UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

Report No. 50-302/79-29

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

Facility Name: Crystal River Unit 3

Docket No. 50-302

License No. DPR-72

Inspection at Crystal River Site near Crystal River, Florida

Inspectors:	C. Julian C. Julian	9/24/79 Date Signed
	P.T. Burnett for M. J. Graham	9-24-79 Date Signed
Approved by:	P.T. Burnett P. T. Burnett, Acting Section Chief RONS Branch	9-24-79 Date Signed

SUMMARY

Inspection on July 24-August 5, 1979

Areas Inspected

This routine, unannounced inspection involved 87 inspector-hours onsite in the areas of precriticality tests, zero-power and low-power physics testing following refueling, and testing of the emergency feedwater system.

Results

Of the four areas inspected, no apparent items of noncompliance or deviations were identified in three areas; two apparent items of noncompliance were found in one area (Infraction - Failure to properly review and approve changes to safety related procedures - Paragraph 5; Deficiency - Failure to follow safety related surveillance procedure - Paragraph 5).

1296 150

7911080222

DETAILS

1. Persons Contacted

   Licensee Employees

   *G. P. Beatty, Jr., Plant Manager
   *W. R. Nichols, Operations Superintendent
   *J. Cooper, Compliance Engineer
   *W. E. Kemper, Technical Specification Engineer
   *G. M. Williams, Compliance Plant Engineer
   *W. A. Cross, Plant Engineer
   L. B. Little, Results Engineer
   M. E. Collins, Plant Engineer
   R. J. Browning, Chem/Rad Protection Technician
   *F. W. Pleubel, Electrical Supervisor

   Other licensee employees contacted included various technicians, and operators.

   NRC Inspector

   *F. Jape

   *Attended exit interview.

2. Exit Interview

   The inspection scope and findings were summarized on July 27, 1979 and August 3, 1979 with those persons indicated in Paragraph 1 above. With regard to the items of noncompliance, discussed with the licensee representatives on July 27, 1979, the licensee representatives had no comment.

3. Licensee Action On Previous Inspection Findings

   Not inspected.

4. Unresolved Items

   Unresolved items were not identified during this inspection.

1296 151

5. Prestartup Preparation and Startup Physics Testing After Refueling

The inspector reviewed the following procedures associated with heatup, approach to criticality, and zero power testing:

- OP-202, Plant Heatup
- PT-100, Precritical Testing
- SP-102, Control Rod Drop Time Test
- PT-110, Initial Criticality and Hot Zero Power Testing
- PT-111, Hot Zero Power, All Rods Out Critical Boron Test
- PT-112, Hot Zero Power Regulating Rod Group Worth and Differential Boron Worth Measurement
- PT-114, Moderator and Temperature Coefficients Determination at Hot Zero Power
- PT-115, Hot Zero Power Ejected Rod Worth Measurement
- PT-116, Sensible Heat Determination

When SP-102 (Control Rod Drop Time Test) was performed on July 26, the latest revision, dated July 25, 1979, was not yet available in the control room. The 7/25 revision corrected the location of each control rod to reflect the current core loading. Changes similar to those in the new revision were inked onto a copy of the outdated revision and used in performance of the test. The changes were unsigned, and there was no evidence of a formal temporary change of procedure.

This finding of a failure to review and approve changes implemented in a surveillance and test procedure of safety related equipment represents noncompliance with Technical Specification 6.8.2 (50-302/79-29-01).

The inspector also found in SP-102 that when the recorder traces were measured to determine the drop times, the end of the drop interval was taken to be the beginning ofthe 25% limit switch voltage pulse, rather than the midpoint, as specified by SP-102 Step 6.9. This resulted in a 2% nonconservatism in each rod drop time. When corrected, all rod drop times still met the acceptance criteria.

The finding of failure to implement SP-102 as written represents noncompliance with Technical Specification 6.8.1(c) (50-302/79-29-02).

On the data and results sheets of SP-102 the inspector noted several instances of white out corrections, and corrections marked on top of the incorrect number. These practices were not widespread, but the inspector brought them to the attention of the licensee, and expressed a concern that care always be taken in correction of erroneous entries.

1296 152

6. Review of Zero Power Physics Testing

The inspector reviewed the data generated during the performance of the following testing procedures:

PT-110, "Controlling Procedure for Zero Power Physics Testing"
PT-111, "Hot Zero Power All Rods Out Critical Boron Test"
PT-116, "Sensible Heat Determination"
PT-114, "Moderator and Temperature Coefficient Determination at Hot Zero Power"
PT-112, "Hot Zero Power Regulating Rod Group Worth and Differential Boron Worth Measurement"
PT-115, "Hot Zero Power Ejected Rod Worth Measurements"

After completion of these tests the major discrepancy identified by the licensee involved procedure PT-115. The reactivity worth of the "worst-case ejected rod" was predicted to be 0.55% delta k/k while the measured value was 0.431. The deviation between measurement and prediction was 27.6% which exceeds the PT-115 acceptance criterion of ± 20%. This means that an ejected rod accident would not introduce as much positive reactivity as predicted. Babcock and Wilcox was consulted on this matter, and on 7/31/79 the Plant Review Committee determined the test results to be acceptable for safe reactor operation.

While reviewing the data of these tests the inspector noted two errors in data manipulation. When these were pointed out to the licensee, they were promptly corrected. Neither error was of safety significance or rendered the test results unacceptable.

With the correction of these items, the inspector had no further questions. No items of noncompliance were identified in this area.

7. Reactor Power Escalation Testing

The inspector observed a major portion of the tests conducted at the 15 and 40% plateaus during power escalation, and reviewed all the data resulting from tests at these levels as documented in procedure PT-120, "Controlling Procedure for Power Escalation Testing".

Some difficulty was encountered in demonstrating correlation between core axial offset measured by incore detectors as displayed by the process computer and axial offset displayed by selected incore detectors which read out on a control room strip chart recorder. It was finally determined that the strip chart recorder was incorrectly identifying the detector channels and when this error was corrected the axial offset correlation at 40% power met the acceptance criteria.

No items of noncompliance were identified in this review.

8. Emergency Feedwater System Test

The inspector observed the performance of procedure PT-123,"Emergency Feedwater Flow Verification and Manual OTSG Level Control Independent of ICS with the Reactor at Power". Following the test, the inspector reviewed the data collected, and found them satisfactory. The test appeared to achieve its stated purposes of demonstrating that emergency feedwater can be manually controlled to the steam generators with the reactor operating and that the emergency feedwater pumps are capable of delivering at least 550 gpm flow to a single steam generator at 1050 psig steam pressure. It is the inspector's understanding that the data collected will be transmitted to the NRC Office of Nuclear Reactor Regulation for further detailed evaluation.