June 8, 1976
L-76-215 Central Files
50-335

Norman C. Moseley, Director
Office of Inspection and Enforcement-Region II
U. S. Nuclear Regulatory Commission
230 Peachtree Street, N. W., Suite 818
Atlanta, Georgia 30303

Dear Mr. Moseley:

Re: IE:II:MSK
50-335/76-6

Florida Power & Light Company has reviewed the subject inspection report and has determined that it contains no proprietary information.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President

REU/LLL/hlc

cc: Jack R. Newman, Esq.

BFB

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

MAY 20 1976

In Reply Refer To:
IE:II:MSK
50-335/76-6

Florida Power and Light Company
ATTN: Dr. R. E. Uhrig, Vice President
of Nuclear and General Engineering
P. O. Box 013100
9250 West Flagler Street
Miami, Florida 33101

Gentlemen:

This refers to the inspection conducted by Messrs. M. S. Kidd and J. D. Martin of this office on April 7-9, 12-16 and 18-22, 1976, of activities authorized by NRC Operating License No. DPR-67 for the St. Lucie 1 facility, and to the discussion of our findings held with Mr. K. N. Harris at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspector.

Within the scope of this inspection, no items of noncompliance were disclosed.

We have examined actions you have taken with regard to previously reported unresolved items. These are identified in Section IV of the summary of the enclosed report.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If this report contains any information that you believe to be proprietary, it is necessary that you submit a written application to this office requesting that such information be withheld from public disclosure. If no proprietary information is identified, a written statement to that effect should be submitted. If an application is submitted, it must fully identify the bases for which information is claimed to be proprietary. The application should be prepared so that information sought to be withheld is incorporated in a separate paper and referenced in the application since the application will be placed

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in the Public Document Room. Your application, or written statement, should be submitted to us within 20 days. If we are not contacted as specified, the enclosed report and this letter may then be placed in the Public Document Room.

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Very truly yours,

F. J. Long, Chief
Reactor Operations and Nuclear
Support Branch

Enclosure:
IE Inspection Report No.
50-335/76-6

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

IE Inspection Report No. 50-335/76-6

Licensee: Florida Power and Light Company
P. O. Box 013100
Miami, Florida 33101

Facility Name: St. Lucie 1
Docket No.: 50-335
License No.: DPR-67
Category: B2

Location: Hutchinson Island, Florida

Type of License: CE, PWR, 2560 Mwt

Type of Inspection: Routine, Announced

Dates of Inspection: April 7-9, 12-16, and 18-22, 1976

Dates of Previous Inspection: March 1-5, 1976

Principal Inspector: M. S. Kidd, Reactor Inspector
Reactor Projects Section No. 2
Reactor Operations and Nuclear Support Branch
(April 12-16 and 18-22, 1976)

Accompanying Inspector: J. D. Martin, Reactor Inspector
Nuclear Support Section
Reactor Operations and Nuclear Support Branch
(April 7-9 and 18-22, 1976)

Other Accompanying Personnel: None

Principal Inspector: M. S. Kidd 5-14-76
M. S. Kidd, Reactor Inspector
Reactor Projects Section No. 2
Reactor Operations and Nuclear Support Branch
Date

Reviewed by: R. C. Lewis 5/19/76
R. C. Lewis, Chief
Reactor Projects Section No. 2
Reactor Operations and Nuclear Support Branch
Date

SUMMARY OF FINDINGS

I. Enforcement Items

None

II. Licensee Action on Previously Identified Enforcement Matters

Not applicable.

III. New Unresolved Items

None

IV. Status of Previously Identified Unresolved Items

75-18/1 Leak Detection System Sensitivity

Sensitivity of the inventory balance method of leak detection was verified during pre-critical testing. This item is closed. (Details I, paragraph 2.b.(2))

75-19/1 Installation and Testing of Hangers and Restraints

Acceptability of test data taken during pre-fuel load testing has been clarified. Monitoring of piping hangers, restraints, and snubbers, and piping deflections was accomplished during pre-critical testing. This item is closed. (Details I, paragraph 2.a(5), and Details II, paragraph 2)

76-2/2 Baseline Inspection Data

An engineering analysis of the ultrasonic indications in the reactor coolant pump support lugs resulted in the conclusion that the lugs are structurally adequate. This analysis was approved by the licensee. This item is closed. (Details I, paragraph 2.a.(4))

76-4/2 Containment Boundary Quality Group Designation (50.55(e))

Errors in the Ebasco report of January, 1976, regarding weld and isometric identifications have been corrected. All upgraded welds have now been non-destructively tested. This item is closed. (Details I, paragraph 2.a.(6))

V. Unusual Occurrences

A. Core Flow

Analysis of pre-critical reactor core flow data by Combustion Engineering revealed that the flow was approximately two percent less than the value required by Technical Specifications. This is to be reported as a fourteen day written report per RG 1.16.

B. CEDM Malfunction

During cold rod drop testing control element drive mechanism (CEDM) number 44 malfunctioned such that the control element assembly (CEA) could not be withdrawn. The CEDM operated normally at rated temperature and pressure. (Details I, paragraph 2.a(3))

C. MSIV Closure Time

The main steam isolation valve (MSIV) for "B" header closed in seven seconds during surveillance testing as compared to the maximum of six seconds allowed by Technical Specification 3.7.15. This item will be reported per RG 1.16.

D. Defective Mechanical Pipe Restraints

During plant heatup for post core load testing, five mechanical pipe restraints were found to be locked up. Corrective actions completed at the time of the inspection were reviewed. This matter was reported per RG 1.16 April 12, 1976. (Details II, paragraph 2)

VI. Other Significant Findings

A. Plant Status

Completion of the activities required prior to initial criticality by Enclosure 1 to the Unit 1 operating license, along with licensee commitments relative to criticality were verified to be complete. (Details I, paragraph 2 and Details II, paragraphs 2 and 3)

B. Initial Criticality

Portions of the approach to initial criticality on April 20-22, 1976, were witnessed by the inspectors, with no discrepancies identified. (Details I, paragraph 3 and Details II, paragraph 4)

VII. Management Interviews

Management interviews were conducted April 9 and 16, 1976, with K. N. Harris and other licensee staff members to discuss findings of the inspection relative to completion of requirements for starting the approach to initial criticality. Items discussed included the requirements listed in Enclosure 1, Section A of the operating license and the previously identified unresolved items in Section IV of this Summary. (Details I, paragraph 2 and Details II, paragraphs 2 and 3)

Another management interview was conducted April 22, 1976, with K. N. Harris and members of his staff to discuss findings of the inspection relative to the approach to initial criticality. The results of the inspection were discussed. (Details I, paragraph 3 and Details II, paragraph 4)

DETAILS I

Prepared by: R. C. Lewis for 5/19/76
M. S. Kidd, Reactor Inspector Date
Reactor Projects Section No. 2
Reactor Operations and Nuclear
Support Branch

Dates of Inspection: April 12-16 and 18-22, 1976

Reviewed by: R. C. Lewis 5/19/76
R. C. Lewis, Chief Date
Reactor Projects Section No. 2
Reactor Operations and Nuclear
Support Branch

1. Persons Contacted

Florida Power and Light Company (FP&L)

K. Harris - Plant Manager
J. Barrow - Operations Superintendent
C. Wells - Operations Supervisor
R. Ryall - Reactor Supervisor
P. Dillon - Technical Staff Supervisor
R. Hayes - Technical Staff Engineer
J. Pride - Technical Staff Engineer
K. Beard - Maintenance Engineer
J. Lenz - Instrumentation and Control Engineer
A. Anderson - Construction QA Engineer
R. Roehn - Construction QA Engineer
Two Nuclear Plant Supervisors
Two Nuclear Watch Engineers
Four Nuclear Control Center Operators

Ebasco Services Incorporated (Ebasco)

J. Albanes - Stress Analyst Supervisor
M. Noronha - Stress Analyst Supervisor
P. Peterson - Quality Control Records

Combustion Engineering Incorporated (CE)

E. Smith - Site Manager
J. Tefft - Chief Test Engineer
T. Oliver - Startup Engineer

2. Completion of Requirements Relating to Initial Criticality

a. Operating License Enclosure 1

Section A of Enclosure 1 of the St. Lucie Unit 1 Operating License, DPR-67, set forth six activities which required resolution or completion prior to achieving initial criticality. During the inspection the inspectors verified that the activities had been completed or resolved. Methods of verifying completion are discussed below and in Details II, paragraphs 2 and 3.

(1) Installation of Tornado Missile Protection

Verification of these installations is discussed in Details II, paragraph 3.

(2) Spent Fuel - Charging System Intertie

Verification of the installation of this line is discussed in Details II, paragraph 3.

(3) Cold Rod Drop Tests

Pre-critical rod drop tests were conducted in accordance with test procedure 0110081, "CEDM/CEA Measurements."

Data sheet 1 of this procedure was used to document rod drop tests at cold (260°F) and hot (532°F) conditions for each full length control element assembly (CEA). Review of rod drop times at the 260°F ± 20°F plateau revealed the following information:

(a) Drop times on all rods except number 44 with two reactor coolant pump (RCP) flow had been conducted as of April 2, 1976, at which time the decision was made to continue plant heatup and testing. CEA 44 could not be drop-tested because of malfunction of the lower gripper.

Drop times for ninety percent insertion ranged from 1.96 seconds to 2.35 seconds as compared to the limit of 3.3 seconds given in Technical Specification 3.1.3.4 for hot conditions and full core flow.

(b) The two fastest CEA's (Nos. 8, 46) and two slowest CEA's (Nos. 51, 69) were dropped once again with two RCP's operating and once each with three RCP's operating and with no flow through the core. The inspector noted that the number of retests did not compare favorably with the number planned for hot retests as discussed in FSAR Table 14.1-2, item 1. Licensee personnel agreed that a comparable number of retests should be performed at cold conditions and two additional retests were conducted on the four CEA's at no flow and three RCP flow on April 14, 1976. Retest times compared favorably with the first tests.

(c) CEA 44 was drop tested successfully three times at the 260°F plateau on April 14, 1976, after cooldown of the reactor coolant system (RCS) from 532°F. These tests were all conducted with 2 RCP flow. The lower gripper malfunctioned again on that date before tests with no flow and 3 RCP could be accomplished. The drop time for CEA 44 was 1.96 seconds, making it the second fastest CEA, replacing CEA 8.

(d) Review of coil current recorder traces of CEA 44 and discussions with licensee personnel revealed that as RCS temperature was raised from 260°F to 532°F the lower gripper began to function properly and demonstrated normal operation at the higher temperature.

On April 16, 1976, the operating license for St. Lucie 1 was amended by NRR, Division of Operating Reactors at the request of FP&L, to delete the special test exception in Specification 3.10.3 which would allow the reactor to be taken critical below 515°F. This action obviated the need for further testing of CEA 44 at cold conditions. Two of the stipulations upon which the change was based included testing of the third fastest CEA as 44 would have been tested and the performance of ten additional drops on CEA 44 at hot conditions. These activities were verified complete by the inspector on April 16, 1976. Licensee personnel were informed that the inspector had no further questions in this area.

(4) Baseline Inspection Results (Unresolved Item 76-2/1)

This unresolved item was last discussed in IE Report No. 50-335/76-4, Details VI, paragraph 3. At the conclusion of that inspection all aspects of the unresolved item had been resolved except for evaluation of ultrasonic indications on the RCP support lugs by CE and FP&L and submittal of the

baseline inspection report. The baseline report, transmitted under CP&L's letter L-76-98, dated March 11, 1976, has been reviewed by IE:II with no further questions resulting. CE's evaluation of the RCP lug indications, forwarded to FP&L by letter F-CE-5640, dated March 2, 1976, concluded that the lugs were structurally adequate based on indicated flaws in them. FP&L Engineering concurred with that conclusion. Licensee personnel were informed that this item was considered closed.

(5) Testing of Hangers and Restraints (Unresolved Item 75-19/1)

This unresolved item was last discussed in IE Report No. 50-335/76-4, Details I, paragraph 4.d. At that time, three questions relative to testing results documented in pre-fuel load testing per procedure 0010185, "Piping Thermal Expansion and Restraint," remained open. Review of the amended test package and discussions with licensee personnel during the current inspection revealed the following information:

(a) The reactor coolant system (RCS) temperatures at which data were taken had not been included in the data sheets. A letter which had been reviewed by the Facility Review Group (FRG) on February 28, 1976, stating that the actual temperatures were provided in test procedure 0010181, "Hot Functional Sequencing Document," on an hourly basis had been entered into the test results file.

(b) A letter from CE to the plant manager, dated February 27, 1976, explaining the acceptability of certain data recorded in Table 1 of the test procedure, had been entered into the test package. This letter had also been reviewed and concurred with by the FRG.

(c) Movements of the pressurizer relief lines were monitored during hot functional testing (HFT), but power operated relief valves (PORV) were not installed at that time. The inspector verified that these lines were monitored during precritical testing on April 21, 1976, per test procedure 0120096, "Pressurizer Control Checkout" (Revision 5), while steam was being discharged through the PORV's to the quench tank. Section 12.6 of 0120096 also was used to

functionally test the ability of the quench tank to receive and quench pressurizer steam and to functionally check the resistance temperature detectors (RTD) in the pressurizer discharge lines used for leak detection purposes.

Inspection efforts regarding retesting of hangers, restraints, and snubbers during precritical testing are documented in Details II, paragraph 2 of this report. This item is considered closed.

(6) Containment Boundary Quality Grouping (Unresolved Item 76-4/2)

This item involved improper quality designation of certain systems penetrating containment and was the subject of a report per 10 CFR 50.55(e) submitted by FP&L on January 22, 1976. The improper designation resulted in a lack of certain nondestructive testing (NDT), and the affected components were subsequently upgraded to quality group "B" from "C" by performance of the necessary NDT. The initial review by IE:II of the corrective measures in this area revealed that the interim report from Ebasco to FP&L contained numerous errors and discrepancies in weld and isometric identification. Additionally, three shop welds in penetration 43 had not undergone required radiography and liquid penetrant testing for upgrading.

During this inspection, a corrected Ebasco report, transmitted by letter number 9063 on March 9, 1976, was reviewed. All instances of incorrect weld and isometric identifications had been corrected in this report except for field welds FW-8 and 8A in piping I-8-CC-168-2. These were still identified as FW-8A and 8B, respectively. The file copy of the report was corrected during the review and licensee personnel stated that recipients of the report would be advised of the change. NDT records on file reflected the proper weld numbers. The inspector also reviewed QC records of NDT performed on three shop welds in the reactor drain tank pump suction line, SW-1, SW-2, and SW-3 (isometric drawing WM-P1). These records included Ebasco QA radiographic summary reports and liquid penetrant test reports. All findings in these reports indicated acceptable welds. Licensee personnel were informed that this item was considered closed.

h. Followup on FP&L Commitments

As discussed in IE Report No. 50-335/76-4, Details I, paragraphs 2.c and 2.e.(1), FP&L committed to resolve certain preoperational test exceptions and establish the sensitivity of the inventory balance method of leak detection prior to initial criticality.[1/] Followup review indicated this status:

(1) Test Exceptions

(a) TP 1400090 - "Data Processor/Process Instrumentation Correlation Measurement Test" - Two temperature indicators and one pressure indicator could not be compared (DDPS versus control board indication) during HFT. During pre-critical testing, temperature element (TE) 2221 outputs were compared with favorable results. TE-2229 is no longer a dual element RTD and is not used by the DDPS. The pressure transmitter involved, PT-08-13B3, is a spare and not yet installed. This is one of three which can be used by DDPS for steam generator 1B pressure.

(b) TP 1900082 - "Miscellaneous Ventilation Systems Startup" - Decontamination room supply and exhaust fans, HVS-11 and HVE-36, were checked for vibration on April 13, 1976, after realignment and found to be acceptable. Test results were reviewed and approved by the FRG on April 14, 1976.

(c) TP 2000084 - "Hydrogen Purge System Startup and Functional Test" - Documentation that flow recorder FR-25-1 had been repaired and retested was reviewed and approved by the FRG on March 22, 1976.

(d) TP 1120080 - "Area Radiation Monitoring Pre-Operational Functional Test" - The remote alarms for these units were disconnected due to intermittent and frequent downscale alarm actuations. Small radioactive sources were placed near the monitors to increase background levels and appropriate steps of TP 1120080 were repeated. The documentation of these activities, which was placed in QC files to clear the deviation, was reviewed by the FRG February 27, 1976.

The inspector stated that he had no further questions.

1/ See FP&L letter L-76-53, dated February 10, 1976.

(2) Leak Detection Sensitivity (Unresolved Item 75-18/1)

An RCS water inventory balance was performed on April 8, 1976, followed by another test wherein a known leak of one (1) gallon per minute (gpm) was superimposed on the RCS through a sample line in accordance with step 12.10 of TP 0010190, "Post Core Load Hot Functional Sequencing Document." The balances were performed per OP 0010125, "Schedule of Periodic Tests, Checks and Calibrations," data sheet 1. Results documented for the first check indicated a leak rate of .68 gpm. Results with a known leak imposed of 1.0 gpm were 1.66 gpm. The two tests, disregarding the known leak, were within the ±.25 gpm acceptance criterion established in TP 0010190. The inspector stated that this item was considered closed.

During review of the data/calculation sheet used for inventory balance, it was noted that RCS temperature was not recorded, making it necessary to verify temperature had not changed during the test from other documentation. Licensee personnel stated that spaces for recording beginning and ending RCS temperatures would be added to the data sheet. The inspector stated that he had no further questions at that time.

3. Initial Criticality

Portions of the preparations for and approach to initial criticality were witnessed by the inspectors on April 18-22, 1976. The objectives were to determine whether the activities conformed to Technical Specification requirements (sampling basis) prior to CEA withdrawal and throughout the evolution, assure that administrative and procedural requirements were met, assure that procedure changes and deficiencies were handled per administrative procedures, independently predict the initial critical boron concentration, and observe overall conduct of the approach.

a. General

The start of the approach to criticality was delayed due to the problems discussed in the Unusual Occurrences section of the Summary of this report. CEA withdrawal was started on April 20, 1976, after completion of sections 4, "Prerequisites," and 6, "Related System Status," of the procedure TP 3200086, "Initial Criticality" (Revision 2). After withdrawal of CEA's and during CEA testing per Appendix A-3 of 3200086,

problems were experienced with CEA Nos. 29 and 33. CEA 29 displayed both an upper and lower limit indication, requiring replacement of a defective reed switch. CEA 33 dropped into the core while being inserted. Licensee personnel found that a test cable, used to feed test devices used to monitor CEDM coil currents, was picking up induced current (noise) from other cables in its rack. A slight shift of the cable in the rack alleviated the problem. Other CEDM's have the same test cable and were being monitored to assure a similar problem did not exist with them.

CEA testing was completed and boron dilution started at 7:56 p.m. on April 21, 1976. Criticality was reached at approximately 8:30 a.m. on April 22, 1976, at a boron concentration of approximately 935 - 940 ppm, as compared to the predicted value of 938 ppm (all rods out except Group 7, which was at 68 inches withdrawn). The approach was conducted in accordance with the procedure and section 14.1.4.4 of the FSAR.

b. Specific Inspection Activities

(1) The licensee's compliance with Technical Specifications requirements for Modes 3* and 2* was ascertained on a sampling basis by observation by the inspectors of plant system conditions and review of appropriate documentation. Completed control room data sheets and surveillance test procedures were reviewed, including the following:

(a) "Minimum Equipment List" - This is a list which reflects the number of components in systems covered by Technical Specifications sections 3/4 which must be operable for the various modes defined by Technical Specifications. It is filled out each shift during modes 1 through 4.

(b) "Minimum Instrumentation List" - This list is similar to the one above and covers all plant instrumentation required to be operable during modes 1 through 4. It is also filled out by the nuclear control center operator (NCCO) once per shift.

(c) "NCCO Log Sheet No. 1 (SDC Secured)" - This data sheet is filled out hourly and is used to monitor several parameters which are limited by Technical Specifications, such as RCS temperature and pressure.

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* See Technical Specifications - Definition Table 1.1.

(d) Periodic test procedures, such as those contained in AP 0010125, "Schedule of Periodic Tests, Checks and Calibrations," (Revision 5) for daily, weekly, shiftly and more frequent surveillances.

Completed documents of the types described above were reviewed at periodic intervals from April 18, 1976, to April 22, 1976. Also the NCCO Log and Nuclear Plant Supervisor Log were reviewed for the period of April 18-21, 1976. No discrepancies were identified.

(2) Shift complements during the approach were observed to assure that the requirements of Technical Specification Table 6.2-1 were met. The requirements were met or exceeded for each shift observed.

(3) The latest revision of the procedure, 3200086 (Revision 2) as amended, was used throughout the evolution. Changes to it and other procedures were accomplished in accordance with station administrative procedures.

(4) Inverse neutron count rate (1/m) plots were maintained per the procedure as was a plot of boron concentration versus dilution time.

(5) Technical support was provided on a shiftly basis by the presence of Reactor Engineering (FP&L) and CE Startup personnel.

(6) Prior to the start of CEA withdrawal, the licensee's prediction of estimated boron concentration at criticality, along with CE's, were discussed with licensee personnel. An independent calculation was made, using data provided in FSAR Table 4.3-1, 4.3-2, 4.3-4 and 4.3-6. The resultant value was in close agreement with FP&L and CE predictions.

Licensee personnel were informed that the inspector had no comments or questions relative to the conduct of the approach to criticality.

4 <u>Followup on Previous Open Items</u>

Licensee actions in response to previous questions and comments by IE:II inspectors on subjects other than those in paragraph 2 were reviewed and discussed. The following information was obtained:

a. Test Procedure Comments

(1) TP 0110086, "Simulated CEA Ejection, Hot Zero Power," and TP 3200081, "Nuclear Design Check Tests"

Comments on these low power test procedures were documented in IE Rpt. No. 50-335/76-1, Details III, paragraphs 5a and 5b. Review of the revised procedures revealed this information:

(a) A reference to the technical manual has been added to provide details on hookup of special test equipment referenced in step 8 of TP 3200081.

(b) Step 9.12 of 0110086 and 3200081 stated that certain approvals had to be obtained prior to proceeding with the various steps of the procedures, but did not require documentation of the approvals. These steps were deleted from Revision 1 of each procedure. Step 4.8 of Letter of Instruction (LOI) No. 0-7, "Conduct of Operations During Core Loading, Low Power Physics Testing and Power Ascension Testing" infers that the Reactor Engineering staff will be available to advise and help coordinate testing efforts.

(c) Revision 0 of the procedures contained blank spaces for predicted design values of certain parameters which were not available at the time the procedures were approved. Review of the revised procedures and other selected startup test procedures revealed that most blanks had been filled in in Revision 1. Certain steps, such as 12.1.5 of 3200081, referenced the Plant Curve Book as the proper source of predicted values of design parameters. Licensee personnel stated that an effort would be made to assure that all blanks were filled in before testing started for any given procedure and that one of the functions of the FRG during review of results would be to assure that proper design values had been entered into each procedure and results compared to them.

The inspector had no further questions on these procedures.

(2). TP 2100089, "Generator Trip"

The inspector reviewed followup action by the licensee in response to comments documented in IE Report No. 50-335/76-2, Details I, paragraph 4.a and determined the following:

(a) Revision 0 did not specify recorder alignment/hookup and calibration for instrumentation used to record data in step 6.5. Review of Revision 1 revealed that no additional information had been added in this area. Further discussions on this and other startup procedures revealed that the procedures do not, in general, provide this type of information. Licensee personnel stated that LOI-0-7, the administrative procedure for the startup program would be revised to require the test engineers to verify that test instrumentation is in calibration (e.g. - current sticker) before starting a test. Also, LOI-0-7 will define the use of the test data "patch panel" which is normally used to hook up recorders for data retention during testing. The "patch panel" was installed especially for that purpose and provides the desired isolation of recorders to prevent feedback into protection channels. It was further stated that any special hookups required other than the "patch panel" would be addressed in the appropriate test procedure, thus receiving prior review and approval.

(b) The manufacturer's recommended value of 111% has been inserted in Step 10.2 as the acceptance criterion for maximum turbine overspeed.

(c) During discussions on whether specific steps are to be done manually or automatically (all steps are signed of as "verified by"), licensee personnel stated all steps of the procedure except 12.8 which start with "Verify..." mean that the operator is to observe automatic functions and verify that they do indeed perform as designed. Step 12.8 references an emergency operating procedure and portions of that procedure would be done manually. Also, the FP&L intent of the word "verify" is that if a function has not operated properly, then the operator should

perform it manually. It was noted that the use of "verify" in this dual role is widespread in St. Lucie procedures including emergency operating types.

In repsonse to the inspector's statement that the dual use of "verify" appeared to be confusing in test as well as emergency operating procedures (EOP), licensee representatives stated that these actions would be taken:

1 LOI-0-7 will be revised to address the dual use of "verify" in startup test procedures.

2 Administrative procedure 0010120, "Duties and Responsibilities of Operators on Shift," will be revised to address the use of "verify" and the philosophy of how it is used is to be brought to all operators' attention.

3 All EOP's will be revised within 45 days to include a statement in the immediate operator actions section that if any automatic actions have failed to take place, the operator should perform them manually. Also, Off-Normal procedures, which utilize a format similar to EOP's, will be reviewed to determine whether revisions are necessary to them.

The inspector stated that he had no further questions on items (a) and (b) above, but would discuss item (c) and review licensee actions relating to it during subsequent inspections.

(3) TP 2100091, "Loss of Off-Site Power"

Comments documented in IE Report No. 50-335/76-2, Details I, paragraph 4.b, have been acted on as follows in Revision 2 of the procedure:

(a) Step 6.1 has been changed to state that pressurizer level and pressure controls are in automatic for the test.

(b) Step 4.4 now provides that all test personnel will be briefed before the test begins.

(c) Step 12.7.6 has been revised to clarify that steam generator water levels are to be controlled after the plant trips by manual control of the auxiliary feedwater system. All other steps involve verification that automatic functions are operating properly. (See paragraph (2)(c) above)

(d) Acceptance criterion 10.8 was added to place a limit of 2500 psia on RCS pressure during the test.

The inspector stated that there were no further questions on this procedure.

(4) <u>OP 1200051, "Nuclear and Delta T Power Calibration," and OP 1200021, "Incore-Excore Flux Monitor Correlation"</u>

Comments on these periodic/surveillance test procedures were discussed in IE Report No. 50-335/76-2, Details II, paragraph 3.b. The computer is required to be in service in order to accomplish various steps, such as 8.2 of 1200021. These procedures no longer have data sheets. The inspector stated that there were no further questions on the procedures.

(5) <u>TP 0110088, "Static CEA Drop Test at 50% Power," and TP 0110089, "Dynamic CEA Drop Test at 50% Power"</u>

Comments on these test procedures were discussed in IE Report No. 50-335/76-2, Details II, paragraph 3.c. The data sheets for these procedures which did not contain spaces for signoffs were deleted in Revision 2. The acceptance criteria (section 10 of each procedure) were rewritten in Revision 1 to make them more definitive. The inspector stated that there were no further questions on these procedures.

b. <u>Emergency Operating Procedures</u>

Selected emergency operating procedures had been reviewed to determine whether they provided adequate instructions regarding alternate methods of supplying cooling water to the reactor vessel in the event of an accident. As noted in IE Report No. 50-335/76-5, Details I, paragraph 5, comments remained outstanding on OP 0120042, "Loss of Reactor Coolant," and OP 0410030, "HPSI-Off Normal." OP 0410030 was revised in Revision 2 to give additional guidance on alignment of "C" HPSI pump in the

event the "A" or "B" pump failed to start. OP 0120042 was under revision at the time of the inspection to add instructions for use of a spent fuel system-charging system inter-tie, a backfit item. The inspector stated that he had no further questions on these procedures.

c. Preoperational Test Results

The status of comments on completed test packages, in addition to those discussed in paragraph 2, were reviewed with licensee personnel.

(1) TP 0010184, "Reactor Coolant System Component Expansion Measurement"

Comments on this test were documented in IE Report No. 50-335/75-19, Details I, paragraph 3.a. Review of licensee actions in response to those comments revealed the following information:

(a) Two letters, reviewed by the FRG, had been entered into the package to explain the details of how two sets of data had been recorded for step 12.3.1.5 for several data sheets, and to define which set was the proper one in each case.

(b) The actual RCS temperatures at the time data were taken could not be retrieved; however, the HFT sequencing document provided data which demonstrated that the temperatures were within the ranges allowed.

(c) Documentation of the dimensions for steam generator sliding bases, calculated per step 12.3.1.8, had been reviewed and entered into the test package.

The inspector stated that he had no further questions on this test.

(2) TP 0410082, "Safety Injection Tank Dump Test"

In IE Report No. 50-335/76-4, Details IV, paragraph 3.a, it was noted that the safety injection tank pressure alarms had not been tested during the initial conduct of the test. They were tested on April 3, 1976, when the tanks were pressurized with nitrogen. Results of the tests showed that the setpoints for the low, low-low,

high, and high-high alarms were within the acceptance criteria of the procedure and the upper and lower limits of Technical Specification 3.5.1.d. (200-250 psig). The inspector stated that there was no further question on this test exception. The other comment, involving clarification of tank discharge times and the acceptability of them, will be reviewed in more detail during a subsequent inspection.

(3) TP 1010082, "Containment Instrument Air System Functional Test"

In IE Report No. 50-335/76-4, Details IV, paragraph 3.b, it was noted that Table 9.3-1 of the FSAR contained an incorrect value (162 scfm) for the design capacity of these compressors. This value was changed to 48 scfm, the correct figure, in Amendment 57 to the FSAR. Licensee personnel were informed that there were no further questions on this test.

c. Core Protection Calculator Design Correction

The changes necessary in the core protection calculator for the Thermal Margin/Low Pressure Trip, first discussed in IE Report No. 50-335/76-5, Details I, paragraph 6, were completed and the applicable reactor protection system channels retested prior to initial criticality. Licensee documentation regarding this problem reviewed included Plant Work Order (PWO) 3771, Plant Change/Modification (PCM) 20-76 and letters from CE to FP&L dated March 19 and 26, 1976, which explained the problem and need for changes. These modifications involved a wiring change in the input to the CEA function summer and a change in a gain coefficient (resistor) in the circuitry. The PCM had been completed on April 15, 1976, and a retest of the channels was in progress per OP 1400050, "Reactor Protection System - Periodic Test." The inspector stated that there were no further questions on this subject.

DETAILS II

Prepared by: H. L. Whitener for 5/12/76
J. D. Martin, Reactor Inspector Date
Nuclear Support Section
Reactor Operations and Nuclear
Support Branch

Dates of Inspection: April 7-9, 18-22, 1976

Reviewed by: H. C. Dance for 5/19/76
H. C. Dance, Chief Date
Nuclear Support Section
Reactor Operations and Nuclear
Support Branch

1. Persons Contacted

Florida Power and Light (FP&L)

K. Harris - Plant Manager
J. Barrow - Operations Superintendent
C. Wells - Operations Supervisor
P. Dillon - Technical Staff Supervisor
G. Vaux - Quality Control Supervisor
N. Roos - Quality Control Engineer
A. Collier - Instrument and Control Supervisor
G. Boissy - Assistant Maintenance Superintendent - Electrical
J. Pride - Technical Staff Engineer
J. Garner - Assistant Plant Engineer
K. Beard - Maintenance Engineer
P. Cheries - Chemistry Engineer
D. Brandt - Nuclear Plant Supervisor
D. West - Nuclear Plant Supervisor
W. Windecker - Nuclear Plant Supervisor
R. Ryall - Reactor Supervisor
D. Stutzman - Reactor Engineer

EBASCO Services, Inc. (EBASCO)

W. Taylor - Construction Superintendent
G. Lemon - Site Project Engineer
H. Rogalski - Mechanical Engineer
M. Noronha - Stress Analyst Supervisor
J. Albanes - Stress Analyst Supervisor

Combustion Engineering (CE)

T. Oliver - Startup Engineer
E. Smith - CE Site Manager
F. Sears - NSSS Test Manager

2. Followup on Previously Identified Unresolved Item

... Installation and Testing of Hangers and Restraints (75-19/1)

Test results documented in the "Piping Thermal Expansion Post Core Load" preoperational test No. 0010194 were reviewed in total by the inspector. The inspector verified that additional testing required by unresolved item (75-19/1) covering the monitoring of all mechanical and hydraulic snubbers and piping restraints during post core load heatup to rated temperature and pressure were completed and the results compared favorably with the predicted valves.

Reportable Occurrence 335-76-9 documented the fact that during heatup for Post Hot Functional Testing five mechanical (I.N.C. Bergen-Patterson Model MSVA-1) snubbers were found locked-up at the first plateau of 260°F. The immediate action was to promptly replace the defective snubbers with self relieving (Pacific Scientific) mechanical snubbers. The inspector has since verified by discussions with the licensee, review and discussion of the Ebasco documentation and spot-checks in the plant, that all I.N.C. snubbers have now been replaced with the Pacific Scientific nonlocking type mechanical snubbers. During the heatup to rated conditions the inspector verified that the mechanical snubbers received 100% surveillance to assure that no other mechanical snubbers had locked up. This was accomplished by physically moving the piping and observing the snubber movement. In the cases where the piping could not be moved the snubber was unpinned and independently stroked. A check sheet was used to verify that the snubber was then properly replaced and repinned. A followup report on the failure mode of the five locked up I.N.C. snubbers is to be submitted by the licensee upon completion of the current investigation.

10 CFR 50.55(e) Final Report dated March 16, 1976, explained the fact that five of the six hydraulic seismic snubbers found by FP&L Power Resources Maintenance personnel to have apparent low fluid levels were really not low fluid levels but appeared low due to calibration plate misalignment. The sixth snubber leaked during static testing through a weep hole due to a

damaged accumulator piston seal. In all cases it was determined that the snubber would operate properly under design basis conditions since the low fluid levels (apparent or real) constituted a loss of reserve fluid and not fluid needed for damping action. The inspector verified that the six hydraulic snubbers discussed above were removed and returned to the Vendor's shop for inspection, repair and retest. The inspector also verified that the corrective actions outlined in the 10 CFR 50.55(e) Final Report dated March 16, 1976, have been accomplished. Fluid level plunger indicator plates recommended by Rexnord Company are now installed.

Unresolved item 75-19/1 is closed.

3. Preparations For Initial Criticality

Enclosure 1 to St. Lucie Plant Unit No. 1 License No. DPR-67 required certain items to be completed to the satisfaction of the NRC prior to achieving initial criticality. Item A-1 of the enclosure required the installation of tornado missile protection for the following systems:

a. Intake Cooling Water Pumps

b. Component Cooling Water Pumps

c. Diesel Generator Air Intakes and Exhausts

d. Diesel Generator Access Doors

e. Diesel Generator Fuel Oil Pumps

f. Auxiliary Feedwater Pumps

The inspector verified that the required tornado missile protection was completely installed prior to achieving initial criticality by physically inspecting in the plant each of the above listed systems.

Item A-2 of Enclosure 1 required the installation of a temporary tornado-protected connection between the spent fuel pool and the charging system to provide makeup water to the reactor coolant system to accommodate moderator shrinkage upon plant shutdown. The inspector verified by personal observation that a tornado-protected connection was in place in the reactor auxiliary building prior to initial critical.

Item A-5 of Enclosure 1 required revision of the sequencing document for "Post Core Load Hot Functional" testing to include provisions for retesting of hangers and restraints. The inspector verified that the sequencing document was revised and that also preoperational test No. 0010194 "Piping Thermal Expansion and Restraint Post Core Load" contained additional sections which covered monitoring all mechanical and hydraulic snubbers and piping restraints during post core load heatup to rated conditions. Unresolved item 75-19/1 is closed. Refer to discussion in Details II paragraph 2 of this report.

Item B-1 of Enclosure 1 required the installation of control circuitry providing the capability to energize and de-energize the ECCS Miniflow bypass valve operators (V-3659 and V-3660) from the control room. The inspector verified that the switches were installed in the control room and that they had been functionally tested for proper operation.

4. Initial Criticality

St. Lucie Unit 1 reached initial critical at 0830, EST on April 22, 1976. Preoperational Procedure No. 3200086 was followed to position the control element assemblies (CEA's) in their normal sequence to produce an essentially rod free core. The procedure was then used to direct deboration to criticality. Critical occurred as predicted when the RCS was diluted to approximately 938 ppm boron. There were no discrepancies identified by the inspector during the approach to initial critical. The items which the inspector witnessed as part of initial criticality are summarized below:

a. Prior to the start of CEA withdrawal the licensee met the applicable technical specification requirements of Sections 3/4 (Limiting Conditions for Operation and Surveillance Requirements), Section 6.2 (Organization), Section 6.8 (Procedures) and Section 6.10 (Record Retention).

b. The required nuclear instrumentation was operating and was properly calibrated. OP No. 1400050 and Plant Work Orders 68441 and 69557 were reviewed for verification of calibration of the wide range monitors. Instrument calibration data sheets 229 and 230 were reviewed for verification of calibration of the Eberline startup scalers. The FSAR Section 14.1.4.4 requirement of at least 1/2 count per second minimum count rate was satisfied.

c. Trip checks of the wide range nulcear instrument channels was verified complete by reviewing OP 1210051.

d. Properly approved procedures were used to provide a safe, organized method for attaining initial criticality. The count rate and dilution data were properly recorded and the resultant 1/m plots developed as expected.

e. The prerequisites required prior to the start of CEA withdrawal were verified complete by reviewing appendix 'A' OP 0030122 Reactor Precritical Check List.

f. During dilution the CEA pattern was independently verified. All CEA groups were withdrawn except group 7 which was at 68 inches withdrawn. This pattern agreed with the approved configuration.

g. One coolant system boron sample and analysis during dilution was observed. The sample was taken and properly analysed according to FPL Chemistry Department OP C-35B.

h. During dilution the "METRA SCOPE" used for CEA position indication was lost for 30 minutes. The inspector observed that the licensee stopped dilution during the period the "METRA SCOPE" was out of service.