubes. A total of 20 tubes were plugged during this outage (7 in the "A" OTSG and 13 in the "B" OTSG) (reference FPC to NRC letter, 3F0594-01 dated May 25, 1994).

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A001

Unresolved Safety Issue A-46 "Seismic Qualification of Equipment

Walkdown of accessible equipment in the Reactor Building has been completed in accordance with the CR-3 Plant Specific Procedure (PSP) for Seismic Verification of Nuclear Plant Equipment (reference FPC to NRC letter, 3F0893-12, dated August 27, 1993). The three letdown coolers and two transmitters inside the Reactor Building that were not accessible will be verified by review of existing video/photos or will be declared outliers. Walkdown of other equipment outside of the Reactor Building is scheduled to begin in August. A Senior Reactor Operator has been assigned to verify the Safe Shutdown Equipment List. This will allow Operations to certify the List as required by Section 3.7 of the PSP. FPC is on schedule to submit our final report for resolution of USI A-46 by the end of 1995. Outstanding questions on our PSP submittal are being resolved in separate correspondence.

4160/480 Volt Transformer Replacement

All ten 4160/480 volt transformers (2 safety related and 8 non-safety related) have now been replaced. The replacement of the "B" Engineered Safeguards transformer was completed during the refueling outage which ended June 3, 1994 (reference FPC to NRC letter, 3F1093-02, dated October 11, 1993).

Emergency Diesel Generator Performance

Maintenance was performed on both emergency diesel generators. Planned work on EGDG-1A was more extensive and included the manufacturer's recommended (internal) maintenance. Part of this maintenance included refurbishment of the air start distributor. A cam that is a part of the distributor was inspected as part of our response to the vendor's 10 CFR 21 report and was replaced as a result. The Woodward governor was also replaced as a part of routine preventative maintenance. Limited work was planned for EGDG-1B. Part of this work included replacing the air start distributor and performing various minor preventative maintenance checks. A check of the crank strain (alignment between the engine and generator) indicated it was slightly out of tolerance. Investigation revealed the engine block thrust bearing saddle has a slight axial misalignment. A special thrust bearing was fabricated and installed to compensate for the misalignment.

The voltage dips measured during the integrated Engineered Safeguards Actuation test were significantly less than has been measured in previous tests. This is due to the maintenance performed on the machines and a change in the way the testing was done. These results are being discussed with the NRC staff.

Cable Separation

FPC began implementation of the enhanced cable separation criteria prior to Refuel 9. Good progress continues to be made toward the resolution of outstanding deviations. Several modifications to cables in the main control board, remote shutdown panel and other areas of the plant were completed during the outage. As concluded in NRC Inspection Report 50-302/94-12, work is on schedule to be completed within the time frame previously provided.

Control Rod Drop Time Testing

Control rod drop time testing was performed on the start-up from the Refuel (reference CR-3 Technical Specification Surveillance Requirement 3.1.4.3) and all control rods met the Technical Specification acceptance criteria on the first drop. During the outage, FPC removed and disassembled one control rod drive mechanism in support of a generic B&W Owner's Group investigation of factors affecting control rod drop times. FPC intends to maintain active participation in the Owners Group work in this area to assure control rod drop time Technical Specification requirements continue to be met at CR-3.

I^{132} Contamination

CR-3 experienced minor personnel skin contaminations in the Reactor Building from I^{132}. The source of the I^{132} is believed to have been from Te^{132}, a normal fission product present in the Reactor Coolant System (RCS). The process of rapid boration increased the concentration of radionuclides in the coolant, including Te^{132}, which are deposited as insoluble matter on the walls of the RCS components. As a result of pH changes in the coolant, Te^{132} and its daughter product I^{132}, could have changed states to a gaseous form and come out of the RCS by way of the steam generator vents. This allowed the concentration of I^{132} to reach detectable levels (approximately 1% of the airborne derived air concentration (DAC)) leading to minute skin contamination which was detected by sensitive personnel contamination monitors.

Intermediate Building High Energy Line Break (IB-HELB)

FPC inspected welds associated with both MSV-55 & MSV-56 in the outage (reference NRC to FPC letter, 3N0394-05, dated March 7, 1994). These inspections complete FPC's actions associated with the postulation of break locations in the CR-3 IB-HELB analysis. FPC will be submitting a separate letter which describes these inspections in more detail.

Pressurizer Missile Shields

FPC performed a 10 CFR 50.59 evaluation for the removal of the concrete missile shields located in the Reactor Building above the pressurizer. FPC determined that their removal would not constitute an unreviewed safety question nor change CR-3's licensing basis. Their removal will enhance the overall safety by improving the flow of air around the top of the pressurizer to reduce the severity of the environment in that area. There is additional safety benefit derived from eliminating the need to handle these heavy concrete objects every outage.

Rolled Plug Inspections

FPC completed all actions to assure all previously installed rolled plugs in the CR-3 OTSG's were in place and properly installed. A visual inspection was performed of all installed plugs. Two plugs were re-rolled as a precautionary measure based on the inspection results. Changes to the "ROGER" roll tool system and training of plugging operators were done just prior to performing activities during this outage.

Routine System Walkdowns

FPC previously initiated planned, routine walkdowns of plant systems for the purpose of identifying problems or potential problems. During this refueling outage, a routine system walkdown found white crystals in a localized area on the lower portion of the BSV-26 outlet weld. Dye penetrant examination revealed cracks in the piping base metal approximately 1/8" from the weld toe. Additional defects in the weld were found after attempting a weld repair. Ultimately the spool piece was replaced. Root cause determinations concluded that the piping configuration did not allow draining approximately 6 inches of water in the piping immediately above the closed BSV-26 disk. Quarterly ISI pump tests introduced water into this non-drainable area. Chronic exposure to the stagnant water resulted in intergranular attack of the 304 stainless pipe material along the lower portion of the heat affected zone of the pipe to valve weld as addressed by previous NRC correspondence. Inspection of the other train did not reveal any similar problem. ISI activities scheduled for the next refueling outage are likely to have discovered this problem had the walkdown not done so.

Intake Structure and Raw Water Pit Cleaning

Extensive cleaning was performed on the intake structure and the raw water flumes during the outage. Our goal was to attempt to clean all the way to bare concrete. Over 200 cubic yards of marine debris was removed. This extensive cleaning should substantially reduce the potential for micro-biological fouling of the raw water system.

Reactor Vessel Internal Vent Valve Replacement

In the course of performing ISI inspections, a broken locking cup on one of the two jack screws that held one of the eight reactor vessel internal vent valves (RCV-170) in position was found. The valve function was not impaired but FPC decided to replace the valve. The replacement of this valve involved detailed pre-job planning, development of special tooling, extensive mock-up training, and development of contingency plans. As a result, the replacement was very efficiently performed with substantially lower radiation dose to personnel. The broken locking cup is currently undergoing failure analysis in an effort to determine root cause of the failure. The other reactor vessel internal vent valves jacking screw locking cups were found to be satisfactory.

MUV-37 Thermal Sleeve

The augmented inservice inspection of the high pressure injection thermal sleeves identified a cracked sleeve in the downstream side of MUV-37. Formal root cause evaluation is underway. Flow induced vibration is very likely the cause. The other three sleeves were inspected and found to be free of cracks. The cracked sleeve, which had remained in place until discovery, was replaced with an improved design.

Many of these activities involved interaction with a number of NRC personnel during the outage. We appreciate the attention given by those NRC staff members. We believe our efforts were complimentary and demonstrate the effectiveness of our ability to resolve issues appropriately. Please contact Ken Wilson at (904)563-4549 if you have need for additional information.

Sincerely,

P. M. Beard, Jr.
Senior Vice President
Nuclear Operations

PMB/EEF:ff

cc: Regional Administrator, Region II
NRR Project Manager
Senior Resident Inspector

ATTACHMENT A

PERFORMANCE INDICATORS:	GOAL	ACTUAL
Commence power reduction from 100%:	March 28, 1994 at 0400 hours	March 28, 1994 at 0100 hours
Scheduled breakers open:	April 7, 1994 at 0200 hours	April 7, 1994 at 0148 hours
Scheduled breakers closed:	June 6, 1994 at 0200 hours	June 3, 1994 at 0335 hours
Scheduled return to 100% power:	June 12, 1994 at 0200 hours	June 6, 1994 at 2150 hours
Scheduled duration (breaker to breaker):	≤ 60 days	57 days 1 hour 47 minutes
Capacity duration (full power to full power):	≤ 64 Capacity days	60.5 Capacity days
Radiation dose (Man-Rem TLD):	≤ 300 Rem	253.2 Rem(PIC)
Contamination incidents:	≤ 75	105
Lost time accidents⁺:	0	0
Minor injuries⁺:	≤ 6	8
First aid⁺:	≤ 12	3
Personnel errors:	≤ 30	19
Personnel security events:	≤ 39	50
Total outage budget:	24.5 million dollars	23.517 million dollars

⁺FPC Full Time, Temporary, and Casual Employees