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

Report No.: 50-302/89-30

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 27 - December 1, and December 11 - 15, 1989

Inspector: John Zeiler 1/29/90
J. Zeiler Date Signed

Approved by: G A Belisle 1/30/90
G. A. Belisle, Section Chief Date Signed
Test Programs Section
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This routine unannounced inspection was conducted in the areas of the containment local leak rate testing, verification of containment integrity, and licensee action on previous inspection findings.

Results:

A violation was identified in the inservice testing program involving the setting of reference values for the reactor building spray pumps. This was a result of failing to establish pump inservice test procedures which incorporated the design requirements contained in the Final Safety Analysis Report (FSAR), paragraph 4.c. A weakness was identified involving the licensee not assuring that other safeguards pumps were being tested within acceptable design requirements, paragraph 4.c.

In general, the inspection results indicate a continuing good performance by the licensee in the area of containment local leak rate testing. However, one weakness was identified involving the lack of containment penetration draining instructions. The licensee committed to incorporate draining instructions for each penetration tested into the local leak rate test procedure after the upcoming refueling outage.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *J. Alberdi, Nuclear Plant Technical Support Manager
   *G. Becker, Nuclear Compliance Manager
   *D. Black, Nuclear Results Specialist
   L. Clewett, System Engineer
   *J. Cooper, Jr., Nuclear Technical Support Supervisor
   *G. Cowles, Senior Nuclear Results Engineer
   *K. Lancaster, Site Nuclear Quality Assurance (QA) Manager
   J. Maseda, Corporate Nuclear Engineering Supervisor
   *P. McKee, Nuclear Plant Operations Director
   *W. Rossfield, Nuclear Compliance Manager
   *M. Williams, Nuclear Regulatory Specialist
   *K. Wilson, Nuclear Licensing Manager

   Other licensee employees contacted during this inspection included engineers, operators, QA inspectors, technicians, and administrative personnel.

   NRC Resident Inspectors

   *W. Bradford, Resident Inspector
   *P. Holmes-Ray, Senior Resident Inspector

   *Attended exit interview

2. Containment Local Leak Rate Testing (61720)

   The purpose of the inspection activities in this area was to ascertain that the licensee's local leak rate test (LLRT) program was being conducted in compliance with NRC requirements. The inspector reviewed LLRT procedures, evaluated test results, and reviewed containment isolation valve (CIV) maintenance records.

   a. LLRT Procedure and Administrative Control Review

      The inspector examined the following surveillance procedures:

SP-177	Local Leak Rate Test of AHV-1A Thru 1D
SP-179A	Containment Leakage Test Types B and C
SP-179B	Containment Leakage Test Type B
SP-179C	Containment Leakage Test Type C
SP-181	Containment Air Lock Test
SP-430	Containment Air Locks Seal Leakage Test

The inspector verified that the following attributes were included in these procedures to ensure adequate leak rate testing of containment isolation boundaries:

(1) All required containment penetration boundaries and CIVs were included in the LLRT program.

(2) LLRTs were performed at containment integrated leak rate test (CILRT) peak design pressure.

(3) The LLRT program utilized approved methods for testing containment penetration boundaries and CIVs.

(4) Penetration leakage rates were determined using the maximum pathway leakage

(5) The criteria and response for LLRTs and combined leakage rate failure were incorporated in the test program procedures.

A detailed review was performed for Type C classified CIVs in the following penetrations to verify adequate alignment for venting and draining, and adequate boundary alignment for leak rate testing:

Penetration	33[illegible]	Letdown line to purification demineralizer
Penetration	339	Reactor building sump to miscellaneous waste storage tank
Penetration	349	Reactor coolant drain tank vent
Penetration	354	Reactor coolant equipment vents
Penetration	374	Reactor coolant drain tank drain
Penetration	377	Reactor coolant pump seals bleed-off
Penetration	439	Pressurizer and reactor coolant sample lines
Penetration	440	Steam generator 3A sample line
Penetration	441	Steam generator 3B sample line

Review of the above procedures and penetrations indicated a weakness in the licensee's LLRT program with regard to penetration venting and draining control. Prior to leak rate testing a valve, the test boundary is drained of all fluid to ensure that no artificial fluid barriers are erected. Venting and draining piping adjacent to the isolation valve under test must be accomplished by adequate controls which ensure that the proper test medium at the correct differential pressure is obtained across the isolation valve. The licensee's procedure for Type C LLRTs, SP-179C, did not provide the level of detail necessary for the inspector to verify and ensure that adequate penetration draining would be accomplished for all leakage tests. Step-by-step instructions of the draining process for each penetration tested were not included in the procedure. From discussions with leak rate test personnel who actually drain the penetrations before testing, the inspector determined that test personnel understand the importance of penetration draining and follow proper draining techniques. At the exit meeting the licensee agreed to incorporate draining instructions for each penetration tested into SP-179C after the upcoming refueling outage. Since the identified problem is a procedural weakness and not a case of actual failure to drain a penetration, the inspector considered the licensee's proposed action acceptable.

The inspector reviewed a sample of the completed "As-Found" and "As-Left" Type B and C LLRT results for the past two outages as well as the corrective maintenance work performed on failed valves and calibration records of the test instruments. No unacceptable conditions were identified.

b. Review of Containment Purge Valve Design

IE Notice 88-73 dated September 8, 1989, identified a possible leakage problem related to the direction of pressurization for the Fisher Series 9200 butterfly valves used in containment purge lines. Florida Power Corporation (FPC) uses Henry Pratt butterfly valves in the containment purge lines. The valve design and leak rate test configuration was reviewed by the licensee and the valve vendor; the results of this review were reported in an FPC letter dated May 23, 1989. The vendor reported that the valve's pressure retention capability should be slightly greater from the shaft side of the valve, due to the tendency of the valve disc to compress into the tapered rubber seat under pressure. FPC performs LLRTs on these valves by pressurizing between them, with both shafts external to the test boundary. Therefore, the inboard purge valve is tested in the non-accident direction, but this is conservative since accident pressure would be in the valve's pressure assisted direction. The outboard valve is tested in the accident pressure direction. The inspector considered that the licensee's current LLRT configuration for these valves to be acceptable and meets the requirements of 10 CFR 50, Appendix J.

c. Leak Rate Test Maintenance Controls

The inspector tracked the repair and retest of 12 CIVs to determine if controls to ensure maintenance and retest of the valves were adequate. All work requests written since 1984 for these valves were reviewed. No unacceptable conditions were identified. The inspector concluded that the licensee has implemented a workable system to ensure that maintenance and retest of CIVs are satisfactorily completed.

The inspector also discussed with the licensee how CIV LLRT data was analyzed or trended in order to detect early valve degradation or to predict valve failures. Currently, no formal program has been established for this particular purpose. The inspector was told that past LLRT data is reviewed and analyzed by the LLRT coordinator. From review of the last outage test data, the inspector found cases in which prior planning was conducted to replace or work valves which had failed leak rate tests or had higher than normal leakage results from previous outage testing.

d. Personnel Training and QA Coverage

The inspector discussed the qualifications and training for LLRT personnel and reviewed training records for selected personnel. Test personnel interviewed were knowledgeable of their responsibilities and technical aspects of leak rate testing.

The inspector also discussed coverage of local leak rate testing with QA representatives and reviewed QA surveillance and audit reports. It was determined that QA has a good auditing program in place for monitoring the overall LLRT program as well as providing visual coverage of a small sample of LLRTs. No unacceptable conditions were identified.

Within the areas inspected no violations or deviations were identified.

3. Verification of Containment Integrity (61715)

The purpose of the inspection activities in this area was to verify the adequacy and implementation of procedures and controls designed to maintain containment integrity and to mitigate contamination releases in the event containment integrity is lost following a loss-of-coolant accident (LOCA).

a. Primary Containment Integrity Controls

The inspector reviewed Operating Procedure OP-202, Plant Heatup, and SP-440, Unit Startup Surveillance Plan, which together ensure all necessary plant conditions are established and that prerequisites are met for reactor startup. The inspector verified that the procedures included the following minimum provisions that ensure primary containment integrity exists before the plant enters operational modes which require containment integrity:

(1) All penetrations required to be closed during accident conditions are closed by operable automatic valves or closed by manual valves, blind flanges, or deactivated automatic valves.

(2) All equipment hatches are closed and sealed.

(3) Each containment airlock is operable.

(4) Containment leakage rates are within technical specification (TS) limits.

(5) Sealing mechanisms associated with each penetration are operable.

The inspector also reviewed SP-341, Monthly Containment Integrity Check, which provides assurance of primary containment isolation by verifying that all manual valves, blind flanges, and deactivated automatic valves are closed and locked as required. The inspector verified that the procedure included all appropriate barriers. Completed records for SP-341 were reviewed over the previous four months of reactor operation. The inspector verified that all valves were inspected and found to be in their correct position.

b. Containment Systems Designed to Mitigate Contamination Releases

The following containment related systems designed to mitigate the consequences of contamination releases following a LOCA were inspected for compliance with plant TSs:

Containment structural integrity
Reactor building spray system
Spray additive system
Containment cooling system
Hydrogen purge system

The inspector reviewed the following surveillance procedures and verified that the procedures complied with applicable plant TS requirements, that adequate information and instruction were provided, and that adequate acceptance criteria and limits were specified:

SP-182	Reactor Building Structural Integrity Tendon Surveillance Procedure
SP-178	Containment Leakage Test - Type A
SP-347	ECCS and Boration System Flow Path
SP-169C	Decay Heat Removal/Building Spray Instrumentation
SP-456	Refueling Interval Equipment Response to an ESAS Test Signal
SP-412	ECCS and Containment Spray System Leak Rate Test
SP-184A	Sodium Hydroxide Flow Verification: Train A
SP-184B	Sodium Hydroxide Flow Verification: Train B
SP-344C	Nuclear Services Containment Cooling System Supply Operability
SP-335C	Radiation Monitoring Instrumentation Functional Test
SP-185	Reactor Building Ventilation Exhaust System Testing
SP-189	Charcoal Test Canisters
SP-443	Master Surveillance Plan

The inspector reviewed surveillance records listed in Table 1 below and verified that the surveillances were performed at the required frequencies, test results met acceptance criteria or limits, and appropriate sign-offs, test reviews, and test concurrences were performed.

Table 1

Containment System	Procedure No.	Records Reviewed	TS
Structural	SP-182	All Records	4.6.1.6.1/2/5
Integrity	SP-178	07/16/83 and 11/11/87	4.6.1.6.3/4
Reactor	SP-347	?6/01/89 to 11/01/89	4.6.2.1.a
Building			4.6.2.2.a
Spray and	SP-169C	09/30/87 to 05/17/89	4.6.2.1.b.1
Additive			4.6.2.2.b.1
	SP-456	12/23/87 to 05/03/89	4.6.2.1.b.1/2
			4.6.2.2.c
	SP-412	09/20/87 to 08/01/89	4.6.2.1.c.1/2
	SP-184A	04/03/89	4.6.2.2.d
	SP-184B	04/15/89	4.6.2.2.d
Containment	SP-344C	10/11/89 to 11/21/89	4.6.2.3.a
Cooling	SP-456	12/23/89 to 05/03/89	4.6.2.3.b
Hydrogen	SP-335C	08/18/89 to 10/20/89	4.6.4.2.a
Purge	SP-185	12/21/87 to 05/08/89	4.6.4.2.b.1/3
			4.6.4.2.d/e/f
	SP-189	10/02/89	4.6.4.2.b.2/c

c. Reactor Building Spray System Walkdown

The inspector conducted a walkdown of portions of the reactor building spray and spray additive systems located outside containment. All valves were verified to be in their required position for proper operation of these systems and both trains appeared operational. In addition, all areas inspected were generally clear and free from debris. No unacceptable conditions were identified.

The inspection findings indicated that required plant systems and components designed to ensure containment integrity were being tested as required by plant TSs.

Within the areas inspected, no violations or deviations were identified.

4. Followup on Previous Inspection Findings (92701)

a. (Closed) Inspector Followup Item (IFI) (302/87-38-01): Determine Pass/Fail Status of CILRT Performed on November 9, 1987

On November 9, 1987, during an attempted CILRT a problem was identified with the integrity of the secondary side of Steam Generator A. The CILRT was aborted and the licensee discovered a leak at the upper hand-hole cover in the steam generator. The licensee determined that the hand-hole cover gasket had been damaged due to improper cover installation. The hand-hole cover was installed after steam generator maintenance was performed early in the refueling outage. The gasket was replaced and a successful CILRT was performed by the licensee. The licensee concluded that the gasket problem would have led to a large feedwater leak during startup, had it not been discovered during the CILRT. Therefore, the unit would not have started up without discovering or first correcting the deficiency.

Upon review, the inspector was confident that the improperly installed gasket would have been detected and corrected prior to unit operation. Based on this, the inspector agreed with FPC's position that this was an aborted test attempt, and not a failed CILRT. As corrective action, the licensee's procedure for installing the hand hole covers, MP-110, OTSG Maintenance, was revised to provide additional guidance for installing the covers. Further, the CILRT procedure, SP-178, was revised with the addition of a prerequisite step which requires maintenance personnel to verify that steam generator covers have been properly reinstalled if removed between CILRTs.

b. (Closed) IFI (87-38-02): Evaluation of Containment Liner Weld Channels

During NRC witnessing of the licensee's CILRT conducted November 9, 1987, it was observed that most containment weld channels were not vented to the containment atmosphere. Unless these weld channels are vented to the containment during the CILRT, they, in effect, represent an artificial barrier which may prevent detection of leakage in the containment boundary. Containment weld channels were constructed over the containment liner welds to allow pressurization and leak testing of the liner welds during construction of the containment. Many of the test connections to the liner weld channels are in inaccessible or difficult to reach locations. The NRC position in this matter has been that, generally, containment liner weld channels should be vented to the containment atmosphere during the CILRT. However, if the licensee can demonstrate that the channels would maintain their integrity when subjected to the loading conditions of a postulated design basis accident (DBA), as well as during normal operation, the channels need not be vented. The licensee submitted a letter to the NRC dated December 22, 1987, concluding that containment integrity would be maintained during the DBA with the channels in either the vented or unvented condition. Region II plans to request the Office of Nuclear Reactor Regulation (NRR) to evaluate the capability of the weld channels at Crystal River to maintain their integrity when subjected to loading conditions characteristic of a DBA. The licensee was advised that further channel information may be requested by NRR during their review. No further action is required by Region II at this time; therefore, this item is considered closed.

c. (Closed) Unresolved Item (URI) (88-05-01): Low Developed Head from Reactor Building Spray Pump BSP-1A

During an NRC inspection conducted January 11-15, 1988, a discrepancy was noted between the manufacturer's head-flow curve and the licensee's test data for reactor building spray pump BSP-1A. From subsequent testing, the licensee determined that BSP-1A would only provide a flow of 1460 gpm at a head of 375 feet which was below the original design rating. A minimum reactor building spray flow of 1500 gpm at a total developed head of 450 feet was assumed for the original reactor building design basis LOCA analysis.

The licensee performed a safety evaluation, summarized in letter dated February 24, 1988, from which calculations demonstrated that a minimum flow rate of 1200 gpm at a head of 382 feet would satisfy all design requirements for containment pressure, iodine removal, pH control, and equipment qualifications. The 1200 gpm flow rate was also confirmed from calculations performed by Gilbert Commonwealth and Babcock and Wilcox. The licensee considers the pumps operable based on these redesign calculations.

The inspector reviewed pump inservice testing data for BSP-1A and noted that a decrease in pump performance occurred after the pump was overhauled in October 1987. The work package for this overhaul indicated that the pump impeller was replaced. Test data taken after the pump maintenance showed that the pump differential pressure dropped from 180 psig to 165 psig. The acceptance range for the pump differential pressure was set at 165.5 - 181.6 psig. No action was taken by the licensee to determine the cause of the decrease in pump performance or verify that the new reference values represented acceptable pump operation. Licensee Event Report (LER) 88-007, dated April 6, 1988, reported that for corrective action, a thorough investigation and mechanical inspection of BSP-1A would be performed to determine the cause of the reduced flow capacity and action would be taken to restore the pump to its initial design capacity. The inspector was told that the corrective action for BSP-1A was scheduled for the next refueling outage, in March 1990.

The inspector reviewed surveillance procedures 340-A and 340-B which are used to conduct the ASME Section XI inservice pump tests for the reactor building spray pumps. The acceptable range for pump differential pressure was specified as 165.5 - 181.6 psig. The original FSAR design basis analyses assumed a reactor building spray flow rate of 1500 gpm with a differential pressure of 195.06 psig. The inspector determined that the procedure did not provide appropriate acceptance criteria which reflected the FSAR design requirements for these pumps. As a result, FSAR design basis requirements were exceeded without being identified. This failure of the inservice test (IST) procedures to incorporate appropriate FSAR design limits has been identified as violation 50-302/89-30-01. This URI is closed by the identification of the violation.

The licensee also committed in LER 88-007, to review head-flow surveillance data for other engineered safeguards pumps and verify that their acceptance design tolerances were met. The inspector found no evidence that this action was ever performed. The licensee's Engineering Department has been tasked to provide minimum design requirements for all IST pumps. This would entail possible reevaluation of system requirements. However, it appears that the licensee has not performed a review to assure that other safeguards pumps are operating within their existing design basis requirements. The inspector considered this to be a weakness in the licensee's corrective action regarding this issue.

5. Exit Interview

The inspection scope and results were summarized on December 15, 1989, with those persons indicated in paragraph 1. The inspector described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report.

Item Number	Description and Performance
50-302/89-30-01	Violation - Failure of IST procedures to incorporate appropriate FSAR design limits for reactor building spray pumps, paragraph 4.c.