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

Report No.: 50-302/91-24

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: November 23, 1991 - January 3, 1992

Inspector: for R. Schin 1/28/92
P. Holmes-Ray, Senior Resident Inspector Date Signed

Inspector: L. Mellen 1/29/92
L. Mellen, Reactor Engineer, Region II Date Signed

Inspector: P. Burnett 29 Jan 92
P. Burnett, Reactor Engineer, Region II Date Signed

Inspector: for R. Schin 1/28/92
R. Freudenberger, Resident Inspector Date Signed

Approved by: K. Landis 1/29/92
K. Landis, Section Chief Date Signed
Division of Reactor Projects


SUMMARY

Scope:

This routine inspection was conducted by two resident inspectors in the areas of plant operations, security, radiological controls, Licensee Event Reports (LERs), facility modifications, and licensee action on previous inspection items. Numerous facility tours were conducted and facility operations observed. Some of these tours and observations were conducted on backshifts.

Results:

Several maintenance activities performed during the midcycle outage were noted to contribute to complicating plant operation during startup evolutions. Two violations were identified during this inspection:

50-302/91-24-01 Violation: Failure to perform channel functional tests of anticipatory reactor trips prior to entering mode 1 (paragraph 4.a).

50-302/91-24-02 Violation: Failure to implement Refueling Procedure FP-412, Canal Seal Plate Removal and Storage (paragraph 4.b).

One LER and one Inspector Followup Item (IFI) were closed:

LER 91-16: Loss of Integrated Control System Power Leads to Emergency Feedwater Initiation and Control System Actuation

IFI 302/89-10-01: Develop Upgraded Consolidated Plant Specific Technical Guidelines for Emergency Operating Procedures.

## REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *J. Alberdi, Manager, Nuclear Plant Operations
   *F. Bailey, Nuclear Electrical/Instrumentation and Control Supervisor
   *G. Becker, Manager, Site Nuclear Engineering Services
   *G. Boldt, Vice President Nuclear Production
   P. Breedlove, Nuclear Records Management Supervisor
   *J. Buckner, Nuclear Regulatory Specialist
   *J. Elam, Nuclear Technical Instructor
   *G. Flakes, Nuclear Engineer II
   *E. Froats, Manager, Nuclear Compliance
   *R. Fuller, Senior Nuclear Licensing Engineer
   *G. Halnon, Manager, Nuclear Plant Systems Engineering
   *B. Hickle, Director, Quality Programs
   *M. Jacobs, Area Public Information Coordinator
   *G. Longhouser, Nuclear Security Superintendent
   *P. McKee, Director, Nuclear Plant Operations
   *R. Murgatroyd, Assistant Nuclear Maintenance Superintendent
   *W. Nielsen, Assistant Nuclear Maintenance Superintendent
   *D. Porter, Nuclear Operations Superintendent
   *S. Powell, Senior Nuclear Electrical Instrument and Control Supervisor
   *A. Riley, Documents Clerk
   *J. Roberts, Assistant Nuclear Chemistry and Radiation Protection Superintendent
   *W. Rossfeld, Manager, Site Nuclear Services
   *D. Shook, Nuclear Engineering Supervisor
   *G. Vaughn, Nuclear Projects Specialist
   *M. Williams, Nuclear Regulatory Specialist

   Other licensee employees contacted included office, operations, engineering, maintenance, chemistry/radiation, and corporate personnel.

   NRC Resident Inspectors

   *P. Holmes-Ray, Senior Resident Inspector
   *R. Freudenberger, Resident Inspector
   L. Mellen, Reactor Engineer, RII
   P. Burnett, Reactor Engineer, RII

   *Attended exit interview

   Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Plant Status and Activities

   During this inspection period three reactor trips occurred. The reactor tripped on November 25, 1991, due to loss of both feed pumps and was returned to service on November 27, 1991. On December 3, 1991, a shutdown was in progress to investigate high reactor cavity temperatures and a failed power range nuclear instrument when a reactor trip from 48% power occurred due to a feed transient induced while adjusting the nuclear overpower trip setpoint. The reactor was re-started on December 8, 1991, and a reactor trip occurred three hours later caused by low reactor coolant pressure when the pressurizer spray valve stuck open. On December 17, 1991, the unit was returned to full power and operated at full power to the end of the inspection period.

A special inspection was conducted from December 8 through December 23, to review the transient that occurred on December 8. The results of that inspection are documented in NRC Inspection Report 50-302/91-25.

3. Plant Operations (71707, 93702, & 40500)

Throughout the inspection period, facility tours were conducted to observe operations and maintenance activities in progress. The tours included entries into the protected areas and the radiologically controlled areas of the plant. During these inspections, discussions were held with operators, health physics and instrument and controls technicians, mechanics, security personnel, engineers, supervisors, and plant management. Some operations and maintenance activity observations were conducted during backshifts. Licensee meetings were attended by the inspector to observe planning and management activities. The inspections confirmed FPC's compliance with 10 CFR, Technical Specifications, License Conditions, and Administrative Procedures.

a. Reactor Trip due to Feedwater System Transient

On November 25, 1991, at 5:16 p.m., the unit was synchronized with the grid when the output breaker was closed. At about 5:20 p.m., with reactor power at approximately 20% of full power, a reactor trip occurred when the operating main feedwater pump tripped. The cause of the main feedwater pump trip was a spurious low Deaerating Feed Tank level signal. The spurious signal was the result of a failure of the level indicator gaskets. This depressurized the level legs and created an indicated low DFT level when actual level was high. The high level was caused by the combination of the operating condensate pump overfeed of the DFT and the failure of the DFT dump valve to operate. Upon investigation of the failure of the dump valve, its breaker was found open. The breaker was closed and functioned properly. The Senior Resident Inspector was in the control room at the time of the trip and he observed operator response. The operators responded to this event in a prompt, proper, and professional manner. Licensee planned corrective actions as a result of this transient were outlined in LER 91-014-00, dated December 12, 1991. Since all corrective actions were not complete at the end of the report period, the inspector will review the adequacy of the corrective actions in a future inspection, in accordance with the NRC inspection program.

b. Reactor Trip due to Inadvertent Selection of Failed Reactor Power Channel for Control

On December 3, 1991, CR-3 was being shutdown from 100% reactor power to investigate the failure of Power Range Nuclear Instrumentation Channel NI-8 and the location of the reactor vessel refueling canal seal plate. At 50%, power was stabilized to adjust the RPS Nuclear Overpower Trip Setpoint down to 64.5% in accordance with TS 3.2.4 action required when steady state QPT limits were exceeded. Power Range detector NI-8 had failed to zero indicated power on November 30, 1991, tripping RPS Channel "D".

Prior to adjusting the trip setpoint, a pre-job meeting was conducted. The effect of the NI-8 failure on the performance of SP-113, "Power Range Nuclear Instrumentation Calibration" on ICS and RPS inputs was discussed. The I&C technician, in accordance with SP-113, placed "A" channel of RPS to "bypass" then placed the Power Range Test Module in "Test/Operate". This caused an immediate plant transient due to an erroneous reactor power signal being supplied to the ICS. Feedwater to the steam generators was reduced and control rods withdrawn by automatic functions of the ICS. The operators took manual control of the ICS and manually opened the pressurizer spray valve. However, a reactor trip occurred within one minute due to high reactor coolant system pressure.

The ICS is fed a reactor power signal from the four Power Range Nuclear Instruments. The signals from NI-5 and NI-6 are averaged as are the signals from NI-7 and NI-8. The higher of the two averages is then selected by a high auctioneer and fed to the ICS as reactor power. When a channel is placed in test, the output from that averager is grounded (0% power signal). The high auctioneer should then elect the output from the other averager which would indicate a greater than zero power level. NI-5 and NI-6 were both indicating approximately 50% power. The average was, therefore, approximately 50%. NI-7 was indicating 0% power but when averaged with the failed NI-8, indicating 0 power, the average of these two channels was now 25%. When the "A" RPS Power Range Test Module was placed in "Test/Operate", The output from that averager was grounded. The auctioneer took the output from the average of NI-7 and NI-8 and sent this to the ICS for control. In this configuration, the ICS saw the reactor power signal drop from 50% to 25% and reacted approximately (ICS withdrew control rods to match indicated power to demand and reduced main feedwater flow to match indicated reactor power).

The cause of this event was a misunderstanding of the processing of power range nuclear instrumentation signals to the ICS and the effects of channel test switches. The response of the plant was correct for the switch positions that occurred. A major contributor to this reactor trip was inadequate and inaccurate operator training. If the ICS feedwater control and reactor control had been placed in "Hand" (Manual) prior to placing the Power Range Test Module to "Test/Operate," no transient would have occurred. Also, SP-113 did not include adequate information regarding performance of the procedure with one Power Range NI channel failed. On December 5, 1991, a change was incorporated into SP-113 to place ICS Feedwater Demand and the Rod Control Station in "Hand" if any Power Range NI channel is not operable. Licensee planned corrective actions as a result of this transient were outlined in LER 91-017-00, dated January 6, 1992. Since all corrective actions were not complete at the end of the report period, the inspector will review the adequacy of the corrective actions in a future inspection, in accordance with the NRC inspection program.

c. Reactor Trip due to Pressurizer Spray Valve Failed Open

On December 8, 1991, the plant was being returned to power operation. Power was being slowly raised to about 15% to bring the Turbine-Generator on-line. At about 2:47 a.m., power was increased from about 11% to 12%. At about 2:49 a.m., power was again increased, this time from about 12% to 14% During both of these power increases, RCS pressure increased sufficiently for the Pressurizer Spray Valve (RCV-14) to open. RCV-14 then failed to close, and RCS pressure decreased over the next 18 minutes when the low RCS pressure trip setpoint was reached. One of the actions taken by the operator during this time was to manually close RCV-14, although it never indicated that it had opened. The reactor automatically tripped on low RCS pressure.

Following the reactor trip, RCS pressure did not recover as expected, but continued to decrease. The operators bypassed automatic ES actuation after the "ES not bypassed" alarm was received at 1640 psig. Six minutes later, when the alarm indicating two of the three ES channels' bistables for low RCS pressure had tripped, the operator came out of the ES bypass, and a full ES actuation occurred. Both Emergency Diesel Generators started automatically but were not needed because the ES busses remained powered. Also, both EFW pumps automatically started, but were later shut down when MFW flow was verified. All ES loads block loaded onto the ES busses correctly.

Full HPI flow to the RCS occurred for about one minute and throttled flow occurred for another minute or two, at which time HPI was stopped as RCS pressure increased. The ES bistables were reset. After HPI was stopped, RCS pressure again began to decrease and about 10 minutes later one of the ES bistables tripped. The operators again bypassed the ES actuation signal. During the time the decision was being made to increase make-up flow to begin filling the Pressurizer (about 10 minutes), the other two ES bistables tripped.

The operators opened MUV-24 (high pressure injection valve) and began filling the Pressurizer, but did not take the RCS solid. After RCS pressure had increased to about 1700 psig, MUV-24 was closed. At about this time, it was decided to close the Pressurizer Spray Block Valve (RCV-13), at which time RCS pressure began to quickly recover. Stable RCS conditions were achieved (RCS pressure at 2155 psig, RCS temperature at 537 F, and Pressurizer level at 100 inches) terminating the event.

This transient was reviewed in detail by the inspectors as documented in NRC Inspection Report 50-302/91-25.

d. Excessive Core Quadrant Power Tilt at Low Power

During the startup on November 24, 1991, the core quadrant power tilt at 15% power exceeded the TS 3.2.4 steady state limit of 4.25% prescribed on page 20 of the COLR. TS 3.2.4 requires that the quadrant power tilt shall be maintained less than the steady state limits specified in the COLR when the reactor is above 15% of rated thermal power. The tilt was believed to be the result of earlier operation with one control rod fully inserted for a period of seven days at about 60% power. It was expected that the tilt would burn out with extended full power operation and that the magnitude and duration of the quadrant power tilt would be within the limits specified in the TS LCO Action Statement. However, TS 3.0.4 prohibits entry into an operational mode or other specified condition in the TSs unless the conditions of the LCO are met without reliance on the provisions contained in the Action Statements.

The licensee verified that an uncoupled or otherwise mis-operating control rod did not exist and requested a temporary waiver of compliance. Region II issued a one-time waiver of compliance exempting the requirements of TS 3.0.4 as it applied to LCO 3.2.4, "Quadrant Power Tilt." Subsequent operation did reduce the tilt to 2% at full power. Before the tilt was reduced to the expected value of 1.4%, the reactor was shut down to replace the failed NI-8 on December 2, 1991. To cover subsequent startups after that date, NRR issued another similar waiver of compliance on December 6, 1991. The inspectors verified that the temporary waivers were properly implemented during plant startups conducted on November 27, December 8, and December 19. They verified that the overpower trip setpoints were reduced to 65.5 percent during those plant startups, as required by the TS LCO Action Statements. On December 16, the NRC issued a license amendment to permanently incorporate a TS change to exempt LCO 3.2.4 from the provisions of TS 3.0.4. The Temporary Waivers of Compliance dated November 26 and December 6, 1991 are closed.

Further investigation by licensee and NSSS vendor engineers revealed that much of the excessive tilt was caused by instrumentation. The SPNDs were confirmed to be putting out the expected, noise-free signals. Those signals were of the order of five to twenty nanoamperes at low power. At full power, SPND outputs range from 200 to 500 nanoamperes. However, the multiplexers, which successively read the SPND outputs into the plant computer for analysis, were shown to be adding as much as ±10 nanoamperes to each signal, independent of power. The multiplexers were calibrated pursuant to SP-140, Incore Neutron Detector System Calibration, every 18 months, as required by TS 4.3.3.2(b). The acceptance criterion was output current equal input current ±25 nanoamperes, where the input current was 500 nanoamperes. Since that error is constant, rather than scaled, a tolerable error of 5% at full power becomes unacceptable for low-power determination of QPTR. B&W now recommends a ±5 nanoampere acceptance criterion for SP-140, and the licensee intends to change the procedure accordingly.

On December 14 and 15, 1991, licensee and vendor personnel adjusted the multiplexers to reduce the offsets to ±4 nanoamperes. No better adjustment could be obtained. During the subsequent startup and power escalation on December 17 - 19, 1991, no power tilt problems were observed.

The licensee plans to issue a formal followup report of multiplexer problems and corrective actions in January 1992.

Correction of another source of power tilt was observed in the control room on December 19, 1991. At approximately 12:00 p.m., the actual value of delta T-cold, the temperature difference between RCS water leaving the OTSGs, was reduced to 0.0. This was done by making a routine operator adjustment to the ICS controller. Correspondingly, the difference between OTSG levels reduced from about 8% of span to about 6% of span. The quadrant flux tilts, which had been acceptable with the greatest slightly over 1.1, reduced to a maximum of 0.78.

e. Radiological Protection Program

Radiation protection control activities were observed to verify that these activities were in conformance with the facility policies and procedures, and in compliance with regulatory requirements. These observations included:

- Entry to and exit from contaminated areas, including step-off pad conditions and disposal of contaminated clothing;
- Area postings and controls;
- Work activity within radiation, high radiation, and contaminated areas;
- RCA exiting practices; and
- Proper wearing of personnel monitoring equipment, protective clothing, and respiratory equipment.

The implementation of radiological controls observed during this inspection period were proper and conservative.

f. Security Control

In the course of the monthly activities, the inspector included a review of the licensee's physical security program. The performance of various shifts of the security force was observed in the conduct of daily activities to include: protected and vital area access controls; searching of personnel, packages, and vehicles; badge issuance and retrieval; escorting of visitors; patrols; and compensatory posts. In addition, the inspector observed the operational status of protected area lighting, protected and vital area barrier integrity, and the security organization interface with operations and maintenance. No performance discrepancies were identified by the inspectors.

4. Maintenance and Surveillance Activities (62703 & 61726)

Surveillance tests were observed to verify that approved procedures were being used; qualified personnel were conducting the tests; tests were adequate to verify equipment operability; calibrated equipment was utilized; and TS requirements appropriately implemented.

The following tests were observed and/or data reviewed:

- SP-110, Reactor Protection System Functional Testing; and

- SP-119, Feedwater Block Valve Functional Testing.

In addition, the inspector observed maintenance activities to verify that correct equipment clearances were in effect; work requests and fire prevention work permits, as required, were issued and being followed; quality control personnel performed inspection activities as required; and TS requirements were being followed.

Maintenance was observed and work packages were reviewed for the following maintenance activities:

- WR 0291273, 0291484, 0291517, and 0291543 Troubleshooting and repair of Nuclear Instrumentation;

- WR 0287111 Troubleshooting and repair of the Reactor Cavity Cooling Fan System;

- WR 0286116 Repositioning of the Refueling Cavity Seal Plate in support of other outage activities; and

- WR 0290801 Response Time Testing of Feedwater Valve FWV-28.

The following items were considered noteworthy.

a. Reactor Protection System Surveillance not Completed Prior to Mode Change

On November 24, an operating mode change from mode 3, Hot Standby, to mode 1, Power Operation, was performed. The shift supervisor believed that the monthly RPS surveillances for the anticipatory reactor trips on loss of main feedwater or main turbine trip were up to date. However, these surveillances had not been performed on schedule. At 5:00 p.m., the plant entered mode 1. At 5:20 p.m. the following day, November 25, the plant tripped from approximately 20% power due to a feedwater system transient (see detail 3.a, above). The loss of both main feedwater pumps anticipatory reactor trip functioned properly during this transient.

The anticipatory trips are so named because they serve to trip the reactor in anticipation that plant conditions have degraded to a point that one of the primary reactor trips will occur. The loss of main feedwater pumps trip senses main feedwater pump status based on the feedwater pump control oil pressure. A reactor trip occurs when both main feedwater pumps are tripped and reactor power is greater than 20%. The main turbine trip senses turbine status based on main turbine automatic stop oil pressure. A reactor trip occurs when the main turbine is tripped and reactor power is greater than 45% power.

TS 4.3.1.1 requires each reactor protection system instrumentation channel to be demonstrated operable by the performance of a channel functional test during the modes and at the frequencies shown in Table 4.3-1. Table 4.3-1 requires the anticipatory reactor trips on main turbine trip and loss of both main feedwater pumps to be operable in mode 1, power operation. TS 4.0.4 requires that entry into an operational mode shall not be made unless the surveillance requirements associated with the Limiting Condition for Operation has been performed within the stated surveillance interval. The plant operated in mode 1 for slightly over twenty-four hours in violation of TS 4.3.1.1.1 and 4.0.4. Violation (50-302/91-24-01): Failure to perform channel functional tests of anticipatory reactor trips prior to entering mode 1.

This violation was identified by the licensee on November 26, at approximately 11:00 a.m., when an Instrument and Controls supervisor discovered the anticipatory reactor trip testing was not up to date. Surveillance Procedure SP-110 "Reactor Protection System Functional Testing" was performed on November 8, 1991, however the anticipatory reactor trips sections of the procedure were not completed at that time. The anticipatory reactor trips were tested satisfactorily prior to further attempts at escalating to mode 1. "As found" values were within specification. The licensee reported the violation in Licensee Event Report 91-015-00, dated December 12, 1991. In the report, the licensee stated that the event was caused by personnel error and that investigation of corrective actions was continuing. A supplemental report was to be provided by February 28, 1992.

The significance of the failure to perform the anticipatory trip testing prior to mode change was minimal due to the following considerations:

- the followup testing "as found" conditions of all of the channels were within specification;

- the anticipatory reactor trip due to loss of both main feedwater pumps functioned properly on November 25; and

- the channel functional tests of the primary reactor trips were completed and these channels were available to trip the reactor.

Although this violation was identified and reported by the licensee, it is being cited because it was indicative of a potential weakness in the verification of plant readiness for restart following an outage and corrective actions to prevent recurrence were not developed at the end of the inspection period.

b. Reactor Cavity Elevated Temperatures

On November 30, 1991, with the unit at full power, channel "D" of the reactor protection system tripped on nuclear overpower based on flow and imbalance signals. The cause of this channel trip was the failure of power range nuclear instrument NI-8. The licensee's initial investigation into the failure of NI-8 identified that the operating reactor cavity cooling fan had been swapped from AHF-2B to AHF-2A approximately twenty-one hours prior to the failure of NI-8. Temperature measurements obtained from thermocouples located in the reactor cavity ranged from 305 F to 499 F with AHF-2A running and from 154 F to 487 F with AHF-2B running. Also, NI-4, the intermediate range nuclear instrument located in the same core quadrant as NI-8, had been replaced due to inaccurate indication during a recent reactor startup. Licensee personnel who had made repairs to NI-4 indicated that the canal seal plate, which seals the reactor vessel flange to the refueling cavity floor for refueling, was in the "sealed" position. During normal operation the canal seal plate should be in a "stored" location with blocks elevating the seal off the seating surfaces.

The reactor cavity is cooled by a ventilation system which consists of two full capacity fans (AHF-2A and AHF-2B), filters, and cooling coils. One of the two fans is normally operated. It recirculates air inside the reactor building through a filter and cooling coil to the reactor cavity. The discharge ductwork at each fan is equipped with a motor operated damper to allow common distribution ductwork to be utilized with either fan running. Damper positioning occurs automatically with the operating fan damper open and the shutdown fan damper closed to prevent backflow. The cooled air is supplied into the reactor cavity below the reactor vessel. The air then flows upward between the reactor vessel and shield wall and out through the reactor coolant and core flood penetrations through the shield wall. A portion of the air flow also continues upward through a channel between the reactor vessel and a shield plug located below the elevation of the canal seal plate. The licensee stated that this flow aids in cooling the nearby connection boxes associated with the nuclear power instrumentation. With the canal seal plate in the "sealed" position, the cooling flow through this area would be blocked.

On December 2, a plant shutdown was initiated to allow investigation of the cause of the high temperature in the reactor cavity and the failure of power range nuclear instrument NI-8. The licensee's investigation identified two causes that contributed to the elevated temperatures in the reactor cavity. The canal seal plate was in the "sealed" position and the discharge damper associated with AHF-2A was mispositioned.

The canal seal plate in the "sealed" position resulted in blocking the cooling air flow through the uppermost portion of the reactor cavity, contributing to the elevated temperatures in this area. The canal seal plate was left in the sealed position following movement during the mid-cycle 8 refueling outage. The seal plate was moved to prevent damage during outage activities and to allow inspection of the sealing surfaces. This work was performed in accordance with procedure FP-412 "Canal Seal Plate Removal and Storage" which was implemented under WR 0286116. The work was performed by B&W Nuclear under a long term maintenance contract with FPC. Section 4.1 of FP-412 describes proper storage of the canal seal plate. The work request was closed with as left conditions described as "Seal Plate Stored as per FP-412" when the seal plate was actually in the "sealed" position. The inspector considered this example to be indicative of a weakness in work control and a violation of TS 6.8.1.6. Violation (50-302/91-24-02): Failure to implement Refueling Procedure FP-412, Canal Seal Plate Removal and Storage.

The discharge damper associated with reactor cavity cooling fan AHF-2A was mispositioned during maintenance performed on the reactor cavity cooling system during the midcycle 8 maintenance outage. The licensee's review of the cause for the mispositioning of the fan damper revealed that in August of 1991, WR 287111 was initiated to troubleshoot the cause of improper cycling of the reactor cavity cooling fans discharge dampers during the transfer of the operating fan. The post maintenance test developed during the planning stage was intended to verify that the damper cycling problem was corrected. During the midcycle 8 maintenance outage, the control wiring associated with the damper operators and the differential pressure switches were inspected. A differential pressure switch had a broken lead and the damper operators had several loose connections. These conditions were repaired and the differential pressure switch was calibrated. The system was tested and the damper cycling problem still existed. It was then noticed that the scribe marks on the AHF-2A discharge damper was opposite of standard practice and the AHF-2B discharge damper. Based on this observation, it was decided that the AHF-2A damper was mispositioned. It was repositioned as added work under the same work request. During post maintenance testing in accordance with the original post maintenance test instructions, the test verified that the damper positioner operated properly. It did not verify that the AHF-2A discharge damper was properly positioned and proper air flow was being supplied to the reactor cavity with either fan running.

After the plant was shut down to investigate the high reactor cavity temperature on December 2, the AHF-2A discharge damper was verified to be mispositioned: when it indicated open it was actually fully closed. This resulted in no cooling air flow to the reactor cavity with AHF-2A running and reduced cooling air flow to the reactor cavity with AHF-2B running due to backflow through AHF-2A.

Since the original work scope did not anticipate repositioning of the fan dampers and the post maintenance test was not revised as a result of the change in work scope, proper operation of the reactor cavity cooling system was not verified prior to returning the system to service. This indicated a weakness in the post maintenance test program in that a revision of the scope of work performed under WR 287111 occurred without a corresponding revision of the post maintenance test.

c. Summary

Several maintenance activities performed during the midcycle outage were noted to contribute to complicating plant operation during startup evolutions. The improper storage of the canal seal plate, the mispositioning of the reactor cavity cooling fan discharge damper, the improper alignment of the power supply associated with the deaerator feed tank dump valve, and the failure of the pressurizer spray valve are examples of equipment that was worked on during the mid-cycle outage that was returned to service in a degraded condition. The inspectors plan to be sensitive to the restoration of plant equipment to service and the adequacy of post maintenance testing in future inspections.

5. Review of Licensee Event Reports (92700)

LERs were reviewed for potential generic impact, to detect trends, and to determine whether corrective actions appeared appropriate. Events that were reported immediately were reviewed as they occurred to determine if the TS were satisfied. LERs were also reviewed in accordance with the current NRC Enforcement Policy.

a. (Closed) LER 91-16: Loss of Integrated Control System Power Leads to Emergency Feedwater Initiation and Control System Actuation

On November 25, the unit was operating at 13% rated thermal power. Reactor power was being maintained less than 15% power because of a quadrant power tilt in excess of the steady state limit of TS 3.2.4. This TS is applicable at greater than or equal to 15% power. The QPT value was approximately 11% in the WX core quadrant. This situation was being evaluated by the NRC staff at the time of the event. FPC was awaiting an NRC Waiver of Compliance exempting TS 3.2.4 from the requirements of TS 3.0.4 thereby allowing power ascension.

Also at this time, utility Instrumentation and Control Technicians were working on a modification to the ICS. New control wiring was being installed to support logic module replacements. As the technician was installing a cover plate over the control wires, his hand slipped. His hand contacted the wiring connector posts, pushing a plus 24 volt DC post into a ground post. This shorted the plus 24 volt DC power supply and tripped the AC breakers feeding power to the ICS power supply modules.

Annunciators immediately indicated the loss of ICS power. The components which receive control signals from ICS responded as expected. The Main Feedwater pumps decreased to minimum speed, the turbine bypass valves failed closed, and the Feedwater Start-Up Control Valves failed to 50% open. This reduced the feedwater flow to the steam generators and stopped steam flow from them, resulting in a reactor coolant system heat up. In response to the resultant Reactor Coolant Pressure increase, the operators opened the pressurizer spray valve to control pressure.

The I&C Technician realized immediately what had happened. He separated the posts and checked that the ICS power supply system had not been damaged. He next reported the events to the control room operators. The operators instructed him to close the ICS power supply breakers and he did so. ICS power was off for a total of 33 seconds.

When ICS power is restored, the ICS control stations are in the manual mode with either a minimum or maximum demand signal. Per design the Main Feedwater pumps restore at maximum demand. This caused the pumps to begin increasing speed which created a small steam generator overfeed situation.

The slight feedwater overfeed had a reactivity feedback. The overfeed lowered the reactor coolant inlet temperature which produced a positive reactivity change and raised power to approximately 19%. This put the plant within the applicability of TS 3.2.4.

To control feedwater, the operator had to manually operate the main feedwater pump and two startup feedwater control valves. During this time, the feedwater flow was reduced to the point that the "B" steam generator reached the low level setpoint that initiates EFIC. The operators returned the unit power to less than 15% as required by TS. After assuring that they had control of feedwater, they secured the Emergency Feedwater Pumps and reset the EFIC system.

The licensee considered the initiator of this event, the I&C Technician's hand slipping, to be an isolated event. His actions in response were prompt and correct. All plant systems performed as expected, therefore no further actions were planned. The inspector's review of the event determined that the licensee's conclusions and actions were appropriate. LER 91-16 is closed.

6. Licensee Action on Previously Identified Inspection Findings (92702 & 92701)

(Closed) IFI 302/89-10-01 - Develop an Upgraded Consolidated PSTG. The inspectors discussed this item with the licensee. While this item is not complete, there is evidence of satisfactory progress. This item is closed.

7. Exit Interview (30703)

The inspection scope and findings were summarized on January 6, 1992, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

| Item Number | Description and Reference |
|---|---|
| 50-302/91-24-01 | VIO - Failure to perform channel functional tests of anticipatory reactor trips prior to entering mode 1 (paragraph 4.a). |
| 50-302/91-24-02 | VIO - Failure to implement Refueling Procedure FP-412, Canal Seal Plate Removal and Storage Paragraph 4.b). |

8. Acronyms and Abbreviations

AC - Alternating Current
AHF - Air Handling Fan
a.m. - ante meridiem
B&W - Babcock & Wilcox
CFR - Code of Federal Regulations
COLR - Core Operating Limits Report
CR - Crystal River
DC - Direct Current
DFT - Deaerating Feed Tank
EFIC - Emergency Feedwater Initiation and Control System
EFW - Emergency Feedwater
ES - Engineered Safeguards
F - Fahrenheit
FP - Refueling Procedure
FPC - Florida Power Corp.
FWV - Feedwater Valve
HPI - High Pressure Injection
I&C - Instrumentation and Control
ICS - Integrated Control System
IFI - Inspector Followup Item
LER - Licensee Event Report
MFW - Main Feedwater
MUV - Make-Up Valve
NI - Nuclear Instrumentation
NRC - Nuclear Regulatory Commission
NRR - Office of Nuclear Reactor Regulation
NSSS - Nuclear Steam Supply System
OTSG - Once Through Steam Generator
p.m. - post meridiem
psig - pounds per square inch gauge
PSTG - Plant Specific Technical Guidelines (for Emergency Operating Procedures)
QPT - Quadrant Power Tilt
QPTR - Quadrant Power Tilt Range
RCS - Reactor Coolant System
RCV - Reactor Coolant Valve
RPS - Reactor Power System
SP - Surveillance Procedure
SPND - Self-Powered Neutron Detector

TS - Technical Specification
VIO - Violation (of NRC Requirements)
WR - Work Request