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

Report No.: 50-302/92-16

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: May 31 - July 11, 1992

Inspector: Ross Butcher for 8/6/92
P. Holmes-Ray, Senior Resident Inspector Date Signed

Inspector: Ross Butcher for 8/6/92
R. Freudenberger, Resident Inspector Date Signed

Approved by: Marvin V Sinkule for 8/7/92
K. Landis, Chief Date Signed
Reactor Projects Section 2B
Division of Reactor Projects

Accompanying Personnel: A. R. Long, Project Engineer

SUMMARY

Scope:

This routine inspection was conducted by two resident inspectors in the areas of plant operations, security, radiological controls, and Licensee Event Reports. Numerous facility tours were conducted and facility operations observed. Backshift inspections were conducted on June 1, 2, 3 4, 14, 22, July 3, 7, 8, 10 and 11.

Results:

One Violation and one Unresolved Item (URI)* were identified:

Failure to establish an adequate procedure for surveillance calibration of the ES actuation channels, resulting in inadvertent decay heat removal isolation (Violation 50-302/92-16-01, paragraph 4.c).

Development and implementation of Corrective Action Plan for PR 92-0031 (URI 50-302/92-16-02, paragraph 5.f).

The following Licensee Event Reports (LERs) were reviewed:

LER 90-02: Fire Dampers May Not Close Under Ventilation Flow Conditions Due to Failure to Consider Flow Conditions in Original Design Criteria Per NRC IN 89-52 (Updated, paragraph 7.a).

LER 92-08: 10 CFR 50 Appendix R Design Requirement Not Entered Into Commitment System Results In Procedure Change That Causes Plant Operation Outside Design Basis (Closed, paragraph 7.b).

Additional inspection results were as follows:

Temporary Instruction 2515/113 "Reliable Decay Heat Removal During Outages" was closed (paragraph 3).

Operator response to the inadvertent isolation of the Decay Heat Removal System was timely and appropriate (paragraph 4.c).

Startup Simulator training was effective in refreshing the operators for the return to power operation (paragraph 4.d).

Immediate actions in response to Bulletin 92-01, concerning Thermo-Lag Fire Barrier System, were timely and appropriate (paragraph 4.g).

Maintenance activities reviewed on the Emergency Diesel Generators noted an improvement in fast start time of approximately two seconds (paragraph 5.a). The use of a mockup for bearing installation technique verification was also noted (paragraph 5.d).

A 10 CFR Part 21 issue associated with Calvert Company electrical bus ducts was reviewed and closed (paragraph 8).

A system walkdown of the Makeup/High Pressure Injection System identified poor housekeeping conditions (paragraph 6).

*Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*J. Alberdi, Manager, Nuclear Plant Operations
*K. Baker, Manager, Nuclear Configuration Management
*D. Bates, Supervisor, Quality Systems
*J. Baumgardner, Senior Quality Auditor
G. Boldt, Vice President Nuclear Production
*P. Breedlove, Nuclear Records Management Supervisor
*E. Froats, Manager, Nuclear Compliance
*H. Gelston, Acting Manager, Site Nuclear Engineering Services
*G. Halnon, Manager, Nuclear Plant System Engineering
B. Hickle, Director, Quality Programs
*D. Kurtz, Manger, Nuclear Quality Assurance
*W. Marshall, Nuclear Operations Superintendent
*J. Maseda, Manager, Nuclear Operations Engineering
*P. McKee, Director, Nuclear Plant Operations
*R. McLaughlin, Nuclear Regulatory Specialist
*S. Robinson, Nuclear Chemistry and Radiation Superintendent
*V. Roppel, Manager, Nuclear Plant Maintenance
*P. Tanguay, Director, Nuclear Operation Engineering Projects
*R. Widell, Director, Nuclear Operations Site Support
*G. Williams, Senior Nuclear Mechanical Engineer
*R. Yost, Supervisor, Quality Audits

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

NRC Personnel

*M. Thomas, Reactor Inspector, RII
*M. Hunt, Reactor Inspector, RII
*L. King, Reactor Engineer, RII
*C. Rapp, Reactor Engineer, RII
*M. Mizuno, Assignee, RII

*Attended exit interview

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

2. Plant Status and Activities

The facility was shutdown for the Cycle 8 refueling outage at the beginning of the report period. By the end of the report period the plant was in Mode 3, heating up in preparation for startup following the completion of refueling and maintenance activities.

During the week of June 1, a specialist inspection of the Outage Radiation Protection activities was conducted. The results of this inspection were documented in NRC Inspection Report 50-302/92-13.

Also during the week of June 1, a specialist inspection of Rad Waste and Transportation was conducted. The results of this inspection were documented in NRC Inspection Report 50-302/92-15.

During the weeks of June 1-5 and June 15-19, a specialist inspection of the Inservice Inspection area was conducted. The results of this inspection were documented in NRC Inspection Report 50-302/92-14.

On June 11 & 12, the Chief of Region II Reactor Projects Section 2B was on site for a routine site visit and to meet individually with FPC site management.

On June 17, a meeting to discuss several electrical issues identified and reported by the licensee was conducted in the NRC Region II Office. Representatives from the licensee's staff, NRC Office of Nuclear Reactor Regulation, and Region II were present.

During the period of July 8-14, a specialist inspection of Reactor Physics Testing was conducted. The results of this inspection were documented in NRC Inspection Report 50-302/92-17.

3. Reliable Decay Heat Removal During Outages (TI 2515/113)

A review was conducted of licensee activities during the refuel outage which had the potential for contributing to a loss of capability to remove decay heat from the reactor. Specifically, this inspection was performed in accordance with NRC Inspection Manual TI 2515/113, Reliable Decay Heat Removal During Outages. TI 2515/113 included review of licensee's planning and coordination of planned equipment outages, tests of systems and components, and plant conditions based on recent events (1991) described in NRC IN 91-22.

The licensee's administrative controls for reduced reactor coolant system inventory operations were contained in AI-504, Guidelines for Reduced Reactor Coolant System Inventory Operations. A detailed evaluation of the planned implementation of the AI-504 administrative controls was performed prior to the refuel outage, as documented in NRC Inspection Report 50-302/92-12, paragraph 3.a.

A review of the operation of the decay heat removal systems during the outage revealed no special test procedures or operations which had the potential for contributing significantly to a loss of capability to remove decay heat from the reactor. Forced circulation decay heat removal was the normal cooling method and there were no planned periods of natural circulation decay heat removal. AP-360, Loss of Decay Heat Removal, provided actions to be taken should forced circulation be lost during decay heat removal operations. Temperature monitoring using incore temperature detectors was recorded in SP-301, Shutdown Daily Surveillance Log, Enclosure 2. On June 27, during a calibration of the Engineered Safeguards Actuation Channels, an unplanned isolation of the Decay Heat Removal System resulted in a short term interruption in forced circulation decay heat removal (See paragraph 4.c, below).

A review of the supply and distribution of electric power to the decay heat removal and supporting systems revealed that AI-504, Enclosure 1, established the electrical power supply requirements for reduced RCS inventory operation. The primary electrical power source was backfeed from the 500 Kv yard through the Unit Output Transformers and the Unit Auxiliary Transformer. The backup power source was the 230 Kv yard through either the Offsite Power Transformer or the CR-3 Startup Transformer. The emergency power sources were the emergency diesel generators, including all required support systems for Technical Specification operability (including control systems and power). The administrative controls allowed only one diesel generator to be removed from service provided that the primary power source and at least one backup power source was available. AI-504 minimum requirements for availability of electric power sources increased as plant conditions become more vulnerable.

The inspectors observed and verified the implementation of the AI-504 administrative controls during the refuel outage. No discrepancies in the implementation of these controls were identified. The licensee's administrative controls provided increased decay heat removal reliability during the refuel outage. TI 2515/113 "Reliable Decay Heat Removal During Outages" is closed.

4. Plant Operations (71707, 93702, & 40500)

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. Reactor Cavity Seal Plate Leakage

On June 7, the fuel transfer canal was refilled in preparation for reload of the fuel into the reactor vessel. Upon refill, leakage of approximately fifteen gallons per minute from the fuel transfer canal to the reactor building sump was noted by the operators. The fuel transfer canal had been filled earlier in the outage to facilitate transfer of the fuel from the reactor to the spent fuel pools. No leakage was evident during the core off load. The licensee took several actions prior to opening the Fuel Transfer Tube isolation valves, which connect the spent fuel pools to the fuel transfer canal. These actions included an evaluation of the safety impact of the leakage on refueling operations, including contingency plans had the leakage increased to worse case conditions, identification of the location of the leak, and a temporary repair to reduce and stabilize the leak.

The leak was found to be the result of a portion of the outer "O" ring seal which had extruded. The licensee performed a temporary repair to prevent further extrusion of the "O" ring and reduce the leakage.

The licensee planned to install an improved design seal plate during the mid-cycle 9 outage, for use during the following refueling outage. The licensee's actions in response to the seal plate leakage were appropriate.

b. Temporary Waiver of Compliance during Core Alterations

On June 10, the licensee requested a Temporary Waiver of Compliance with Technical Specification 3.9.4, Refueling Operations, Containment Penetrations, which required each penetration providing direct access from the containment atmosphere to the outside atmosphere be either 1) closed by an isolation valve, blind flange, or manual valve, or 2) be capable of being closed by an operable automatic containment purge and exhaust isolation valve. The licensee's interpretation of the Technical Specification concluded that the specification prohibited the opening of the station air valves during refueling activities since the LCO applicability is during core alterations or movement of irradiated fuel in the containment. Based on this interpretation the licensee requested a Temporary Waiver of Compliance to allow administrative controls to be utilized for the station air penetration valves and temporarily installed air penetration isolation valves. The administrative controls consisted of a dedicated operator stationed at the valves, who would be assigned the responsibility of closing the valve when requested to do so by the control room.

The Temporary Waiver of Compliance was granted for up to seven days of cumulative core alteration and fuel movement time in the containment. The inspectors verified proper implementation of the administrative controls during core alterations and fuel movement. The licensee first utilized the Temporary Waiver of Compliance at 5:30 p.m. on June 10, 1992, and several times thereafter for a total of 20 1/2 hours cumulative time. The Temporary Waiver of Compliance was exited at 4:00 a.m. on June 17, 1992. This Temporary Waiver of Compliance is closed.

c. Decay Heat Removal Isolation

On June 27, with the plant in Mode 5, Instrument and Controls Technicians were calibrating Engineered Safeguards actuation channels in accordance with Surveillance Procedure SP-132, Engineered Safeguards Channel Calibration. The procedure is performed on a refueling interval to meet STS requirements and is to be performed in operating modes 4, 5 or 6. In modes 5 and 6 the decay heat removal system is in service providing core cooling.

During performance of Section 4.1.1, Calibration of Reactor Coolant Pressure Strings RC-3A-PT3, RC-3A-PT4, and RC-3B-PT3, the voltage buffer amplifier module output to the recall and computer systems from RC-3A-PT3 was within tolerance but not as accurate as desired. The procedure included provisions to remove the buffer amplifier module should it require recalibration. A note was included in the procedure that "removal of the buffer amplifier module will cause the respective ES channel to trip." The I&C technician recognized that removal of the buffer amplifier module may also cause the ACI system to actuate. The ACI system provides for automatic isolation of the Decay Heat Removal System from the Reactor Coolant System to prevent overpressurization of the relatively low pressure rated piping in the Decay Heat Removal System.

By a review of plant drawings, the RC-3A-PT3 string was verified to be powered from the Remote Shutdown Panel, versus the ES panel. It was recognized that removal of the buffer amplifier module in the ES panel would cause the ES power supplies to shutdown. This information was discussed with the on-shift operators. Although closure of the motor operated isolation valve, DHV-3, was not expected the operators reviewed the actions required by AP-360, Loss of Decay Heat Removal as a precaution. Although the operators and the I&C technicians involved questioned and investigated whether the removal of the buffer amplifier module might actuate the ACI system, the assessment of the I&C technicians performing the calibration was relied on to allow work to continue. Involvement of other plant personnel such as a System Engineer or I&C Supervisor and/or placing the ACI Channel in Bypass as a precaution was not performed.

Upon removal of the ES Channel 1 buffer amplifier module, the ES cabinet powered down and DHV-3 stroked closed. The I&C technician reinstalled the buffer amplifier module and reset the ACI bistable. The operators shut down the running DHR pump, allowed DHV-3 to completely close, then reopened it. The operators were aware of work ongoing in the vicinity of the DHR pump, therefore prior to restarting the pump the switchgear room and the decay heat pit were cleared of personnel. Decay heat removal was restored in approximately ten minutes. During that time, incore temperatures increased from 98.4 to 103.0 degrees F.

Operator response to the inadvertent isolation of decay heat removal complied with TS 3.4.1.4, was in accordance with AP-360, and was timely and appropriate.

The licensee reported the isolation of decay heat removal to the NRC in accordance with 10 CFR 50.72. The report was noted to be sufficiently thorough to provide for a complete understanding of the event, plant systems and operator response.

Inspector review of the cause of the isolation of decay heat removal indicated that Surveillance Procedure SP-132, Engineered Safeguards Channel Calibration, authorized removal of the buffer amplifier module without requiring the ACI channel to be placed in bypass or providing information that this action would result in automatic closure of the decay heat removal isolation valve. Therefore, SP-132 was inadequate. This is a violation of TS 6.8.1.c, which requires the establishment and implementation of written procedures for surveillance and test activities of safety-related equipment. The significance of the violation was evidenced by the inadvertent isolation of decay heat removal.

Violation (302/92-16-01): Failure to establish an adequate procedure for surveillance calibration of the ES actuation channels, resulting in inadvertent decay heat removal isolation.

d. Startup Simulator Training

On July 10, the inspector attended a simulator training session for Licensed Operators in preparation for restart of the unit. The training was included as one of the corrective actions following the evaluation of the December 8, 1991 transient. The scenario involved initial conditions with the reactor critical at the point of adding heat. In an approximately four hour session, the operators increased reactor power, phased on to the grid and increased power until the integrated control system could be placed in automatic. The training included realistic failures such as the failure of a main feed pump at low power, requiring the operators to quickly place the second main feed pump in service, the failure of a steam generator level instrument that had a recent design change to incorporate the Smart Automatic Signal Selector (SASS) controls, and a main generator failure to take sufficient load following phase on. The inspector considered the training effective in refreshing the operators for the return to power operation following the refueling outage.

e. Radiological Protection

Radiation protection control activities were observed to verify that these activities were in conformance with the facility policies and procedures, and in compliance with regulatory requirements. These observations included:

- Entry to and exit from contaminated areas, including step-off pad conditions and disposal of contaminated clothing;
- Area postings and controls;
- Work activity within radiation, high radiation, and contaminated areas;
- RCA exiting practices; and
- Proper wearing of personnel monitoring equipment, protective clothing, and respiratory equipment.

The inspector noted that proper postings and controls for the temporary storage of highly radioactive materials and components in the spent fuel pool and the reactor cavity were implemented.

The licensee's overall personnel exposure ALARA goal was apparently based on an administrative goal rather than an accumulation of the individual work activity ALARA goals. This resulted in a plant wide ALARA goal that was unrealistic and was exceeded by a significant margin prior to the end of the outage.

f. Security Control

In the course of the monthly activities, the inspector included a review of the licensee's physical security program. The performance of various shifts of the security force was observed in the conduct of daily activities to include: protected and vital areas access controls; searching of personnel, packages, and vehicles; badge issuance and retrieval; escorting of visitors; patrols; and compensatory posts. In addition, the inspector observed the operational status of protected area lighting, protected and vital areas barrier integrity, and the security organization interface with operations and maintenance. No performance discrepancies were identified by the inspectors.

g. Fire Protection

Fire protection activities, staffing, and equipment were observed to verify that fire brigade staffing was appropriate and that fire alarms, extinguishing equipment, actuating controls, fire fighting equipment, emergency equipment, and fire barriers were operable.

On June 24, 1992, NRC Bulletin 92-01 "Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage" was issued. The Bulletin described recent test results that indicated the Thermo-Lag 330 Fire Barrier System did not perform as rated on wide cable trays and small diameter conduits.

Immediately upon receiving the Bulletin, licensees were requested to 1) identify areas of the plant that have the material installed in similar configurations as those that failed the testing, and protect equipment that provide for safe shutdown capability, and 2) implement appropriate compensatory measures such as fire watches, in accordance with plant procedures, that would be required by the Technical Specifications or the operating license for an inoperable fire barrier.

The inspector reviewed the licensee's immediate actions in response to the Bulletin. The licensee maintained a roving hourly fire watch that covered most areas of the plant. The roving watch was verified to cover all affected areas. Technical Specification 3.7.12 compensatory measures for inoperable barriers required a continuous fire watch or a hourly watch if there are operable detectors in the area. A review of the installations in the plant indicated that there were only two areas in the Auxiliary Building that contained the configurations that failed testing and did not have operable detection systems. Continuous fire watches were implemented in these areas until detection systems could be installed and made operable.

The licensee's immediate actions in response to the information provided in Bulletin 92-01 were timely and appropriate.

5. Maintenance and Surveillance Activities (62703, 61726, & 61701)

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 0296554, String Checks and as-left Data for several reactor coolant pressure transmitters;

- WR 0297200, Motor Operated Emergency Feedwater Valve, EFV-33, internals inspection and rebuild;

- WR 0244098, Air Operated Core Flood Tank Nitrogen Supply Valve, CFV-28, Preventive Maintenance - Rebuild of Actuator;

- WR 0298758, Motor Operated Core Flood Valve, CFV-6, erratic operation; and

- WR 0291982, WR 0298253, WR 0298138, and WR 0290555, Reactor Building Cooling Fan, AHF-1A, Troubleshooting high vibration, bearing lubrication change, motor inspection/lubrication, and alignment.

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.

Portions of the following calibration or test procedures were observed and/or data reviewed:

- SP-630, MUP/HPI Check Valves Full Flow Test;

- SP-605, PRR #31, Emergency Diesel Generator Engine Inspection/Maintenance;

- SP-414, High Pressure Injection Flow Verification Test;

- SP-402, Core Flooding System Isolation Valves Alarms Actuation;

- SP-405, Core Flooding System Check Valve Operability and Demonstration;

- SP-440, Unit Startup Surveillance Plan;

- SP-603, DH/CF Check Valve Leak Testing; and

- PT-315, Remote Shutdown Relay Operability.

Inspector comments on the above maintenance and surveillance items are as follows:

a. EDG Inspection/Maintenance

TS 4.8.1.1.2.d requires each EDG to be inspected in accordance with the manufacturer's recommendations at least once per 18 months. This surveillance requirement is implemented by SP-605, Emergency Diesel

Generator Engine Inspection/Maintenance. The 18-month inspection requirements of SP-605 were last performed in full on the B diesel generator during the midcycle outage (8M) in October 1991. Although SP-605 was not performed in full on the A diesel generator during the 8M outage, in accordance with REA 91-1334 most of the 18-month inspection activities were performed, with the exception of the engine block and intercooler heat exchanger hydros. In addition, any problems encountered during maintenance on the B EDG, such as the discovery of a failed thrust bearing (reference NRC Inspection Report 50-302/91-23 and licensee Problem Report SYPR-91-0027), were also addressed for the A EDG.

During the current refueling outage (8R), the 18-month inspection per SP-605 was performed in full on the A EDG and a more limited scope of maintenance activities was performed on the B EDG, placing the two diesels on a staggered testing schedule.

The EDG surveillance inspections for outage 8R were conducted in accordance with an "immediate issue" of SP-605 (PRR-31), effective May 14, 1992. This procedure revision was a major rewrite which incorporated new inspection requirements and actions based on vendor recommendations, licensee experience during previous EDG maintenance, and other industry experience. As a result of the thrust bearing failures in October 1991, new guidance was incorporated for setting engine crank strain and for measuring and setting the generator air gap. To eliminate the risk of damage to the ex iter-regulator and generator field, the unloaded test operation and overspeed trip testing were revised to ensure the generator field will not flash. Revision PRR-31 also added requirements to document as-found conditions for use in performance trending, incorporated an engine run-in procedure, and included specifications for controlling torque values during disassembly/reassembly of engine flanged/mechanical joints. While reviewing SP-605, the inspector randomly selected a sample of vendor requirements transmitted via Service Information Letters and other correspondence, and confirmed that these requirements had been incorporated into the action steps or caution statements of the procedure.

Following completion of the 18-month inspection and maintenance activities per SP-605, EDG A successfully completed the required 30 minute unloaded test run and overspeed trip test on May 28, 1992, and the four-hour loaded test run on May 29, 1992. Effectiveness of the maintenance activities was evidenced by a reduction in the start time of the A EDG from an average of 7.8 seconds before the outage, to 5 seconds.

During the review of the completed SP-605 procedure package for the A EDG, the inspector noted that on the blower checks performed per Enclosure 3, the measured average lower rotor to inner bearing plate clearance of .0353 inches exceeded the specified acceptance criterion of .034 inches maximum clearance. However, the procedure package contained no documentation that this nonconformance had been addressed through an engineering evaluation and resolved. The inspector discussed the lack of an engineering evaluation with the system engineer, and REA 92-1131 was promptly initiated. REA 92-1131 contained a data comparison demonstrating that no measurable component degradation had occurred between the 8M and 8R outages, and referenced a previous REA (REA 91-148, generated during the 8M outage EDG work) which established that a clearance of .036 inches was acceptable. At the time of the nonconforming outage 8R measurement, the system engineer was aware that REA 91-148 had previously established the acceptability of the measured clearance value. The failure to issue another REA for the current EDG surveillance was an administrative oversight without technical significance. All other nonconforming conditions identified during the surveillance inspection were adequately addressed.

The lower thrust bearings on both diesels were inspected during outage 8R and found to be within established tolerances. Also among the maintenance activities performed on the B EDG were governor maintenance, fuel control leakage checks, and regasketing to repair jacket coolant leaks on the water bypass fittings.

Based on discussions with the system engineer, several significant elastomer failures have occurred on the B EDG since the beginning of the 8M outage in October 1991, as described below. Two of these resulted in failures (actual or administratively declared) of the EDG to start on demand.

- Near the beginning of midcycle outage 8M, the B EDG failed to start on demand due to an elastomer failure on the stop flow check valve on the fuel header, which allowed fuel to flow back out of the line to the day tank.

- On March 27, 1992, with inverter VBIT-1C inoperable (A train), an undervoltage condition on the 4160 V safeguards busses and subsequent reactor trip generated a diesel generator autostart. Both diesels started and re-energized their busses. However, start of the B EDG worsened a pre-existing coolant leak caused by degraded elastomers in the mechanical seal of the engine-driven coolant pump. Although it was possible that with dedicated operator assistance the EDG could have been considered operable, the decision was made to declare the B EDG inoperable, invoke Technical Specification 3.0.5, and initiate cooldown. Although the B diesel successfully started and ran despite the coolant leak, this event was considered by the licensee to be a failure of the EDG to start on demand because it had been declared inoperable.

- During the current refueling outage, it was discovered that failed elastomers on a lower main bearing oil booster had allowed lube oil into the start-air header, potentially compromising the ability of the engine to fast-start. The O-rings and gaskets were replaced, and the start-air check valves disassembled and cleaned.

In response to these and other identified elastomer problems, the licensee has made numerous improvements to the EDG maintenance program. The corrective action for Problem Report PR-SS-92-05, issued to address the jacket coolant pump seal leakage, will address other similar systems and components in which elastomers are used, and develop a PM program item to change seal parts before a reasonable end of lifetime point is reached.

Overall, the licensee continues to emphasize the quality of the EDG 18 month interval inspections and maintenance, with particular importance placed on items affecting fast start. Lessons learned from unit and industry maintenance experience have been factored into procedural upgrades, resulting in inspections and actions beyond those specified by the vendor requirements.

b. Fuel Cladding Damage

Ultrasonic testing of unloaded fuel assemblies identified 20 leaking fuel pins. None of the leaking pins were in assemblies which were to be returned to the core for the next cycle. Three of the leaking fuel pins were pulled and examined visually. Two of these were from peripheral assemblies with inconel grids, and showed no visible damage. However, visual examination of the third pin, from an assembly with zircalloy grids located three positions in from the periphery, showed noticeable damage in the upper portion of the rod. The inspector viewed portions of the video tapes of the visual inspections of the damaged fuel pin, which was separated into two sections with a two-inch gap between. The licensee initiated an investigation to determine whether the damaged rod became separated before or during the pulling process and whether any fuel pellets were missing, and to evaluate possible implications for future operation. Based on the size of the gap, up to four fuel pellets could have come out of the pin. Also during the visual inspections, the licensee observed darkened, evenly-spaced circular rings on certain sections of the pins. These did not appear to be related to the fuel failure, because they were not observed in the upper portion of the rod where the failure occurred, and the cladding did not visually appear to be breached in these areas. The investigation of fuel pin damage will be completed after restart. Health physics was alerted to monitor for potentially higher than normal radiation levels in systems which contain reactor coolant.

c. Repair of Damaged Fuel Assembly Alignment Guide

The alignment guide, or "ear", of fuel assembly NJ0486 was damaged during installation of a hold down spring retainer plug after spring replacement. The damaged alignment ear was on the serial number side, and had an inward deflection of approximately 3/4 inch. The damage was repaired by the licensee in accordance with B&W procedure FO-102, Revisions 0 and 1, "Repair of MK-B4 Upper End Fitting at Crystal River." An engineering justification for movement of the repaired assembly into the core was documented in an interoffice memorandum to the PRC, dated June 8, 1992. The inspector reviewed procedure FO-102, the licensee's engineering justification for fuel assembly movement, and additional supporting documentation provided by B&W as Engineering Information Record (EIR) 51-1213844-00, dated June 4, 1992.

To develop acceptance criteria for the number of bending cycles (two bends per cycle) allowed during the repair, B&W bent and straightened the alignment ears on a sample of upper and lower end fittings, then examined them for evidence of cracking or structural weakening. The results of the study showed that an angle of over

15 degrees could withstand in excess of 40 cycles without cracking, and an angle of 25 degrees would have to be cycled more than four times before cracks would develop. B&W therefore conservatively recommended a limit of four bend cycles of approximately 6 1/2 degrees during the repair.

The methodology for the alignment ear repair was to push the ear in, then pull it out from behind. During repair of the damaged ear, some minor bending of the adjacent and opposite ears occurred. This damage was also straightened in accordance with procedures. The most cycles any ear received during the straightening process and spring replacement was three cycles, and the maximum bend was 40 mils versus the 375 mils allowed by procedure. The repaired ears were visually examined before and after spring replacement, and no evidence of cracking was observed. The engineering documentation for the repair provided adequate assurance that movement of the repaired fuel assembly into the core satisfied the requirements of 10 CFR 50.59. The alignment ear repair did not significantly increase the probability or consequence of a fuel handling accident or malfunction previously evaluated in the FSAR, create the possibility of a previously unanalyzed accident, or reduce the margin of safety as defined in the basis for the Technical Specification pertaining to fuel handling. The fuel assembly was moved into the core on June 12, 1992.

During core load verification, it was identified that the assembly was installed rotated 90 degrees out of position. The assembly was originally rotated as part of the repair effort. Following completion of the repair, the move sheet that removed the assembly from the work location did not include direction to restore the assembly to its original orientation. This was an oversight on the part of the reactor engineer that developed the move sheets.

The licensee initially planned to remove the assembly and replace it in the proper configuration. Based on a review of the assembly's position within the core (periphery) and the risk of damaging the assembly during movements with all the surrounding assemblies in place, the licensee chose to leave the assembly installed rotated 90 degrees out of position. The inspector had no further questions.

d. Containment Cooling Fan Maintenance

The three Containment Cooling Fans, designated AHF-1A, AHF-1B, and AHF-1C, are Westinghouse fans equipped with a two speed motor which operates at either 150 HP (1800 RPM) or 75 HP (900 RPM). Normally, two of these fans are operated simultaneously in high speed to provide normal reactor building cooling and air distribution. During periods of high outdoor temperature all three fans have historically been operated to maintain reactor building average temperature at or below the Technical Specification required 130 degrees F. Following a design basis LOCA, or main steam or feedwater line break, two of the fans operate in the Reactor Building Emergency Cooling Mode. In this mode the two selected fans operate in slow speed. Technical Specifications require at least two of the containment cooling fans operable in Operating Modes 1, 2, and 3.

In August of 1991, the A Containment Cooling Fan failed while operating in high speed. Following repairs to AHF-1A during the mid-cycle maintenance outage in October 1991, the unit exhibited higher than normal, but acceptable, vibration. The licensee performed a failure analysis of the August 1991 failure. The Failure Analysis (No. 92-AHF-1A-01) was unable to identify the specific cause of the failure. However the analysis identified likely contributing factors and many previously unknown or unrecognized facts related to the fans' maintenance and vendor recommendations. Recommended corrective actions as a result of the failure analysis included: (1) complete physical inspections of the fan units, (2) cleaning and balancing of the supply registers and dampers, (3) enhanced training, (4) amendments to the PM Program, (5) revisions to the fan maintenance procedure, vendor manual, and FIMIS, (6) establishment of a basis for tracking and trending performance, and (7) an analysis to determine the acceptability of fan operating practices.

During the refuel outage, troubleshooting and preventive maintenance activities were performed. A revised MP-138, Maintenance of Reactor Containment Cooling Fans (AHF-1A, AHF-1B and AHF-1C), was used to perform this maintenance. During post maintenance functional testing, significant vibration occurred shortly after starting the fan and the fan shaft shifted axially. Investigation of these difficulties during post maintenance testing revealed that the measurement of initial bearing clearances for uninstalled fan bearings varied significantly between individuals. Accurate measurement of bearing clearances is critical for proper installation of the fan shaft bearings to be able to accept thrust loads. A standardized method of measuring bearing clearances and clearance reduction during bearing installation was developed and performed on a mockup shaft in the cold machine shop. The shaft was installed in a press and test loaded to verify that the installation technique resulted in bearing thrust load performance within the manufacturer's recommendations. The details associated with the installation technique and mockup testing were documented in REA No. 92-1177.

The inspector noted that the use of the mockup was a positive initiative to demonstrate the performance of the revised bearing installation technique.

e. Reactor Building Closeout Inspections

A new procedure for a closeout inspection of the Reactor Building was issued on July 1. The procedure, AI-1305, Administrative Inspection of Reactor Containment, designated responsible managers who were assigned the task of inspecting specific areas of the containment prior to plant heatup. The inspections were intended to ensure the containment was restored to its as designed condition after a major outage.

Inspector walkdowns of various areas of the reactor building, review of the RB Walkdown Master Deficiency List and disposition of the deficiencies indicated that the procedure was effective at identifying deficiencies and improving the material condition of equipment in the reactor building.

f. High Pressure Injection Valve Limit Switch Settings

On May 8, 1992, Nuclear Operations Engineering raised a concern with the settings of the MOV limit switches which control injection flow following an ES actuation. The concern was that there were two open limit switches associated with each injection valve and the current method for setting the limits did not clearly define which limit was the ES throttled position and which was the full open limit. The ES throttled position was used to balance flows among the injection lines. The full open position was used to prevent back seating of the valves. Since the HPI valves are plug type throttle valves and the openings in the lower portion of valve cage are smaller than the reactor building recirculation sump screens, the full open limit was provided to allow the operators to open the valves fully, uncovering larger openings in the cage, during the recirculation phase of an accident.

System Engineering review concurred that the present method was not adequate to ensure correct as-left limit switch settings. Based on this conclusion, Problem Report 92-0031 was initiated and work requests were generated to measure as-found limit switch settings prior to the performance of preventive maintenance activities scheduled on the MOVs during the refueling outage.

The as-found data was collected on May 26, 1992. It indicated that both the ES throttled position and the full open limit switches were set at the same, industry standard full open position of 90% to 95% open. It appeared that both the open limit switches on each of the four valves were set to the full open position in 1987, when the MOV torque switch bypass modification (MAR 87-03-13-02) was performed. In that configuration, past performance data existed only to demonstrate that MUP-1B lineup was capable of meeting the Technical Specification 4.5.2.g. requirements. Data taken during the performance of SP-414, High Pressure Injection Flow Verification Test, in 1986, showed the Technical Specification requirements of 500 gpm total flow and a minimum of 350 gpm for any combination of three out of four injection lines, at a minimum RCS pressure of 600 psig was achieved with the injection valves in the full open position and MUP-1B running. No conclusive data existed to show that MUP-1A or MUP-1C could meet the minimum flow of 350 gpm for any combination of three out of four injection lines with the injection valves full open.

A revised version of SP-414 was performed in Mode 3 during startup from the refuel 8 outage. This test demonstrated that with any makeup pump in operation and the High Pressure Injection Valves full open flows balanced within the acceptance criteria of Technical Specification 4.5.2.g were achieved. Therefore, the operation of the High Pressure Injection System was within its design basis with both sets of open limit switches set to the full open position.

The inspector noted that a questioning attitude on the part of plant operators identified this issue and System Engineering pursuit of addressing the issue until the safety impact was understood was timely and aggressive. The Corrective Action Plan associated with Problem Report 92-0031, which addresses the cause of the improper MOV limit switch settings, was under development at the end of the report period. This issue is unresolved pending development and implementation of the Corrective Action Plan.

Unresolved Item (302/92-16-02): Development and implementation of Corrective Action Plan for Problem Report 92-0031.

6. Safety Systems Walkdown (71710)

The inspector conducted a walkdown of portions of the Makeup/High Pressure Safety Injection Systems to verify that the lineup was in accordance with license requirements for system operability and that the system drawing and procedure correctly reflect "as-built" plant conditions. This walkdown was conducted after the Makeup System had been placed in service during heat up of the reactor coolant system.

As part of the system walkdown the inspectors attempted to review the status of outstanding work orders associated with the system following its return to service. A listing of open work orders was obtained from the licensee's MACS computer system. Based on the list there was a significant amount of outstanding work. Closer review identified that many of the work requests shown as open on the MACS system were actually closed and functionally tested. No outstanding work that significantly compromised the operability of the system was identified by the inspector.

Observations of the material condition and housekeeping in the Auxiliary Building during the system walkdown indicated that conditions had degraded significantly during the outage. Excessive amounts of scaffolding was in place in the vicinity of safety related equipment and tools and other work materials were left within contaminated areas. Following the Reactor Building Closeout, resources were focused on the Auxiliary and Intermediate Buildings and conditions improved.

7. Review of Licensee Event Reports (92700)

Licensee Event Reports 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. (Open) LER 90-02: Fire Dampers May Not Close Under Ventilation Flow Conditions Due to Failure to Consider

Flow Conditions in Original Design Criteria Per NRC IN 89-52.

There have been three revisions of this LER issued; the original on March 19, 1990; Revision 1 on January 9, 1991; and Revision 2 on May 17, 1991.

In June of 1985, the need to verify the ability of fire dampers, especially multi-section dampers, to close under ventilation flow conditions was identified.

Several control complex fire dampers listed in this LER, were scheduled for modification during the refueling outage that started on April 30, 1992, but were dropped for outage scope reduction. The dampers are now scheduled for Mid-Cycle 9 outage in the spring of 1993. This LER will remain open pending completion of the fire dampers work.

b. (Closed) LER 92-08: 10 CFR 50 Appendix R Design Requirement Not Entered Into Commitment System Results In Procedure Change That Causes Plant Operation Outside Design Basis.

Prompted by Appendix R related issues previously reported in LERs 89-38, 89-39 and 88-12 and numerous plant modifications and procedure revisions that have been performed since the Appendix R safe shutdown analysis was initially implemented in 1985, the licensee utilized an independent contractor (United Energy Services Corporation) to perform an evaluation of CR-3's Appendix R safe shutdown analysis to assess the documentation and procedural controls for maintaining conformance to the regulations.

The evaluation was conducted in March and April of 1992 with preliminary results presented in early May. The licensee's initial review of the results identified the following issue which was evaluated to require immediate corrective action and reporting. The licensee's initial review did not identify other issues with immediate impact on plant operations.

This issue was reported to the NRC on May 7, 1992, (EN 23419).

The contractor determined from the evaluation that the procedure for alignment of cooling water to the HPI pumps did not conform with the design requirements of Appendix R, Sections III.G and III.J. The line-up of cooling water required by procedure did not assure at least one safe shutdown train remained functional for all postulated fires.

The Appendix R Fire Study issued in 1985 evaluated the necessary safe shutdown equipment trains to assure protection is provided for at least one of the HPI pumps in order for the unit to achieve Mode 3 (Hot Shutdown) following all Appendix R postulated fires. Protection means the necessary power supplies, components, control circuits, and support systems remain functional and undamaged from fire. The approach taken for CR-3, as stated in the Appendix R Fire Study, was to keep the A HPI pump available for fires postulated in the fire area containing the HPI pumps, and the C HPI pump available for fires postulated in the fire area one elevation above which contains power and control circuitry for the HPI pumps. The B HPI pump is available to replace either the A or C HPI pump should one become inoperable and is protected from fire damage in accordance with Appendix R as necessary. The component cooling water supplies, which are considered to be a necessary support function for all the HPI pumps, are located in the same fire area as the HPI pumps.

Component cooling for the HPI pumps is comprised of two systems: the Nuclear Services Closed Cycle Cooling (SW) and the Decay Heat Closed Cycle Cooling (DHCCC) systems. Component cooling water can be supplied to the HPI pumps as follows.

A HPI pump - SW or DHCCC
B HPI pump - SW only
C HPI pump - SW or DHCCC

The SW system has three pumps and four heat exchangers and the DHCCC has two pumps and two heat exchangers. For the purpose of Appendix R, one train of SW is protected from fire damage while neither of the DHCCC trains are protected. Since all trains of component cooling are located in the same fire area as the HPI pumps, the Appendix R Fire Study concluded that the A HPI pump must have component cooling supplied from the protected train of SW at all times. The C HPI pump can receive cooling water from either the DHCCC or SW since it is credited for operation only in the event of a fire on the elevation above, and a fire in this area has no adverse effect on either SW or DHCCC.

The operational procedures which establish the required cooling water alignment to the HPI pumps were reviewed in 1985 to assure the proper line-up for SW to the A

HPI pump as addressed. The procedures at that time provided the proper alignment. However, in 1986 the procedures were changed to align the a HPI pump to DHCCC whenever the C HPI pump was taken out of service. This change was the result of actions taken in response to concerns identified in a 1986 NRC violation involving conformance to component cooling water alignments contained in the FSAR. It was not recognized at that time that these procedure revisions and the contents of the FSAR were in conflict with the requirements of Appendix R.

The inspector verified that Operating Procedures OP-402, Makeup and Purification System; OP-408, Nuclear Services Cooling System; OP-404, Decay Heat Removal System; and Surveillance Procedure SP-3[illegible]1, Locked/Sealed Valve Check List Position Verification of Locked/Sealed Valves, were revised to incorporate proper cooling [illegible] alignment to the makeup pumps. The requ[illegible] the associated implementing procedu[illegible] included in the licensee's commitment tracki[illegible] assure future revisions of the procedu[illegible] include the proper alignment.

LER 92-08 is closed.

8. Review of Part 21 Report - Calvert Bus Duct

On June 30, 1992, the Calvert Company, manufacturers of non-segregated phase electrical bus duct used in Safety and Non-Safety Related applications at Crystal River Unit 3, made a 10 CFR, Part 21 report to the NRC Operations Center (reference EN 23761). This report was made as a result of non-compliance with the original purchase specifications (Florida Power Corporation Purchase Order PR3-1686Q, dated 7/27/70) for the bus ducts from the offsite power transformers (Startup and Unit Auxiliary Transformers) to the 6900 and 4160 volt busses. The purchase order specified that the busses be able to withstand the forces generated by an 80,000 ampere asymmetrical fault. By letter dated April 27, 1992, the Calvert Company informed Florida Power Corporation of the results of a recent analysis which indicated that the momentary short circuit capability of the bus duct configurations was less than that specified by the Florida Power Corporation purchase specification.

Upon receipt of the April 27 letter from the Calvert Company, Florida Power had initiated a Problem Report which included corrective action to perform an analysis to determine the actual momentary currents per bus duct. Due to margin in the initial design, the calculations concluded that the rating of all of the sections of the non-segregated phase bus ducts provided by the Calvert Company was adequate for the available fault current. Therefore, the licensee's Problem Report was closed with no modifications to the bus duct required. The licensee planned to submit a 10 CFR, Part 21 Report, had the manufacturer failed to do so.

This 10 CFR Part 21 issue is closed.

9. Exit Interview

The inspection scope and findings were summarized on July 13, 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	Description and Reference
50-302/92-16-01	Violation: Failure to establish an adequate procedure for surveillance calibration of the ES actuation channels (paragraph 4.c)
50-302/92-16-02	Unresolved Item: Development and implementation of Corrective Action Plan for PR 92-0031 (paragraph 5.f)

10. Acronyms and Abbreviations

AI - Administrative Procedure
ACI - Automatic Closure and Interlock
ALARA - As Low as Reasonably Achievable
AP - Annunciator Response Procedure
B&W - Babcock & Wilcox
CFR - Code of Federal Regulations
CR3 - Crystal River Unit 3
DHCCC - Decay Heat Closed Cycle Cooling System
DHR - Decay Heat Removal
DHV - Decay Heat Valve
EDG - Emergency Diesel Generators
EIR - Engineering Information Record
EN - Enforcement Notification
ES - Engineered Safeguards
F - Fahrenheit
FIMIS - Fully Integrated Materials Information System
FO - Fuel Operations Procedure
FPC - Florida Power Corporation
FSAR - Final Safety Analysis Report
gpm - gallons per minute
HP - Horse Power
HPI - High Pressure Injection

I&C - Instrumentation and Control
IN - Information Notice
Kv - Kilovolt
LCO - Limiting Condition for Operation
LER - Licensee Event Report
LOCA - Loss of Coolant Accident
MACS - Maintenance Activity Control System
MAR - Modification Approval Record
MOV - Motor Operated Valve
MP - Maintenance Procedure
MUP - Make-up Pump
NRC - Nuclear Regulatory Commission
OP - Operating Procedure
PM - Preventive Maintenance
PR - Problem Report
PRC - Plant Review Committee
PRR - Procedure Review Report
psig - pounds per square inch gauge
PT - Performance Testing Procedure
RB - Reactor Building
RCA - Radiation Control Area
RCS - Reactor Coolant System
REA - Request for Engineering Assistance
RPM - revolutions per minute
SASS - Smart Automatic Signal Selector
SP - Surveillance Procedure
STS - Standard Technical Specification
SW - Nuclear Services Closed Cycle Cooling System
TI - Temporary Instruction
TS - Technical Specification
V - Volt
WR - Work Request