



UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
230 PEACHTREE STREET, N.W. SUITE 818
ATLANTA, GEORGIA 30303

IE Inspection Report No. 50-302/76-18

Licensee: Florida Power Corporation
3201 34th Street, South
P. O. Box 14042
St. Petersburg, Florida 33733

Facility Name: Crystal River 3
Docket No.: 50-302
License No.: CPPR-51
Category: B1

Location: Crystal River, Florida

Type of License: B&W, PWR, 2452, Mwt

Type of Inspection: Routine, Unannounced

Dates of Inspection: October 5-8, 1976

Dates of Previous Inspection: September 28 - October 1, 1976

Principal Inspector: F. Jape, Reactor Inspector
Reactor Projects Section No. 2
Reactor Operations and Nuclear
Support Branch

Accompanying Inspectors: J. D. Martin, Reactor Inspector
Nuclear Support Section
Reactor Operations and Nuclear
Support Branch

W. B. Swan, Reactor Inspector
Engineering Support Section No. 1
Reactor Construction and Engineering
Support Branch
(October 6-8, 1976)

Other Accompanying Personnel: None

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Principal Inspector: Frank Jape 10/26/76
F. Jape, Reactor Inspector
Reactor Projects Section No. 2
Reactor Operations and Nuclear
Support Branch
Date

Reviewed by: R. C. Lewis 11/2/76
R. C. Lewis, Chief
Reactor Projects Section No. 2
Reactor Operations and Nuclear
Support Branch
Date

SUMMARY OF FINDINGS

I. Enforcement Matters

None

II. Licensee Action on Previously Identified Enforcement Matters

Failure to Maintain Instrument Calibration Labels
Up-to-Date (IE Report 50-302/76-11)

Licensee's letters, dated August 9 and September 17, 1976, have been reviewed and commitments were verified as complete. Item is closed. (Details I, paragraph 2)

III. New Unresolved Items

None

IV. Status of Previously Identified Unresolved Items

75/8-2 Safeguards Systems Pump Runout

Licensee has installed flow control in place of limit switches to control pump runout. Testing of the modifications is underway. Item remains open. (Details I, paragraph 3)

76-9/1 Diesel Generator Concrete Panels

Licensee has completed the required modifications to electrical equipment enclosures. Item is closed. (Details I, paragraph 4)

V. Unusual Occurrences

None

VI. Other Significant Findings

Followup on IEB's and IEC's

1. IEB 75-3 "Incorrect Lower Disc Spring and Clearance Dimension in Series 8300 and 8302 ASCO Solenoid Valves"

Licensee's response, dated August 9, 1976, reported that ASCO valves of the type discussed in the IEB, are in use at CR-3.

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Repair kits have been received and repairs are underway. The repairs consist of installing a new metal-to-metal seat and verifying proper lower disc stem clearance. The inspector observed the work being done and examined the repair records. Both were found satisfactory. Item is closed.

2. IEB 76-02, "Relay Coil Failures - GE Type HFA, HGA, HKA and HMA Relays"

Licensee's action on this IEB was reported in IE Report 50-302/76-11. Followup revealed that the replacement of nylon coil spools with hexan has been completed. No further effort is planned for this bulletin.

3. IEC 76-01, Crane Hoist Control - Circuit Modifications

Licensee's report, dated September 29, 1976, reported that no modifications have been made to the spent fuel cast crane controls. Interviews with operations and maintenance personnel revealed that the crane has performed satisfactorily to date. No further effort is planned for this circular.

4. IEC 76-02, "Relay Failures - Westinghouse BF (ac) and BFD (dc) Relays"

Licensee's report, dated September 3, 1976, stated that these type relays are not in use at CR-3. Therefore, no further effort is planned for this circular.

VII. Management Interview

A management interview was held on October 8, 1976, with J. Alberdi, and members of his staff. The inspection findings were discussed. In particular, the status of the facility in relation to the plans for loading fuel by November 1, 1976 was discussed.

Repair status of the reactor building dome was also discussed. Licensee management stated that repairs were slightly ahead of schedule.

DETAILS I

Prepared by:

Frank Jape
 F. Jape, Reactor Inspector
 Reactor Projects Section No. 2
 Reactor Operations and Nuclear Support
 Branch

10/26/76
 Date

Dates of Inspection: October 5-8, 1976

Reviewed by:

R. C. Lewis
 R. C. Lewis, Chief
 Reactor Projects Section No. 2
 Reactor Operations and Nuclear Support
 Branch

11/2/76
 Date

1. Individuals Contacted

J. Alberdi - Project Manager
 G. P. Beatty, Jr. - Nuclear Plant Superintendent
 J. C. Hobbs, Jr. - Manager Generation Testing
 E. E. Froats - Manager Site Surveillance
 W. E. Kemper - Assistant Nuclear Operator
 F. W. Pluebell - Electrical Supervisor
 J. E. Barrett - Plant Engineer
 T. C. Lutkehaus - Maintenance Engineer
 Instrument Technician
 Nuclear Electrician

2. Instrument Calibration

Additional response, dated September 17, 1976, was received from the licensee regarding the noncompliance item which involved out-of-date calibration labels on safety-related instrumentation. Followup on the licensee's commitments revealed that the commitments have been completed. These are summarized below:

- a. QP 14.11, "Use of Status Indicators," was revised on September 17, 1976, to delete reference to permanently installed instrumentation and controls.
- b. Inter-office memo, dated September 9, 1976, was issued to the plant staff stating that the old, outdated stickers are to be disregarded. In their place, current calibration records are being maintained by the computer and controls engineer.

The inspector had no further comments on this matter. Item is closed.

3. Safeguards System Pump Runout

The status of the problem of safeguards system pump runout was reviewed. This concern has been carried as unresolved item 75-8/2 and remains open pending completion of associated testing and acceptance of test results.

The items remaining to be resolved are summarized below:

a. High Pressure Injection System

The throttle setting for MOV-2, -6, and -10 needs to be determined and locking devices installed.

b. Low Pressure Injection System

Testing of the newly installed differential pressure flow control method for DHV-110 and -111 to control pump runout.

Followup on these items will be conducted on future inspections.

4. Diesel Generator and Associated Electrical Cabinets

During inspection 50-302/76-9, unresolved item 76-9/1 was identified. This item dealt with the possible functional loss of the diesel generator Class IE cabinets due to fire water spray entering the ventilation opening located on top of the cabinets.

The licensee has conducted a field inspection and engineering evaluation of all Class IE electrical equipment enclosure to ensure that they are adequately sealed or protected against water entry from overhead spray.

The physical work of sealing all identified openings has been completed. The inspector examined the sealing methods and documentation and had no further comments. Unresolved Item 76-9/1 is closed.

5. Bent Valve Stems in Safety-Related Systems

Followup on this item, which was reported by the licensee on January 9, 1976, was completed during this inspection. Through examination of work records and stores inventory records, the

inspector verified that bent valve stems have been replaced and new stems are in stock at the site, for presently undamaged, original stems that may exhibit a tendency to bend.

The inspector also examined several valves with new stems and witnessed their operation. There were no comments regarding this item. The matter is closed.

6. Hydraulic Snubber Sight Gauge Leakage

A design deficiency involving leaky hydraulic snubber sight gauges was reported by the licensee on June 21, 1976. Followup on this problem revealed that the licensee has begun replacement of the gauges with a newly designed gauge. Completion date for the changeover has not been determined. Snubbers located in inaccessible areas during reactor operation have been given priority on the replacement schedule to avoid delay in the scheduled plant startup.

Followup on this problem will be conducted on future inspections.

7. Reactor Protection System

During inspection 50-302/76-13, a comment was raised concerning the Nuclear Overpower based on Reactor Coolant flow and Axial Power Imbalance trip in the reactor protection system. Followup on this item was conducted and the comment is closed.

The comment concerned proper adjustment of the trip setpoint if the actual RC flow was determined to be greater than design after fuel loading. With RC flow greater than design, and no adjustment of the setpoint, the limit, as specified in Technical Specification 2.2 would be exceeded.

Discussion with licensee personnel revealed that adjustment of the trip setpoint is planned as part of the power ascension testing program. A test to determine reactor coolant flow scaling factors is to be run at 15%, 40%, 75% and 100% power, with setpoint adjustments being made to remain within the technical specification.

8. Plant Tour

A tour of the reactor building, portions of the auxiliary building and the spent fuel area was conducted. During the tour the inspector noted extensive housekeeping activity in the reactor building. The condition of the auxiliary building varied with the amount of on-going construction activity. The spent fuel area was found to be clean and access to the area is controlled.

DETAILS II

Prepared by: J. B. Martin
J. B. Martin, Reactor Inspector
Nuclear Support Section
Reactor Operations and Nuclear
Support Branch

11/3/76
/Date

Dates of Inspection: October 5-8, 1976

Reviewed by: H. C. Dance
H. C. Dance, Acting Chief
Nuclear Support Section
Reactor Operations and Nuclear
Support Branch

11/2/76
Date

1. Personnel Contacted

Florida Power Corporation (FPC)

- J. Alberdi - Project Manager
- G. P. Beatty, Jr. - Nuclear Plant Superintendent
- J. C. Hobbs, Jr. - Manager, Generation Testing
- E. E. Froats - Manager, Site Surveillance
- A. P. Vogt - Testing Superintendent
- J. J. Kelly - Shift Testing Superintendent
- B. L. Chastain - Data Collection Supervisor
- S. Parker - Auditor
- E. A. Jesensky - Procedure Coordinator

2. Preoperational Test Results Evaluation

Completed Category I and II preoperational tests were reviewed to assure that the licensee had performed an adequate evaluation of test results and that the test results were within established acceptance criteria. The review also verified the adequacy of the methods used for identifying and correcting test deficiencies and assured that administrative control practices required by Section 13.2 of the FSAR and the Test Program Guide (TPG) were being followed in test results documentation. The inspector tracked the incomplete test steps and found that open items were identified on the "Exception and Deficiency" page and also were identified on the "Punch List" of open test items. The following preoperational tests were reviewed:

- TP 71-201-3, Core Flooding System Functional Test
- TP 71-202-3, Makeup and Purification System Functional Test
- TP 71-203-3, Decay Heat Removal System Functional Test

TP 71-204-3, Reactor Building Spray System Functional Test
TP 72-330-1, CRD Control System Preoperational Calibration
TP 72-330-2, CRD Control System Operational Test

At the completion of this inspection period there were still several open items identified on the punch list as tests restraining fuel loading. Followup inspections of the open preoperational test results will be required. Within the areas inspected no discrepancies were identified.

3. Licensee Preoperational Test Results Evaluation

Thirty completed Category I, II and III preoperational tests were reviewed to assure that the licensee had performed an adequate evaluation and acceptance of test results according to established administrative controls required by Section 13.2 of the FSAR and the Test Program Guide (TPG). Within the test procedures inspected no discrepancies were identified. Followup inspections of the open test results will be required.

4. Hot Functional Test Results

Nine tests of the Hot Functional Test program were reviewed to determine that the testing was adequate and the systems functional as designed. The test results were reviewed to assure that the licensee had performed an adequate evaluation of the test results and that the test results were in conformance to Section 13.2 of the FSAR and NRC Regulatory Guide 1.68 dated November, 1973. The following tests were reviewed:

TP 71-600-0, Hot Functional Testing
TP 71-600-11, Emergency FW and Once Through Steam Generator Level Control
TP 71-600-14, Pipe and Component Hot Inspection Test
TP 71-600-23, Reactor Protection System Functional
TP 71-600-32, Main Steam System
TP 71-600-36, Reactor Building Cooling System
TP 71-600-1, Unit Heatup Test
TP 71-600-7, Nuclear Services Closed Cooling, Decay Heat Closed Cooling
TP 71-600-25, Integrated ES Actuation Test

Within the test procedures inspected no discrepancies were identified. Followup inspections of the open test results will be required.

DETAILS III

Prepared by:

W. B. Swan
W. B. Swan, Reactor Inspector
Engineering Support Section No 1
Reactor Construction and Engineering
Support Branch

10/21/76
Date

Dates of Inspection: October 6-8, 1976

Reviewed by:

T. E. Conlon
T. E. Conlon, Chief
Engineering Support Section No. 1
Reactor Construction and Engineering
Support Branch

10/26/76
Date

1. Individuals Contacted

a. Florida Power Corporation (FPC)

E. L. Griffin - Vice President and Program Manager
E. E. Froats - Manager, Site Surveillance
R. S. Dorrie - Quality Engineer
D. W. Pedrick, IV - Compliance Engineer
C. E. Jackson - Construction Superintendent
T. L. Baker - Supervisor, Quality Records
W. W. Nisula - Structural Engineer
C. Pachos - Architectural and Structural Superintendent

b. Contractor Organization

Gilbert Associates Commonwealth Corporation (GAI)

J. C. Ferr, P.E. - Structural Engineer

2. Scope of Inspection

This followon special announced inspection was made to determine licensee's progress on repair of the containment dome and included review of the licensee's interim reports of this 10 CFR 50.55(e) matter, specifications, field change notices, the FPC work procedure for repair of the dome, and QC records. The adequacy of preparations for placement of the first concrete pour on the top of the dome was verified. Placement of the concrete was continuously monitored from start to finish.

3. Dome Repair - Documents Reviewed

- a. GAI Specification SP-6476, "Removal and Replacement of Concrete, C. R. Unit No. 3, Florida Power Corporation, final Revision 2. 9/30/76." Revision 2 is a complete revision of the specification incorporating all changes to date.
- b. GAI Specification SP-6467, "Radial Anchor Test Program."
- c. Appendix B to SP-5646, "Fabrication and Delivery of Reinforcing Steel," Dated 8/17/76 specifying use of ASTM A615 Grade 60 reinforcement bars instead of Grade 40.
- d. Specification P-SP-6482, "Cadmolding for Dome Repair," revision No. 1 dated 8/27/76.
- e. Specification SP-6460C dated 9/27/76, "Procurement and Installation of Radial Anchors (No. 6 Deformed Bar)."
- f. SP-6463, "Procurement and Installation of Epoxy Grout (Congresive)."
- g. SP-6463B, "Procurement of Epoxy Grout," (Sikcadur Hi-Mod), dated 10/6/76.
- h. SP-6455, "Detensioning and Retensioning of Tendons," dated 7/15/76.
- i. SP-6466, "Dome Instrumentation Installation."
- j. FPC CR No. 3 Work Procedure FPC W-200, "Repair of the Reactor Building Dome, Rev. 10, dated 9/27/76 with Change Notice No. 1 dated 9/27/76. This procedure references drawings which show the locations of the radial anchors, and which anchors are to be tested. Paragraph 9.2 details stipulations on concrete placement, bonding concrete lifts, reinforcing steel, cadwelding, grouting and dome instrumentation. The need to update paragraph 4.1 "Specifications" to list the latest revision of each referenced document was discussed with the licensee.
- k. Amended pages dated September 22, 1976 to FPC Crystal River No. 3 Dome Delamination Report dated June 11, 1976, transmitted to NRC Division of Project Management by letter of September 22, 1976. Included are new Appendixes E through J and a revised dome repair schedule.

1. Quality Control Records including data on in-place tests of designated radial anchors, concrete mix test data, qualification of cadwelders, tests of cadweld sister splices and epoxy grout tests.

4. Design Considerations for Replacement of Dome Cap

For radial anchors the A-E selected, on the basis of site tests, No. 6 Grade 60 rebar grouted to the existing dome concrete with Masterflow 814 (Master Builders) for dry premixed, nonshrinkable, nonmetallic grout, having 6,000 psi strength at 7 days. The anchors are designed for a maximum stress of 11.0 kips and are proof tested at a specified stress of 15.0 kips. A new concrete mix, designed and tested to develop 6,000 psi at 28 days, was specified for the repair although the existing concrete was designed for 5,000 psi at 28 days. A different type of 3/4" aggregate, less frangible than the dolomite previously used was specified. A second layer of rebar below the upper layer in the replacement cap has been placed. The result of these design changes will be that combinations of tensile stresses which caused the dome delamination will now be transmitted as compressive stresses into a stiffer, stronger upper dome cap. Instrumentation cells are being incorporated into the replacement concrete to monitor the stress-strain responses of the repaired dome.

5. Review of Quality Control Records

The inspector reviewed quality control records, pertinent to the completed operations, prior to the concrete placement. These included test data on concrete and on grouts, data on anchor tests, results of cadweld sister splice tests, and qualification of cadwelders.

During the concrete placement, test data taken on delivered concrete batches was monitored.

Within the areas examined, there were no items of noncompliance identified.

6. Observation of Preparations for Concrete Placement

For two days prior to, and on the morning of the concrete placement, the IE inspector observed cadwelding, anchor placement, forming, cleanup, instrumentation embedment and verification of rebar and embedment installations in the area of the first concrete placement and in adjacent areas. The preparations were monitored by the A-E's representative as well as those of the licensee and the builder. The placement work procedure was reviewed.

Within the areas examined there were no items of noncompliance noted.

7. Concrete Placement

Approximately 100 cubic yards of concrete was placed in the first of five placements designated for the dome repair. The placement was made at the top section of the dome. The inspector observed concrete transport, testing, placement, finishing and initial curing. Inspections and QC activities by the builders, licensee and A-E representative were observed.

No violation of design or QC requirements or of work procedure requirements was noted.

8. Repair Operations Remaining To Be Done

- a. Complete concrete placements
- b. Cure all concrete placements for 28 days
- c. Retension 18 tendons

9. Structural Integrity Test of Containment (SIT)

The new repair schedule indicates completion of concrete curing and retensioning of tendons by November 15, 1976. A decision was made that no two of the remaining concrete placements will be made concurrently. At least four days between placements will be required to verify adequate gain strength in each placement. However, if no unforeseen difficulties arise, the SIT could proceed on or soon after November 15, 1976.