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

Report No.: 50-302/92-27

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

Docket No.: 50-302 License No.: DRP-72

Facility Name: Crystal River 3

Inspection Conducted: October 18 - November 14, 1992

Inspector:	A.R. Long for P. Holmes-Ray, Senior Resident Inspector	12/11/92 Date Signed
Inspector:	A.R. Long for R. Freudenberger, Resident Inspector	12/11/92 Date Signed
Inspector:	N. Merriweather N. Merriweather, Reactor Inspector	12-11-92 Date Signed
Approved by:	M V Sinkule K. Landis, Section Chief Division of Reactor Projects	12/11/92 Date Signed

SUMMARY

Scope:

This routine inspection was conducted by two resident inspectors and one specialist inspector in the areas of plant operations, security, radiological controls, Licensee Event Reports, plant modifications, and licensee action on previous inspection items. Numerous facility tours were conducted and facility operations observed. Backshift inspections were conducted on October 24, 25, 27, 31, and November 4, 6, 7, 11.

Results:

In the area of plant operations, the following violations were identified:

VIO 50-302/92-27-01: Failure to Follow Procedure Results in Valve Misalignment and Reactor Building Spray (paragraph 3.a)

VIO 50-302/92-27-02: Failure to enter action statement 3.8.1.1 and perform surveillance requirement 4.8.1.1.1.a with the "B" Emergency Diesel Generator inoperable as the result of non-TS surveillance testing (paragraph 4.a).

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REPORT DETAILS

1. Persons Contacted

Licensee Employees

G. Boldt, Vice President Nuclear Production
*J. Buckner, Nuclear Regulatory Specialist
*R. Davis, Manager, Nuclear Plant Maintenance
*E. Froats, Manager, Nuclear Compliance
*A. Gelston, Manager, Site Nuclear Engineering Services (Acting)
*G. Halnon, Manager, Nuclear Plant System Engineering
B. Hickle, Director, Nuclear Plant Operations
*S. Johnson, Nuclear Chemistry and Radiation Protection Superintendent
*K. Lancaster, Nuclear Maintenance Work Controls Superintendent
*G. Longhouser, Nuclear Security Superintendent
W. Marshall, Nuclear Operations Superintendent
*P. McKee, Director, Quality Programs
*L. Moffatt, Nuclear Shift Manager
B. Moore, Manager, Nuclear Integrated Scheduling
*D. Porter, Nuclear Shift Supervisor
*S. Robinson, Manager, Nuclear Quality Assessments
*W. Rossfeld, Manager, Site Nuclear Services
*J. Terry, Supervisor, Site Nuclear Engineering Services
*R. Widell, Director, Nuclear Operations Site Support
K. Wilson, Manager, Nuclear Licensing

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

NRC Resident Inspectors

*P. Holmes-Ray, Senior Resident Inspector
*R. Freudenberger, Resident Inspector

*Attended exit interview

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

2. Plant Status and Activities

The plant continued in power operation (Mode 1) for the duration of this inspection period.

On October 21, the Chief of Region II Reactor Projects Section 2B was on site for a routine visit.

During the week of November 2, a specialist inspection was conducted to observe the Emergency Drill. The results of this inspection were documented in NRC Inspection Report 50-302/92-26.

3. Plant Operations (71707, 93702, 40500, & 82301)

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

a. Inadvertent Reactor Building Spray

On October 15, 1992, with the unit at full power and a quarterly surveillance run of the "A" Building Spray Pump in progress, a valve misalignment resulted in the discharge of borated water into the Reactor Building. A detailed description of the event and licensee initial actions was included in NRC Inspection Report 50-302/92-25, detail 3.a. Unresolved Item 50-302/92-25-01 was identified pending the completion of licensee evaluations and the planned submittal of a voluntary report on the event.

Inspector review of the circumstances of the event concluded that the root cause was licensed operator error. The surveillance flow path was to be from the Borated Water Storage Tank (BWST) through the BSP-1A recirculation line back to the BWST. Step 4.0.5 of SP-340B aligns recirculation flow for testing the "A" Reactor Building Spray Pump, including prepositioning manual valve BSV-28 at approximately 7 and 1/2 turns open. After the pump has been started, step 4.7.1 states "Establish flow at 1500 gpm (with allowable oscillation averaged value between 1470 gpm and 1530 gpm), by throttling BSV-28." The licensed operator mistakenly opened and throttled with BSV-3, the motor operated valve to the spray nozzles, allowing flow to the Reactor Building spray header.

The licensed operator involved did not use proper self-checking techniques while performing the surveillance. Nevertheless, he volunteered his error when he recognized it and candidly assessed his own performance. Contributing factors to the error included the wording and structure of the procedure step and simulator training that practiced throttling of Reactor Building spray flow using BSV-3 during accident conditions.

The licensee performed an immediate assessment of plant conditions and developed an action plan to recover from the reactor building spray event. A preliminary evaluation was performed by the licensee in the following areas:

- Wetted areas of the RB

- Metal corrosion, including galvanic corrosion, boron corrosion and hydrogen production
- RB instrumentation condition
- Electric motors including:
    - RCP motors
    - AHF motors
    - Motor operated valves
    - CRDM stators
    - Cable issues
- Primary system thermal stress
- RB cranes
- Comparison to RB hydrolysing experiences
- Fan cooler effects of boron precipitation
- EQ equipment impact assessment

The NRC's overall conclusion was that no immediate safety concern existed and that continued operation was justified. The short term actions consisted primarily of increased monitoring of equipment and instrumentation for indications of degradation caused by the spray, and a RB entry to assess the condition of the RB and equipment. The Resident Inspector participated in the RB entry. No damage to safety-related equipment as a result of the spray was identified. The long term actions required additional visual inspections of equipment for signs of degradation due to spray.

A region-based inspector was dispatched to the site on October 16, to aid in the review of licensee's actions to recover from the inadvertent actuation of Building Spray. The inspector met with the Engineering Manager to discuss the event and the licensee's action plan status. The action plan contained 14 action items of which most were considered complete. The open items were identified as follows:

- Walkdown main control board thoroughly and frequently to look for indication of failed instruments or components.

  Status: Ongoing on October 16, now complete.

- Accumulate a list of known damage and effects.

  Status: Ongoing. As of October 16, only two failures had occurred and they were discussed in NRC Inspection Report 50-302/92-25. Two more Control Rod Drive Shroud Fans subsequently failed, which might be attributable to the spray event, as described below.

- Determine what actions were taken at other plants having similar events.

  Status: Ongoing on October 16, now complete.

The licensee's action plan was judged to contain action items that were similar to those described in a Safety Evaluation (SE) developed by the NRC Division of Systems Technology concerning "Inadvertent Containment Spray Events at Commercial Nuclear Power Plants." The SE was issued by an NRC memorandum from Thomas Murley, Director, Office of Nuclear Reactor Regulation, to all Regional Administrators, dated March 13, 1991. This SE was attachment A to NRC Inspection Report 50-302/92-25. The safety evaluation indicated that continued reactor operation following a spray event was acceptable if an immediate assessment of plant conditions was performed and an action plan was developed to fully evaluate the consequences of the event on plant systems and components.

Inspection of short term actions taken by the licensee was performed. Increased surveillance of instruments and equipment for degradation were observed and the engineering assessments of potential problem areas were evaluated. One licensee action item required operations to walk down the main control board twice per shift to look for indication of failed instruments or components. A Short Term Instruction (STI) 92-019 was issued which required performance of SP-300, Operating Daily Surveillance Log, twice per shift for one week, until October 22, 1992. The NRC inspector toured the control room on October 17 and 18, 1992, attended shift turnover meetings, and witnessed the control room board walkdowns at 8:00 a.m., 12:00 noon and 4:00 p.m. The data sheets for SP-300, completed on October 17, 18 and 19, 1992, and the operating shift log were reviewed to verify that no equipment failures had occurred that could be attributed to the BS event. During the week of additional monitoring, no other failures were identified. Other licensee action items required engineering to perform evaluations of potential problem areas as well as provide recommendations for short and long term action. The preliminary evaluations performed by the licensee were reviewed and found to be acceptable.

As discussed in NRC IR 50-302/92-25, immediately following the spray, one of twelve control rod drive shroud exhaust fans (AHF-51A) and one area radiation monitor (RMG-18) failed. Later in this report period, two more of the control rod drive fans (AHF-51D & AHF-51K) failed. The effect of these failures on reactor head service structure temperature was negligible. Cause analysis of the recent failures was added to the licensee's long term action plan, to determine if these failures were the result of the spray.

The inspector confirmed that the licensee had a program in place to perform increased surveillance of instruments to identify degradation of components. The licensee's long term action plans, to assess equipment for possible damage due to spray, appeared to be appropriate. The plan included such actions as visual inspections of selected plant equipment and increased sensitivity to abnormal component conditions during routine maintenance and calibration in future outages.

The licensee submitted a voluntary report on the event in a letter dated November 4, 1992, to the Regional Administrator, NRC, Region II. The letter described the event and addressed the scope and sequence of licensee actions. Actions performed or planned included a human performance evaluation and technical evaluation of the short and long term affects. The licensee's human performance evaluation concluded that the operator failed to apply proper self-checking measures to assure adequate comprehension and correct implementation of required procedural guidance. The human performance evaluation proposed several recommendations for consideration to prevent recurrence.

The operator's failure to properly implement the Step 4.7.1 of Surveillance Procedure SP-340B was a violation, and was identified as VIO 50-302/92-27-01: Failure to Follow Procedure Results in Valve Misalignment and Reactor Building Spray. Although the licensee promptly initiated extensive corrective actions, as described in detail above, the violation is being cited because of the inherent significance of an operator error of this nature. Unresolved item 50-302/92-25-01 is closed.

b. Emergency Drill Observation

On November 5, the licensee conducted an annual Emergency Preparedness exercise. Specialist Inspectors from the Region II office of the NRC were onsite to evaluate the exercise. The resident inspectors observed activities in the Simulator Control Room, The Technical Support Center, and the Emergency Operations Facility.

This was the first occasion in which the control room simulator was used to simulate accident conditions and provide realistic time frames for emergency condition parameters.

Late in the scenario, the operators attempted to use a recently installed high pressure auxiliary spray line to continue a rapid cooldown and depressurization of the reactor coolant system following the simulated steam generator tube rupture. Although the simulator had been updated to provide simulation of use of the high pressure auxiliary spray, the operators identified that an out-of-date revision of procedure OP-305, Operation of the Pressurizer, was in the simulator control room and did not include a section for use of the high pressure auxiliary spray. The procedure was replaced with a current revision. The licensee's drill critique identified the deficiency and the inspector verified that the correct revision of the procedure was in the plant control room.

An assessment of the overall effectiveness of the exercise is documented in NRC Inspection Report 50-302/92-26.

4. Maintenance and Surveillance Activities (62703 & 61726)

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

The following tests were observed and/or data reviewed:

- SP-907B, Monthly Functional Test of 4160V ES Bus "B" Undervoltage Relaying;
- SP-907A, Monthly Functional Test of 4160V ES Bus "A" Undervoltage Relaying; and
- SP-317, Reactor Coolant Water Inventory Balance.

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

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

- WR 302973, Installation of Phase Gating Unit for Phase B for Control Rod Group 6A Power Supply;
- WR 262140 & 299344, Preventive Maintenance on Inverter 1A;
- WR 302712 Troubleshooting Inverter 1E intermittent output voltage fluctuation;
- WR 303112 & 303086, Cleaning and Ultrasonic Inspection of Nuclear Services Closed Cycle Cooling System Suction Header; and
- WR 290604, Prefabrication of Replacement Nuclear Services Closed Cycle Cooling Suction Header Piping.

The following items were considered noteworthy:

a. Technical Specification Action Statement Applicability During Surveillance Testing

On October 27, 1992, the licensee performed SP-907B, Monthly Functional Test of 4160V ES Bus "B" Undervoltage Relaying. The purpose of the procedure was to demonstrate operability of the 4160V ES Bus B undervoltage protection scheme. The undervoltage protection scheme is comprised of three undervoltage modes: First Level Undervoltage Relays (loss of voltage); Second Level Undervoltage Relays (degraded voltage); and Second Level Undervoltage with an ES actuation. Each mode is capable of stripping the 4160V ES Bus B and initiating a start of Emergency Diesel Generator 3B. The performance of SP-907B is <u>not</u> required by TS.

Performance of the procedure included disabling the Emergency Diesel Generator by tripping the fuel rack and isolating starting

air to prevent an inadvertent start. Degraded voltage conditions were then simulated to actuate the FLUR and SLUR relays and measure their response time. The inspector observed the test and equipment restoration, which was performed satisfactorily.

The inspector noted that although the "B" Emergency Diesel Generator was disabled for approximately two hours as directed by the procedure, the licensee did not enter TS action statement 3.8.1.1 nor perform action b, performance of surveillance requirement 4.8.1.1.1.a, which requires verification of the breaker alignments and indicated power availability to the offsite circuits within one hour. This was identified as VIO 50-302/92-27-02: Failure to enter action statement 3.8.1.1 and perform surveillance requirement 4.8.1.1.1.a with the "B" Emergency Diesel Generator inoperable as the result of non-TS surveillance testing.

The safety significance of this violation was minor because offsite power circuits remained operable throughout the performance of the SP. In addition, the Emergency Diesel Generator was inoperable for significantly less than the 72-hour time period allowed by the action statement before shutdown must be initiated.

Review of the circumstances of the above violation revealed that historically, the licensee has not entered TS action statements when TS related equipment became inoperable as the result of performing SPs. It was recognized that this practice was not consistent with the NRC staff's position stated in Part 9900: Technical Guidance - Operability of the NRC Inspection Manual published in Generic Letter 91-18 "Information to Licensee's Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," dated November 7, 1991. This guidance states "If TS surveillances require that safety equipment be removed from service and rendered incapable of performing its safety function, the equipment is inoperable. The LCO action statement shall be entered unless the TS explicitly direct otherwise." However, some surveillance requirements such as RPS testing which removes a RPS channel from service cannot be completed within the action statement time allowance.

FPC is currently examining their position on entering the required actions of TS when equipment is rendered inoperable as a direct result of periodic surveillance testing based on their review of NRC GL 91-18 and implementation of the Revised Standard Technical Specifications (RSTS).

A licensee written position on the issue provided to the inspectors following discussions with FPC management states that FPC "understands this issue was discussed at the Region I

NRC/Public workshop on GL 91-18 and that as the result of issues identified at that time, it is the NRC Staff's intent to further discuss it at subsequent GL 91-18 workshops. FPC intends to remain involved in these discussions and work, together with other utilities, towards a resolution that provides adequate operational latitude while maintaining full cognizance of equipment availability/operability." RSTS accommodates implementation of the NRC Staff's position and is currently scheduled for implementation in the Fall of 1993.

b. Reactor Coolant Water Inventory Balance

The reactor coolant drain tank level indicator (WD-23-LT) has experienced sporadic perturbations in indication since startup from the 1992 refueling outage. Due primarily to inaccessibility of the components during power operations, troubleshooting activities have not been successful at identifying and correcting the cause of the indicated perturbations.

The level change in the reactor coolant drain tank is monitored during the performance of SP-317, Reactor Coolant Water Inventory Balance, to determine the identified leakage in accordance with TS 3.4.6.2. Leakage from the reactor coolant system to the reactor coolant drain tank would originate from the pressurizer power operated relief valve, code safeties, core flood tank drains or reactor coolant system vents and drains. During initial troubleshooting efforts, reactor coolant leakage to the reactor coolant drain tank was considered in the unidentified leakage calculation.

STI 92-0021 was issued on October 19, 1992, after long term trending of the WD-23-LT indication. The STI allows use of WD-23-LT level indication for the calculation of identified leakage provided a review of the history of the output of WD-23-LT for a period including one hour before and after the collection of SP-317 data is reviewed and no perturbations exist.

Plans were established to conduct further troubleshooting in the event of an unplanned outage. The inspector reviewed SP-317 data from throughout the report period. Reactor coolant system leakage has remained stable and low following the 1992 refueling outage. The inspector considered the licensee's actions to address this issue to be acceptable.

c. Nuclear Services Closed Cycle Cooling Suction Header Degradation

On November 5, a maintenance planner performing a walkdown of a WR for replacement of the Nuclear Services Closed Cycle Cooling (SW) suction header identified leakage from the header in the vicinity of the joint of the six inch surge line to the eighteen inch suction header.

The suction header and the surge line are located in a trench in the floor of the Seawater Room in the Auxiliary Building. The suction header is common to three SW system circulating pumps. One pump is the normal duty pump, with two ES pumps normally in standby. The SW system provides cooling to safety related components such as; high pressure injection pumps, letdown coolers, reactor building coolers, spent fuel pool coolers, reactor coolant pumps, and control rod drive mechanism coolers. Decay Heat Removal is supplied by separate closed cycle cooling systems.

On November 6, the licensee performed a detailed visual inspection of the leaking section of pipe. Two additional pin hole type leaks were identified on the bottom of the suction header. A temporary non-code repair was designed and installed to reinforce the structural integrity of the leaking section adjacent to hanger SWH-18. The repair consisted of encasing approximately three feet of the surge line and suction header in high density grout, filling the trench in the area of the hanger SWH-18. The repair was installed in accordance with a temporary modification, MAR number T92-11-02-01. The MAR was reviewed and the PRC meeting that approved the MAR was attended by the inspector. Prior to installation of the temporary non-code repair, the NRC reviewed and approved the installation in accordance with GL 90-05, "Guidance For Performing Temporary Non-Code Repair of ASME Code 1, 2, and 3 Piping."

During the week of November 9, an ultrasonic inspection to characterize the condition of the remainder of the surge line and suction header, including the two additional leaks, was performed. Portions of this inspection were observed by the inspectors. Inspection results indicated that significant external corrosion had caused general wall thinning on the lower third of the eighteen-inch portion of the header and the full circumference of the six inch surge line in the trench. A stress analysis of the piping was completed by the licensee which indicated the piping remained operable and the leaks did not compromise the structural integrity of the piping. The inspectors reviewed the analysis and provided the information to NRC personnel involved in approval of the temporary non-code repairs. It was determined that the analysis provided sufficient justification that the piping remained operable with minimal margin for further degradation. The licensee submitted a request for approval of non-code repairs for the remaining leaks in a letter dated November 13.

An operations STI was issued on November 9, which directed the Primary Plant Operator to monitor the suction header at least twice per shift for any change in leakage. The STI remains effective until December 1, 1992.

Prefabrication of permanent replacement piping for the affected areas was ongoing at the end of the inspection period. The licensee's actions to monitor the condition of the SW will be assessed in future inspections.

5. Review of Licensee Event Reports (92700)

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

a. (Closed) LER 91-04: Personnel error caused reactor vessel level required for inadequate core cooling to be incorrectly calibrated.

See paragraph 6.b for details.

b. (Closed) LER 91-05: ECCS pump and flow paths were secured due to valve leak resulting in a condition outside the plant design basis.

Crystal River Unit 3 was operating at 100 percent full power on May 30, 1991. Between 9:00 p.m. and 9:07 p.m., operators received indications of leakage from the Makeup and Purification system in the Auxiliary Building. Operators located the leakage source by visual observations and by measuring piping surface temperatures. Leakage was originating from the inter-stage packing leakoff connection on MUP discharge crosstie valve, MUV-3. The leak was stopped at 10:23 p.m. by backseating the valve.

Between 9:42 p.m. and 9:55 p.m., while attempting the isolate the leaking valve, operators stopped all makeup flow, closed another discharge crosstie valve, MUV-4, the recirculation valve, MUV-264, and removed control power to the circuit breaker for one of the MUP/HPI Pumps. These actions placed the plant in a condition outside the design basis. This condition existed from 9:42 p.m. until approximately 11:05 p.m.. At this time, operators opened the valves and restored control power to the pump circuit breaker.

The FSAR for CR-3 discusses combinations of operable MUPs and HPI valves necessary to ensure adequate HPI flow in the event of an accident. With control power removed from the MUP-1C circuit breaker, the pump could not be started. Since MUP-1B was not aligned to automatically start, only one pump was available to provide HPI flow. With MUV-4 closed, only two nozzles were available. Based on information in the FSAR, it is not clear that this pump/valve combination would have provided adequate HPI flow under certain accident scenarios. Control Room personnel did not initially consider that these actions placed the unit in a condition outside the plant design basis. During subsequent review of the event, plant personnel determined that the unit was temporarily outside the plant design basis.

The licensee developed detailed guidance regarding HPI flow path operability documented in TSI 92-02. The failed valve packing was replaced during the 1991 maintenance outage with an improved design packing arrangement. This LER is closed.

c. (Closed) LER 91-10: Wiring problem causes transformer breakers to open actuating the emergency diesel generator.

On October 20, 1991, CR-3 was in Mode 5 (COLD SHUTDOWN) for a scheduled maintenance outage. At 2:43 p.m. the breakers for the offsite power transformer opened, disconnecting the ES busses from the offsite power supply. Decay heat cooling was interrupted for less than a minute while the emergency diesel generator loaded the ES bus and operators restarted the DHP. Upon starting the DHP, a purification relief valve lifted causing a drop in pressurizer level. Operators quickly identified the source and isolated the purification system. At 2:47 p.m., the operators manually energized the remaining ES bus via the CR-3 startup transformer. This event was caused by a pre-existing wire installation which inadvertently applied 115V AC to the CR-3 battery bus.

The improperly installed wire was removed by WR 0289924. The breaker relays were replaced with less sensitive ones by WR 0289619 to reduce the possibility of repeating this event. An Operations Study Book entry describing this event and lessons learned was issued November 7, 1991, and was signed off by all active operators. Procedure AP-770, Emergency Diesel Generator Actuation, was revised to address restarting the DHP if required following 4160V bus undervoltage. This LER is closed.

d. (Closed) LER 91-12: Procedure deficiency leads to inadvertent emergency diesel generator actuation during engineered safeguards testing.

The procedure inadequacy in SP-457, Refueling Interval ECCS Response to a Safety Injection Test Signal, was corrected by Revision 10 which was effective July 22, 1992. Some test switches are spring loaded to return to normal position some are not. The revision clearly required placing the test switches back to pre-test positions verses releasing the test switches. This LER is closed.

e. (Closed) LER 92-21: Lack of Required Lube Oil Leakage Collection Tank Reserve Capacity For Reactor Coolant Pumps Violates 10 CFR 50 Appendix R Design Criteria

The issue described in this LER was the subject of a violation detailed in NRC Inspection Report 50-302/92-25, detail 3.b. Corrective actions will be reviewed as part of the violation followup VIO 50-302/92-25-02, therefore this LER is closed.

Violations or deviations were not identified.

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

   a. (Closed) NCV 50-302/90-32-02: Failure to provide adequate design control measures as required by 10 CFR 50, Appendix B, Criterion III.

      On September 24, 1990, with the reactor at full power, the "area owner" was making a walkdown of the "B" decay heat pit. He noticed that the guard encapsulation for DHV-43 was not leak tight in that two small diameter pipe nipples did not contain plugs. The area owner, an operations assistant shift supervisor, wrote a memorandum to engineering requesting information as to whether the nipples should be open or plugged. On September 25, 1990, the system engineer, accompanied by a senior reactor operator, examined the guard enclosure for both decay heat trains and found the plugs missing on both decay heat train encapsulations. The system engineer referred operations to the FSAR chapter 5 which stated:

      "Penetrations 345 and 346 have only one containment isolation valve each. The second barrier, which is required in order to meet the CR-3 containment isolation design basis, is provided by an encapsulation around each recirculation line from the containment to beyond the first isolation valve. This encapsulation is leak-tight at containment design pressure and is not directly connected to the containment sump or atmosphere. A single passive or active failure in these lines or encapsulations will not provide a path for leakage to the environment."

      The drawings for the encapsulation (FPC Drawing Number PO-301-621) details the two socket weld adapters but does not show any attachment or closure devices. Therefore, the encapsulations are in accordance with the drawings. There is no test data, readily available, to indicate that the encapsulation is capable of withstanding containment design pressure. The licensee has plugged the openings but has not tested the encapsulation.

      A violation was considered for this issue because the fact that the encapsulations in the plant were in accordance with plant drawings but not in accordance with the design basis indicated that the design control measures required by 10 CFR 50, Appendix B, Criterion III were not effective.

      This was one of the issues discussed during an Enforcement Conference at the NRC Region II Office on October 31, 1990. The licensee presented information that the design basis of the encapsulations was different from that described in the current revision of the FSAR. Information from B&W correspondence from 1971 stated:

"... the intent of the guard pipe is to prevent excessive quantities of fluid escaping from the Reactor Building sump in the unlikely event a LOCA and subsequent rupture of this Decay Heat line were to occur."

B&W correspondence in 1987, relative to the function of encapsulations stated:

"The function of the piping jacket is to prevent drainage of the RB sump in the event of failure or breakage of the piping between the RB penetration and the isolation valves."

ANSI N271-1976 states the following relative to these containment penetrations:

3.6.4 Single valve and closed system outside containment:

> "The single valve and piping between the containment and the valve shall be enclosed in a protective leak tight or controlled leakage housing to prevent leakage to the atmosphere."

Inaccurate information relative to the definition of this type penetration was included in the 1989 revision of the FSAR.

Based on the information presented at the Enforcement Conference, the issue was categorized as a non-cited violation. The violation was held open to verify the revision of the FSAR to accurately describe the design basis of the encapsulations.

The current FSAR description of the design basis of the encapsulations was reviewed. It accurately described the design basis. This item is closed.

b. (Closed) URI 50-302/91-04-01: Review and approval of inadequate procedure steps for calibration of reactor vessel level instruments.

On April 24, 1991, after licensee followup of NRC questions, errors were identified in the calibration data sheets for reactor level instruments RC-163A-LT and RC-163B-LT, RC-164A-LT and RC-164B-LT, and RC-201-LT and RC-202-LT. The errors associated with the calibration status of reactor vessel head level instruments (RC-164A-LT and RC-164B-LT) was evaluated to be a condition outside the design basis as referenced in FSAR section 7.3.4.2.1 and TMI Action Plan Item II.F.2. Therefore, the licensee made a one hour ENS call on May 14, 1991, and submitted LER 91-04 on June 13, 1991.

The licensee attributed the calibration errors to FPC engineering personnel who incorrectly transposed calibration data from a contractor instrument data sheet. The document apparently did not clearly indicate instrument zero. This resulted in the RC-164A-LT and RC-164B-LT level devices measuring the reactor vessel head stand pipe level rather than the level from hot leg to reactor head. A modification was implemented in August 1985, (MAR 83-03-04-02) to meet NRC Order requirements which added these new reactor vessel level monitoring devices.

The licensee verified the instrument calculations and data sheets, and revised the appropriate SPs (SP-144A, RCITS Reactor Vessel and Hot Leg Level Channel Calibration; and SP-195, Remote Reactor Vessel Level Instrumentation Calibration). WRs 283679 and 283680 recalibrated the affected instruments on May 17, 1991 and May 18, 1991, respectively.

Problem Report SYPR-91-0012 was issued to document this issue, to determine root cause and to recommend corrective actions. The licensee determined the level instruments to be operable and therefore the associated TS action statements (TS 3.3.3.6.b and c for TS Table 3.3-10 items 21 and 22) were not entered. In addition, a PRC meeting on May 13, 1991, reviewed operability and concluded these instruments to be operable. The basis for this determination was that the RC-164A-LT and RC-164B-LT devices would still measure a vessel voiding condition as they were monitoring the reactor vessel stand pipe level. Also, other instrumentation was available to the operator (e.g., void monitor, core exit thermocouples and hot leg level monitor).

The inspector reviewed the LER, the PR, the SPs and WRs, and the FSAR. The inspector also discussed this item with licensee engineering, maintenance, operations and management personnel. The inspector questioned the licensee's operability determination since the licensee concluded the RC-164A-LT and RC-164B-LT instrument calibration was outside the design basis. The design basis as stated in the FSAR and in the TMI Action Plan Item II.F.2 was that the level from the reactor vessel hot leg to the vessel head was to be the measured range. Also, the LER appeared deficient in that it did not address the related TS issues nor provide the basis for instrument operability determination. This issue appears to be mitigated based on the licensee corrective actions including instrument re-calibration with four days of identifying the condition, (well within the seven day TS Action Statement) and the availability of redundant monitors for potential core voiding. In addition, this instrument provides an indication function only and operator implementation of EOPs would direct them to successfully assess a condition of inadequate core cooling. Also, the licensee was proactive in their corrective actions in that they reviewed all safety related and NRC Regulatory Guide 1.97 level instruments for similar type errors and none were found. All system engineers were required to review this event.

Based on the above actions, URI 50-302/91-04-01 and LER 91-04 are considered closed.

Violations or deviations were not identified.

7. Exit Interview

The inspection scope and findings were summarized on November 16, 1992, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

Item Number	Status	Description and Reference
VIO 50-302/92-27-01	Open	Failure to Follow Procedure Results in Valve Misalignment and Reactor Building Spray (paragraph 3.a)
VIO 50-302/92-27-02	Open	Failure to enter action statement 3.8.1.1 and perform surveillance requirement 4.8.1.1.1.a with the "B" emergency diesel generator inoperable as the result of non-TS surveillance testing (paragraph 4.a)
UNR 50-302/92-25-01	Closed	Followup of Inadvertent Actuation of Reactor Building Spray (paragraph 3.a)
LER 50-302/91-04	Closed	Personnel error caused reactor vessel level required for inadequate core cooling to be incorrectly calibrated (paragraph 5.a, 6.b)
LER 50-302/91-05	Closed	ECCS pump and flow paths were secured due to valve leak resulting in a condition outside the plant design basis (paragraph 5.b)
LER 50-302/91-10	Closed	Wiring problem causes transformer breakers to open actuating the emergency diesel generator (paragraph 5.c)
LER 50-302/91-12	Closed	Procedure deficiency leads to inadvertent emergency diesel generator actuation during

		engineered safeguards testing (paragraph 5.d)
LER 50-302/92-21	Closed	Lack of Required Lube Oil Leakage Collection Tank Reserve Capacity For Reactor Coolant Pumps Violates 10 CFR 50 Appendix R Design Criteria (paragraph 5.e)
NCV 50-302/90-32-02	Closed	Failure to provide adequate design control measures as required by 10 CFR 50, Appendix B, Criterion III (paragraph 6.a)
URI 50-302/91-04-01	Closed	Review and approval of inadequate procedure steps for calibration of reactor vessel level instruments (paragraph 6.b)

8. Acronyms and Abbreviations

AHF - Air Handling Fan
ANSI - American National Standards Institute
a.m. - ante meridiem
AP - Administrative Procedure
ASME - American Society of Mechanical Engineers
BS - Reactor Building Spray
B&W - Babcock & Wilcox
BWST - Borated Water Storage Tank
CFR - Code of Federal Regulations
CRDM - Control Rod Drive Motor
DHP - Decay Heat Pump
ECCS - Emergency Core Cooling System(s)
ENS - Emergency Notification System
EOP - Emergency Operating Procedure
EQ - Environmental Qualification
ES - Engineered Safeguards
FPC - Florida Power Corporation
FSAR - Final Safety Analysis Report
gpm - gallons per minute
HPI - High Pressure Injection System
LCO - Limiting Condition for Operation
LER - Licensee Event Report
MAR - Modification Approval Record
MUP - Makeup and Purification System
NCV - Non-cited Violation
NRC - Nuclear Regulatory Commission
OP - Operating Procedure
p.m. - post meridiem
PR - Problem Report
PRC - Plant Review Committee
RB - Reactor Building

RCP - Reactor Coolant Pump
RPS - Reactor Protection System
RSTS - Revised Standard Technical Specifications
SE - Safety Evaluation
SP - Surveillance Procedure
STI - Short Term Instruction
SW - Nuclear Services Closed Cycle Cooling System
TMI - Three Mile Island
TS - Technical Specification
TSI - Technical Specification Interpretation
URI - Unresolved Item
V - volt
VIO - Violation
WR - Work Request