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

Report No.: 50-302/92-06

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

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

Facility Name: Crystal River 3

Inspection Conducted: February 1 - 28, 1992

Inspectors: P. Holmes-Ray 3-19-92
P. Holmes-Ray, Senior Resident Inspector Date Signed

For 3-19-92
R. Freudenberger, Resident Inspector Date Signed

3/23/92
R. Schin, Project Engineer Date Signed

Appproved by: 3-20-92
K. Landis, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope:

This routine resident inspection was conducted on site in the areas of plant operations, security, radiological controls, Licensee Event Reports, facility modifications, and licensee action on previous inspection items. Numerous facility tours were conducted and facility operations observed. Some of these tours and observations were conducted on backshifts.

Results:

No violations or deviations were identified during this inspection. Plant mechanics demonstrated a questioning attitude during a system outage on the Emergency Feedwater System, and this is considered a strength. However, complications in the efficient performance of the system outage prompted senior plant management to initiate a quality assurance surveillance and this is considered a positive step for improving job efficiency and thereby reducing safety system outage time. Corrective actions as a result of a main steam line

water hammer were considered weak. Control of troubleshooting activities was inconsistent. Two 10 CFR Part 21 reports and two LERs were closed:

50-302/P2191-03: Part 21 Report From Rockbestos Re KS-500 Silicone rubber Activation Energy Values For Firewall GR Silicone Rubber Insulated Cable and Firezone R Special Purpose Cable.

50-302/P2190-04: Part 21 Report From Rosemount re: Resistance Bridge Can Exhibit Premature Long Term Degradation Under Certain Combinations of Humidity, Power, and Duration.

LER 302/90-10: Voltage Dips Caused by Sluggish Voltage Regulator Response Exceed Regulatory Guide 1.9 Limits.

LER 302/90-19: An Incorrect Motor Installed On a Valve Operator Results in a Condition Outside the Design Basis.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *J. Alberdi, Manager, Nuclear Plant Operations
   G. Boldt, Vice President Nuclear Production
   *J. Buckner, Nuclear Regulatory Specialist
   *E. Froats, Manager, Nuclear Compliance
   *H. Gelston, Supervisor, Site Nuclear Engineering Services
   *G. Halnon, Manager, Nuclear Plant System Engineering
   *B. Hickle, Director, Quality Programs
   *D. Kurtz, Manager, Site Nuclear Quality Assurance
   *G. Longhouser, Superintendent, Nuclear Security
   *W. Marshall, Nuclear Operations Superintendent
   *P. McKee, Director, Nuclear Plant Operations
   *S. Robinson, Nuclear Chemistry & Radiation Superintendent
   *V. Roppel, Manager, Nuclear Plant Maintenance
   *W. Rossfeld, Manager, Site Nuclear Services
   *R. Widell, Director, Nuclear Operations Site Support
   *K. Wilson, Manager, Nuclear Licensing
   *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

   *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.

   During the week of February 10 through 14, a specialist inspector from the NRC Region II office performed an inspection of the licensee's Inservice Inspection Program. The results of the inspection were documented in NRC Inspection Report 50-302/92-05.

3. 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 activity observations were conducted during backshifts. Radiation protection activities were observed to verify conformance with facility procedures and regulatory requirements. In the course of the monthly observations, the inspector included a review of the implementation of licensee's physical security program. The performance of various shifts of the security force was observed in the conduct of daily activities. In addition, the inspector observed the operational status of selected security related equipment. 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. The following items were considered noteworthy:

a. NGRC Reorganization

In order to provide enhanced safety perspective, the NGRC was restructured into four major subcommittees. Previously the subcommittees were:

- Significant Events and LER
- Safety Evaluation
- Audit Program
- Technical Specification
- Environmental Monitoring
- Violation
- Corporate Review

The new subcommittees are:

- Engineering and Technical Support
- Operations and Maintenance
- Radiation, Chemistry and Environmental
- Quality and Regulatory Verification

Where before one subcommittee reviewed all LERs, now the appropriate subcommittee will review those LERs in that specific area. The licensee expects to apply more expertise to each issue by the new structure of NGRC and therefore provide a greater safety influence.

This restructure of the NGRC subcommittees was a direct result of information gathered by a self-assessment questionnaire discussed in NRC Inspection Report 50-302/91-20, detail 3.d and appears to be a positive initiative to improve the quality of safety reviews by applying the expertise of the subcommittee members more specifically. The resident inspectors will remain cognizant of these changes and their impact on the effectiveness of NGRC.

b. Control Rod 7-1 Position Indication

On February 19, control room personnel initiated a formal operability evaluation of the position indication associated with control rod 7-1. Control rod 7-1 is in group 7, which is one of the three control rod groups automatically sequenced to control RCS Tave when the ICS is operating in automatic. The operability evaluation was performed and documented in accordance with NOD-14 "Determining Operability."

During the performance of SP-300 "Operating Daily Surveillance Log" the operators noted that while control rod 7-1 and the rest of the group was positioned at greater than 99% withdrawn, the Absolute and Relative Position Indications were in agreement and met the requirements of TS 3.1.3.3. As the group was inserted, the API for control rod 7-1 remained high, while the RPI tracked down with the rest of the group. Therefore, with the RCS boron concentration adjusted to maintain control rod group 7 withdrawn such that control rod group 7 was greater than 98% withdrawn, the API and RPI were in agreement and meet the TS requirements. If control rod group 7 was inserted, the limiting condition for operation of TS 3.1.3.3 would apply. The operability concern raised questioned the operability of control rod 7-1 position indication regardless of rod position.

The operability evaluation was performed in a timely fashion and completed the following day. Based primarily on a two-day trend of group 7 control rod position indications, control rod 7-1 API appeared to have significant noise, independent of rod position. Therefore, the position indication for control rod 7-1 was declared inoperable and the actions of TS 3.1.3.3 were implemented. Specifically, group seven was withdrawn to its 100% zone reference position indication to verify actual control rod positions. Since group seven was the normal control group, the ICS was placed in "track" by placing the control rod station in "hand" (manual). This effectively parked the group at 100% withdrawn and resulted in RCS Tave control being shifted to feedwater control and power level control by RCS boron concentration. Power level was stabilized at approximately 99% to prevent inadvertent prolonged operation at greater than 100% due to normal variations in plant efficiency.

The inspector noted that a procedure such as NOD-14 "Determining Operability," which allowed operability evaluations to be researched by operations support staff, could be misused to inappropriately delay declaring a system or component inoperable. The inspector reviewed NOD-14 and the circumstances under which it was implemented in this case. NOD-14 stated that it was to be used when the operability status may not be immediately evident and an evaluation was required to assist the shift supervisor in his determination. NOD-14 also stated that the timeliness of operability determinations shall be commensurate with the safety significance and in all cases prompt.

The inspectors discussed the potential for misuse of NOD-14 with plant management. Licensee management stated that it was infrequently used, only in particularly unclear situations. The inspector concluded that in this case the conditions for the operability determination was sufficiently unclear, and the NOD-14 operability determination was timely, therefore this case was not considered to cause an unreasonable delay in the operability determination.

c. New Fuel Receipt

On February 24, the licensee received the first of several new fuel shipments for use during the refueling outage currently scheduled for May and June 1992. The inspectors observed portions of the preparations for receipt and unloading of the fuel, including testing of fuel handling equipment. Unloading and Inspection of the new fuel was also observed. These evolutions were primarily controlled by Refueling Procedure FP-302 "New Fuel Assembly Unloading, Inspection, Storage, and Container Reclosing." The evolutions observed were performed in a controlled manner.

d. Main Steam Line Water Hammer

On November 24, 1991, a water hammer occurred in the main steam lines associated with the B OTSG. This was evidenced by a loud noise heard in the Control Room concurrent with several main steam annunciator alarms and a decrease of approximately 40 psig in the B OTSG. This water hammer occurred during a plant heatup in accordance with operations procedure OP-202 "Plant Heatup" following approximately twenty-four hours of steady state operation with the RCS at 300 F. The RCS had previously been heated to normal operating temperature (532 F) and then cooled down to 300 F to support repairs to a malfunctioning intermediate range nuclear instrument (NI-4). The water hammer was documented in the shift supervisor's log and the shift supervisor recalled initiating a problem report at the time of the water hammer.

During this inspection, the inspector reviewed the status of the licensee's corrective actions as a result of the water hammer. A problem report documenting the water hammer and corrective action plans could not be located. Based on discussions with System Engineering personnel, a walkdown of the main steam lines had been conducted immediately following the water hammer. This inspection did not include the removal of significant sections of insulation. Request for Engineering Assistance 91-1925 was generated to repair the only pipe hanger found damaged. The damaged hanger was in a non-seismically qualified portion of the system. The licensee had generated a Problem Report (POPR-92-0004) and developed a corrective action plan by the end of the inspection.

The inspectors judged the licensee's actions in response to the water hammer to be weak based on the apparently cursory inspection of the steam line piping and the lack of a coordinated documented corrective action plan due to the missing problem report. Actions to review the operations evolutions performed prior to the water hammer to identify potential improvements to appropriate procedures and preclude repetition appeared to be warranted and were not established. The resident inspectors will review the corrective action documented in the problem report.

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-101, Moderator Temperature Coefficient Determination at 300 PPM Boron;

- SP-146A, EFIC Monthly Functional Test During Modes 1,2,3; and

- SP-349B, EFP-2 and Valve Operability Surveillance.

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 259806 repack emergency feedwater pump, EFP-2;

- WR 269574 repack and valve stem replacement, valve EFV-12;

- WR 284117 & 291942 repack and MOVATS test motor operated steam admission valve (ASV-204) to emergency feedwater pump;

- WR 293884 repack, replace stem, and inspect the seat of valve ASV-204;

- WR 294118 ground in ASV-204 motor while stroking;

- WR 292753 Troubleshooting of Absolute Position Indication for CRD 7-1; and

- WR 293805 EFV-32 troubleshoot valve cycling during surveillance test.

The following items were considered noteworthy.

a. Troubleshooting

On February 7, during the performance of surveillance procedure SP-349A, EFP-1 and Valve Surveillance, the motor operated emergency feedwater isolation valve (EFV-32) stroked closed then open continuously while the test EFIC actuation was present. Since this condition did not normally occur, the SSOD declared the valve inoperable and initiated work request 293805.

A work package was developed to troubleshoot the cause of the valve cycling utilizing maintenance procedure MP-531 "Troubleshooting Plant Equipment" The inspector observed troubleshooting in progress and noted that planned actions were discussed and implemented in a controlled fashion and that work was performed in only one actuation train of EFIC at a time. However, the troubleshooting activities were not methodically documented in a predetermined plan. Licensee personnel involved in the troubleshooting included a senior instrument and controls technician, an instrument and controls supervisor, and the EFIC system engineer. They were knowledgeable, effectively determined the cause of the erratic behavior of the valve, and adequately documented their actions.

The inspector concluded that this troubleshooting was performed in a controlled fashion and was adequately documented; however, a detailed troubleshooting plan was not developed prior to initiating work and revised as information was gathered.

On February 20, following the determination that control rod 7-1 position indication was inoperable (see detail 3.b.), the licensee initiated troubleshooting activities utilizing work request 292753. Prior to initiating troubleshooting, the licensee developed a detailed plan and evaluated potential impacts on plant operations prior to beginning work. The cause of the failed position indication was not identified due to the inability to access portions of the system inside the reactor building during power operations.

In conclusion, the inspectors observed inconsistent performance in the implementation of troubleshooting activities during this inspection.

b. Emergency Feedwater System Outage

On February 4, the licensee removed the steam driven emergency feedwater pump from service to perform several corrective and preventive maintenance activities. These activities included repacking the pump, several manual valves, and one of the two motor operated valves installed in parallel in the steam supply line to the pump turbine.

Inspector observations included the following:

The mechanics who were repacking the pump noted that the work instructions did not provide specific instructions as to the proper orientation of the lantern ring and the design of the ring was not symmetric. They demonstrated a questioning attitude in identifying this issue and appropriately pursued resolution with their supervisor.

Valve ASV-204 is one of the two motor operated valves installed in parallel in the steam supply line to the pump turbine. It was repacked and MOVATS tested as a result of the repack. Apparently due to an error in the sequencing of the testing and/or degraded valve stem condition, ASV-204 was returned to service with a small packing leak. The leak continued to degrade such that the valve, and therefore the steam driven emergency feedwater pump, was removed from service a second time on February 11 to rework the valve. The rework included a seat inspection, valve stem replacement and a repack of the valve. MOVATS testing was performed after the line was repressurized and packing adjustments were made. During the MOVATS testing, plant operators noticed that a DC ground alarm was received while the valve was being stroked. Troubleshooting identified a significant ground in the DC motor in the valve's motor operator. The motor was replaced and the valve successfully retested.

Due to the complications noted in the efficient performance of this system outage, the licensee initiated a Quality Assurance surveillance at the request of Senior Plant Management. The Surveillance plan included assessment of various aspects of performance effectiveness relative to the scheduling, planning and preparation, communication and coordination, and operability determinations performed during the two system outages. The inspectors plan to review the results of the licensee's evaluation in a future inspection.

Overall, surveillance and maintenance activities observed and discussed above were performed in a satisfactory manner in accordance with procedural requirements and met the requirements of the TS.

Violations or deviations were not identified.

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. (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 IEN 89-52

This LER, dated 2/21/90, identified five ventilation fire dampers that may not close under flow conditions. The dampers or fan controls were to be modified to ensure that the dampers would be capable of closing automatically in the event of a fire. Licensee records showed the status of those five dampers to be:

FD-239: Modification per MAR 90-05-01-01 FCN3 to be completed by 7/31/92.

FD-266, 271, & 273: Modification and testing per MAR 90-05-01-01 FCN1 has been completed.

FD-278 Modification and testing per MAR 90-05-01-02 has been completed.

In addition, all five of these dampers were in the current control room "Fire barrier and habitability envelope breach report log," were included in the roving fire watch list of items/areas to check hourly, and the fire watch was checking them hourly.

Supplement 2 to this LER, dated 5/17/91, identified five more fire dampers that may not close under ventilation flow conditions. Licensee records showed the status of those five dampers to be:

FD-223, 224, 225, & 226: Work packages 290644, 290645, 290646, & 290647 had been written to replace the existing four pound spring with a nine pound spring. Parts were received in January, 199[illegible], and the work packages were being sent from the planner to the maintenance shop.

FD-238: This damper had been inspected, found to have a nine pound spring installed, and determined to need no modification.

None of these five fire dampers were included in the current control room "Fire barrier and habitability envelope breach report log" nor were they on the roving fire watch list of items to be checked hourly. The inspector informed the assistant shift supervisor in the control room, who promptly added the five fire dampers to the "Fire barrier and habitability envelope breach report log" and the roving fire watch list of items to be checked hourly. TS 3.7.12 requires that all fire barrier penetrations, including fire dampers, in fire zone boundaries protecting safety related areas shall be functional at all times. Action statement a. requires that, with one or more of these required fire barrier penetrations non-functional, within one hour either: establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch.

The inspector, with the fire systems engineer, observed that fire dampers FD-223 and 224 were located in the wall between the control room and adjacent operations offices, and both of these rooms were within fire zone 8. Fire dampers FD-225 and 226 were located in the floor between the operations offices adjacent to the control room (fire zone 8) and the cable spreading room below (fire zones 6 and 7). In addition to the list of breached fire barriers, the roving fire watch had a procedurally prescribed route to check hourly, which included most safety related areas in the plant. The inspector went with the roving fire watch and observed that the prescribed route included areas on both sides of FD-225 and 226. The inspector also observed the fire detectors on each side of FD-225 and 226 and the monitoring panel for them, which indicated that they were in operation and not in an alarm or trouble condition. The panel continuously monitored the detectors, and would sound a trouble alarm in the event of a fire detector malfunction. The inspector also reviewed records of the last semi-annual surveillance test on these fire detectors, when they were tested with smoke and found to operate properly. The inspector concluded that in this case the requirements of TS 3.7.12 were satisfied.

Procedure CP-137, Fire Barrier and Habitability Envelope Breach Report, Rev. 10, dated 10/14/91, states: "Fire barriers are required to be functional at a' times. Whenever a breached fire barrier is idenitified, a Fire Barrier Breach Report must be initiated; no delay is allowed. A fire watch must be established within one (1) hour after the breach has been identified." The fire systems engineer wrote problem report SY-PR-92-0004 to initiate corrective action for the failure to initiate a Fire Barrier Breach Report. This will address training for personnel involved as well as a review of the GET training on fire barrier breach reporting.

b. (Open) LER 90-03: Present Plant Design Leads to Instrument Fluctuation Exceeding Required Range and Subsequent Technical Specification Violation

In the performance of a surveillance on a control room emergency ventilation chilled water pump, the flow instrument fluctuations were observed to exceed the 2% limit of ASME Section XI. A relief request was written and the surveil nce was completed based on meeting the intent of the ASME requirements for "steady state" conditions. For permanent corrective action, MAR 91-05-25-01 was written and approved by the PRC on 2/27/92. This MAR will replace existing chilled water flow instrumentation with an ultrasonic device to enable a flow reading with less than 2% oscillation.

c. (Open) LER 90-05-02: Calculation of RB Flood Level Shows Level Exceeds that of Safe Shutdown Equipment Due To Design Error Causing Operation Outside Design Basis

Interim corrective action consisted of EOP procedural changes so that operators would shift to recirculation at an acceptable level in the reactor building to prevent the affected instrumentation from becoming submerged. Permanent corrective action will be moving all affected instrumentation to higher levels in the reactor building. This is scheduled to accomplished by the end of Refuel 8.

d. (Closed) LER 90-10: Voltage Dips Caused by Sluggish Voltage Regulator Response Exceed Reg. Guide 1.9 Limits

IR 50-302/91-21 documented the completion of all corrective actions with the exception of completion of the load sequencing voltage monitoring test. The test was done during the mid-cycle 8 outage. Results were that the B EDG voltage and frequency response met Reg. Guide 1.9 limits. However, the A EDG slightly exceeded Reg. Guide 1.9 limits during block one loading. Improvement in the A EDG performance was noted in that during all other block loadings, the voltage and frequency dips were now within Reg. Guide 1.9 limits. The licensee is not ommitted to Reg. Guide 1.9, but plans to accurately monitor EDG voltage and frequency during block loading while performing the related surveillance tests in the future. Further NRC actions to resolve this issue will be tracked by TAC 82149.

e. (Closed) LER 90-19: An Incorrect Motor Installed On a Valve Operator Results in a Condition Outside the Design Basis

During an outage, the installed motor on the pressurizer spray control valve (RCV-14) was found to have a 15 foot-pound rating vice the original design 25 foot-pound rating. Analysis showed that the installed motor was acceptable. Engineering review identified other valves that were not safety related and not part of the 10 CFR 50.49 EQ program that might be susceptible to a similar event. These valves were inspected and found to all have the correct size motor.

6. 10 CFR Part 21 Inspection (36100)

a. (Closed) 50-302/P2191-03 - Part 21 Report From Rockbestos Re KS-500 Silicone Rubber Activation Energy Values For Firewall GR Silicone Rubber Insulated Cable and Firezone R Special Purpose Cable.

In a letter dated June 22, 1990, the Rockbestos Company informed Florida Power Corporation of recent test results regarding KS-500 silicone rubber insulated cable which was purchased by FPC and may have been used in safety related applications.

As a result of questions regarding calculation of activation energy for KS-500 silicone rubber, Rockbestos had recently conducted re-aging and testing of samples to verify or revise Arrhenius time-temperature curves for this insulation.

The results of the testing indicated a need to revise the thermal life data utilized in the original qualification reports. In summary, the test results indicated that the qualified life of 40 years at a continuous operating temperature of 257 degrees F (125 degrees C) could be supported for KS-500 silicone rubber insulated cables where the total integrated radiation dose was 150 megarads or less. This represented a reduction in the activation energy of KS-500 silicone rubber insulation which under certain operating conditions, could result in a reduction of the qualified life of affected cables. This condition was reported to the NRC in a letter dated June 25, 1990, in accordance with 10 CFR 21.

The inspector reviewed the licensee's evaluation and disposition of this information. The licensee determined that the Crystal River Unit 3 Vendor Qualification Package number CABL-R352-16 identified that all KS-500 silicone rubber insulated wire utilized for environmentally qualified circuits were in the Auxiliary Building. The conditions in the two most severe environmental zones in which the affected cable was installed were a maximum 29 megarads and 154 degrees F, which are significantly less than the revised qualification information. Therefore, no further action was required by the licensee.

The inspector noted that revision 2 of Vendor Qualification Package number CABL-R352-16 incorporated the revised qualification values provided by Rockbestos. This 10 CFR 21 report is closed.

(Closed) 50-302/P2191-03 - Part 21 Report From Rockbestos Re KS-500 Silicone Rubber Activation Energy Values For Firewall GR Silicone Rubber Insulated Cable and Firezone R Special Purpose Cable.

b. (Closed) 50-302/P2190-04 - Part 21 Report From Rosemount re: Resistance Bridges Can Exhibit Premature Long Term Degradation Under Certain Combinations of Humidity, Power, and Duration.

In a letter dated August 17, 1989, Florida Power Corporation was informed that twenty-one Rosemount Resistance Bridges (Model 414E/F) which had been manufactured and shipped with precision resistors that exhibited premature long term degradation.

The potential safety hazard was that under certain combinations of humidity, temperature, power, and duration, these resistors increased in value or failed, causing an "open" state. If undetected, this could cause an electronic output not representative of the sensed parameter.

Rosemount identified the cause of the problem as an unrequested, vendor initiated, process change in the manufacture of precision resistors.

The licensee accounted for all of the linear bridges identified as being affected. Seventeen were located in the store room, three had been previously discarded, and one was installed in the high reactor coolant system temperature trip in channel A of the Reactor Protection System. This condition was documented and corrected by nonconforming operations report 89-204. Since the plant was shutdown, there was no immediate impact on safe operation of the facility. The licensee's evaluation indicated that the failure mechanism identified would have caused the instrument string to fail conservatively. Work Request 0249792 was performed prior to restart of the facility to replace the deficient resistance bridge. The remaining deficient components were returned to the manufacturer. This 10 CFR 21 report is closed.

(Closed) 50-302/P2190-04 - Part 21 Report From Rosemount re: Resistance Bridges can Exhibit Premature Long Term Degradation Under Certain Combinations of Humidity, Power, and Duration.

7. Exit Interview
The inspection scope and findings were summarized on March 2, 1992 with those persons indicated in paragraph 1. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

8. Acronyms and Abbreviations

ALARA - As Low as Reasonably Achievable
a.m. - ante meridiem
API - Absolute Position Indication
ASME - American Society of Mechanical Engineers

B&W - Babcock & Wilcox
CCTV - Closed Circuit Television
CFR - Code of Federal Regulations
CRD - Control Rod Drive
ECCS - Emergency Core Cooling System(s)
EDG - Emergency Diesel Generators
EFIC - Emergency Feedwater Initiation and Control
EFP - Emergency Feedwater Pump
EOP - Emergency Operating Procedure
F - Fahrenheit
FCN - Field Change Notice
FPC - Florida Power Corporation
FSAR - Final Safety Analysis Report
gpm - gallons per minute
HP - Health Physics
I&C - Instrumentation and Control
ICC - Inadequate Core Cooling
ICS - Integrated Control System
IFI - Inspector Followup Item
ISI - Inservice Inspection
IST - Inservice Test
LCO - Limiting Condition for Operation
LER - Licensee Event Report
MAR - Modification Approval Record
MOV - Motor Operated Valve
MP - Maintenance Procedure
MUP - Make-up Pump
MW - Megawatt
NOD - Nuclear Operations Directive
NRC - Nuclear Regulatory Commission
NGRC - Nuclear General Review Committee
OTSG - Once Through Steam Generator
p.m. - post meridiem
PM - Preventive Maintenance
PORV - Power Operated Relief Valve
psig - pounds per square inch gauge
QC - Quality Control
QA - Quality Assurance
RB - Reactor Building
RCA - Radiation Control Area
RCP - Reactor Coolant Pump
RCS - Reactor Coolant System
RO - Reactor Operator
RWP - Radiation Work Permit
SG - Steam Generator
SP - Surveillance Procedure
SSOD - Shift Supervisor on Duty
STI - Short Term Instruction
SW - Nuclear Services Closed Cycle Cooling System
TS - Technical Specification
WR - Work Request