U.S. NUCLEAR REGULATORY COMMISSION

REGION 2

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License No:	DPR-72
Report No:	50-302/96-17
Licensee:	Florida Power Corporation
Facility:	Crystal River 3 Nuclear Station
Location:	15760 West Power Line Street Crystal River, FL 34428-6708
Dates:	November 3 through November 30, 1996
Inspectors:	T. Cooper, Acting Senior Resident Inspector B. Crowley, Reactor Inspector, paragraphs E1.1, E1.2, E1.3, E8.1 R. Schin, Reactor Inspector, paragraphs E1.1, E1.2, E1.3, E8.1
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EXECUTIVE SUMMARY

Crystal River 3 Nuclear Station
NRC Inspection Report 50-302/96-17

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 4-week period of resident inspection; in addition, it includes the results of announced inspections by two reactor inspectors from Region II.

Operations

Weaknesses were identified in the implementation of shift briefing and pre-job briefing practices and communication of overtime scheduling. (Paragraph O1.1)

A Non-Cited Violation (NCV 50-302/96-17-01) was identified for failure to, within the time required, take actions required by TS 3.3.1 Condition A which required placing the effected channel in bypass or trip. The fact that the information in the equipment out-of-service log was not sufficient to identify the impact of changing conditions on technical specification requirements and the lack of a requirement to maintain the log with the unit shutdown is indicative of weaknesses in both the log keeping practices and the log keeping process. (paragraph O2.1)

Maintenance

The communications between the maintenance technicians and the operations personnel were identified as good, being concise and conveying accurate information. (paragraph M2)

Engineering

None of the licensee's modifications or test activities to address the eight design issues were completed; all were in various stages of design or testing. In the modification packages and 50.59 safety evaluations reviewed, no errors or oversights in the design approaches being considered were noted. The Preliminary Design Review Board packages were detailed and there appeared to be a good mix of personnel on the Boards. The preliminary 50.59 safety evaluations were more complete than most previous safety evaluations. Two of the analyses and conclusions in the draft 50.59 safety evaluations were questioned, and it was discovered the new 50.59 Review Group was already reviewing the preliminary 50.59 safety evaluations and questioning those same items. (paragraph E1.1)

One Inspector Follow-up Item (IFI 50-302/96-17-02) was identified for the potential overflow of the make-up tank to the auxiliary building while the High Pressure Injection/Low Pressure Injection piggyback lineup is in the recirculation mode. (paragraph E1.1)

The licensee planned to perform several extent of condition reviews for engineering design issues. At the time of this inspection, none of these reviews were completed. All were in a conceptual, planning, or partially completed condition. (paragraph E1.2)

The draft of the new Conduct of Engineering administrative procedure documented expectations and goals for the department and was almost ready to issue. The various oversight groups were all in some state of change to improve management oversight. The changes were potentially positive, but it will take time to determine the net effect. Most of the oversight organizations were working to expectation documents issued by interoffice correspondence. Implementing procedures had not yet been issued. (paragraph E1.3)

An Unresolved Item (URI 50-302/96-17-03) was identified for failure to conduct required Technical Specification surveillance testing on safety related circuitry (GL 96-01). (paragraph E2.1)

The lack of procedural controls for the Design Review Board process and the lack of adherence to the Design Review Board Expectations interoffice correspondence is a weakness. (paragraph E1.3)

Plant Support

An Inspector Follow-up Item (IFI 50-302/96-17-04) was opened pending the completion of an independent review of the adequacy of 10 CFR 50 Appendix R Fire Study and Documentation. (paragraph F2.1)

Report Details

Summary of Plant Status

The unit began this inspection period in Mode 5. The plant originally shutdown on September 2, 1996 due to low turbine lube oil pressure. The unit outage was extended in order to resolve potential Unreviewed Safety Questions concerning emergency diesel generator loading concerns and emergency feedwater system single failure vulnerabilities. On October 4, 1996, the licensee notified the NRC that they planned to remain shutdown for an extended period of time. The extension is to make modifications to several safety systems to obtain additional safety margin for accident conditions.

I. Operations

O1 Conduct of Operations

O1.1 Review of Shift Practices

a. Inspection Scope (71707)

The inspectors reviewed shift turnover and briefing practices and overtime scheduling practices for the operations department.

b. Observations and Findings

Licensee Procedure OI-04, Shift Turnover, includes the expectations for a shift briefing. One of these is a discussion for any planned evolutions for the up-coming shift. Licensee Procedure OI-14, Evolution Briefings, states that each person involved in the activity should feel comfortable in signing FORM OI14-03, Personal Pre-Job Safety Culture Questionnaire. During the power ascension on September 1, 1996, the shift supervisor did not address the planned evolution during the shift briefing. The supervisor did not include the non-licensed building operators in the pre-job briefing. This is a weakness in the administration of the briefing practices.

The sampling review of overtime worked during the outage, did not identify any violations of overtime restrictions; however, the justification for overtime scheduling was not consistently communicated well to operations personnel.

c. Conclusions

Weaknesses were identified in the implementation of shift briefing and pre-job briefing practices and communication of overtime scheduling.

O2 Operational Status of Facilities and Equipment

O2.1 Equipment Out-of-Service Log/Pressure Transmitters (71707, 92700, 92901)

a. Inspection Scope (71707)

The inspector reviewed the Equipment Out-of-Service Log to determine if the appropriate operator actions had been taken for the identified equipment conditions.

b. Observations and Findings

On September 26, 1996, a clearance was issued which isolated four Reactor Coolant System transmitters, to repair a leak in tubing coming off isolation valves RCV-61 and RCV-86. One of the transmitters was RC-3A-PT2, which is a narrow range transmitter that provides input to the C Reactor Protection System (RPS). The other instruments included a wide range transmitter that provides input to the Engineered Safeguards Actuation System, a low range transmitter that provides input to the Remote Shutdown Panel, and a high range transmitter that provides input to the Diverse Scram System (DSS). At the time the tubing leak clearance was issued, none of the transmitters were required to be in service since the plant was in Mode 5 with the Control Rod Drive (CRD) breakers open.

On September 28, 1996, at 7:12 p.m., the CRD breakers were closed to dry out four stators which had exhibited low insulation resistance readings during a preventative maintenance procedure. At 8:10 a.m. on September 29, 1996, a different operations shift observed that with the CRD breakers closed and the RPS input disabled, Technical Specification (TS) 3.3.1, RPS Instrumentation, required four channels of RPS instrumentation to be Operable during Shutdown Bypass operations with any CRD trip breakers in the closed position and the CRD control system capable of rod withdrawal.

The Shift Supervisor on Duty (SSOD) entered TS 3.3.1, Condition A, which requires the inoperable channel to be placed in bypass or trip within one hour.

The licensee's root cause evaluation determined that the primary cause of the event was insufficient documentation of out-of-service equipment. The Equipment Out-of-Service log lists the pressure transmitters as being out of service, but the entry does not identify that one of the transmitters affects the RPS channel. The clearance used to isolate the pressure transmitters for the maintenance does identify the transmitter as affecting RPS. However, this information was not carried over to the Out-of-Service log.

Licensee Procedure OI-07, Control of Equipment and System Status, states that the equipment out-of-service log shall be maintained by the control room operator when in modes 1 through 4. The fact that the information in the equipment out-of-service log did not contain enough information to recognize the impact of changing conditions on technical specification requirements and the lack of a requirement to maintain the log with the unit shutdown was indicative of weaknesses in both the log keeping practices and the log keeping process. The licensee's corrective actions to prevent recurrence included an ongoing review and revision of the practices for the equipment out-of-service log and training for the operations crews to improve their grasp of log keeping practices.

c. Conclusions

The licensee failed to, within the time required, take the actions required by TS 3.3.1, Condition A, which required placing the effected channel in bypass or trip. The licensee identified this condition and complied with the action statement. The unit was in Mode 5 in an extended shutdown and had procedural controls in place to preclude control rod withdrawal. Consequently, this failure constituted a violation of minor safety significance and was treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy (NCV 50-302/96-17-01, Failure to comply with Technical Specification 3.3.1 Condition A action statement).

The fact that the information in the equipment out-of-service log was not sufficient to identify the impact of changing conditions on technical specification requirements and the lack of a requirement to maintain the log with the unit shutdown is indicative of weaknesses in both the log keeping practices and the log keeping process.

II. Maintenance

M2 Maintenance and Material Condition of Facilities and Equipment

a. Inspection Scope (61726, 62707)

The inspector observed the performance of maintenance activities to ensure that all prerequisites were being met, that the applicable procedures and work instructions were adhered to in the performance of the maintenance, and that any identified discrepancies were documented and resolved.

b. Observations and Findings

Work Request (WR) NU 0338460 was performed to troubleshoot and resolve concerns with the indication from source range neutron monitor NI-2. The inspectors witnessed portions of the planning of the WR and the pre-job briefing conducted between operations, maintenance, radiation protection, and engineering. The discussion was comprehensive and discussed prerequisites, expectations, contingency actions and assigned responsibilities. Following completion of the task, the inspector observed portions of the post-maintenance testing. The work was conducted satisfactorily, with no observed problems.

The inspectors observed portions of WR NU 0338528, which was performed on RPS channel B. The licensee was removing each control module from the RPS cabinet, verifying integrity on the circuit card, cleaning the card, recording the part identification number for each card, and returning the card. No problems were observed during the performance of this task.

c. Conclusions

The inspectors concluded that communications between the maintenance technicians and the operations personnel were good, being concise and conveying accurate information. No problems or concerns were identified.

III. Engineering

E1 Conduct of Engineering

E1.1 Design Issues and New Modifications

a. Inspection Scope (37550, 37551)

The inspectors reviewed the licensee's status on resolving the eight design issues identified in a letter to the NRC dated October 28, 1996. In that letter, the licensee described their plans to address eight design issues prior to restarting the plant from a forced outage for secondary plant maintenance. Those plans included modifications and tests to improve design margins of safety-related systems. The inspectors reviewed the licensee's plans, schedules, preliminary modification packages, and preliminary 50.59 safety evaluations, and discussed them with licensee personnel.

b. Observations and Findings

The inspector's review of the eight design issues was as follows:

1) High Pressure Injection (HPI) Pump Recirculation to the Makeup Tank

On October 30, 1996, the licensee identified a concern during a review of a Request for Engineering Assistance (REA). During piggyback operation of the HPI and Low Pressure Injection (LPI) pumps, with recirculation established in accordance with EOP-08, Loss of Coolant Accident (LOCA) Cooldown, the Make Up Tank (MUT-1) is isolated by closure of check valve MUV-65, due to the 200 psig suction pressure to the make up pumps from the discharge of the decay heat pumps. Another way to isolate the MUT is by closing MUV-64 per the instructions contained in EOP-08, prior to the radiation levels within the Auxiliary Building reaching unacceptable levels. Closure of MUV-64 or MUV-65 has the potential to allow the MUT to go solid, creating the potential of releasing reactor coolant into the Auxiliary Building from the

Reactor Building (RB) sump, when the relief valves on the MUT actuate. The licensee is currently evaluating this postulated scenario to determine its effects on the design basis and to develop any needed modifications to the Make-up (MU) system. This item is identified as IFI 50-302/96-17-02, Potential for HPI/LPI recirculation resulting in make-up tank overflow.

The licensee had a conceptual modification package to support procedurally throttling the LPI pump discharge to the HPI pumps. This throttling would be done during recirculation with low flow that could occur during a small break loss of coolant accident (SBLOCA). The throttling would reduce the LPI discharge pressure to below the MUT pressure to allow some of the MUT water to go to the HPI pump suction. This would prevent overflow of the MUT to the auxiliary building (the HPI pump recirculation to the MUT could otherwise fill up the MUT and cause it to overflow).

The inspector reviewed the Preliminary Design Review Board (DRB) modification package and discussed the preliminary design and related issues with Engineering personnel. The Preliminary DRB package included a Modification Approval Record (MAR) Project Description, preliminary Design Input Record, and preliminary 50.59 Screening and Evaluation documents. It described conceptual design alternatives to resolve the issue. The inspector noted no obvious errors or oversights in the design approach being considered. The licensee's schedule was to have a modification package issued by January 22, 1997, and to have the modification installed by mid-February, 1997.

2) HPI System Modifications to Improve SBLOCA Margins

The licensee was considering three modifications to improve HPI design margins; the result of these modifications would eliminate the need for certain manual operator actions in emergency operating procedures (EOPs); and make the HPI system design more like other Babcock & Wilcox (B&W) plants. The modifications being considered were: installing cavitating venturis to limit flow in an injection leg due to a postulated downstream break, installing cross-tie piping between injection legs downstream of the injection control valves, and establishing automatic isolation of normal makeup flow upon an engineered safeguards (ES) actuation. The licensee considered the HPI system, as currently designed, to be fully capable of meeting its design function. The licensee's schedule was to prepare these modifications for installation during the next refueling outage (11R) and to write a justification for continued operation (JCO) as needed prior to restart from this outage.

3) LPI Pump Mission Time

The licensee was currently conducting tests to demonstrate that the LPI pumps could operate for a long time at low flow conditions. Successful testing could result in the decay heat (DH) drop line (which could violate the single failure design criterion) not being required to fulfill the emergency core cooling system (ECCS) long term cooling function for SBLOCA mitigation. The licensee had completed a test run, at a vendor location, using an identical pump from another utility, that demonstrated the ability of an LPI pump to operate for over 30 days at 100 gallons per minute (gpm) without degradation. The licensee expected to have the test report and revised vendor flow guidance in December, and to have revised operating procedures in place by January 31, 1997.

4) Reactor Building Spray Pump

The licensee planned to modify the pump impeller and conduct tests to demonstrate the pump's ability to operate with less required net positive suction head (NPSH). This would resolve the issue of little margin between required and available NPSH for the 1B reactor building spray pump. The licensee's schedule was to start testing, at a vendor facility, using a licensee spare pump, by December 2, 1996. The licensee planned to complete the impeller modification and pump testing and to have two modified impellers returned to Crystal River by mid-January, then installed in the plant and tested by January 31, 1997.

5) Emergency Feedwater (EFW) System Upgrades and Diesel Generator Load Impact

The licensee planned to install cavitating venturis to eliminate the NPSH concern, restore the A train emergency feedwater initiation and control (EFIC) system actuation of Auxiliary Steam Valve (ASV-204) to ensure that EFP-2 auto-starts on a failure of the B side initiate logic or ASV-5, and install motor operators on EFW pump discharge cross-tie valves.

The inspector reviewed the Preliminary DRB modification package and discussed the preliminary design and related issues with Engineering personnel. The Preliminary DRB package included a MAR Project Description, preliminary Design Input Record, and preliminary 50.59 Screening and Evaluation documents. It described conceptual design alternatives to resolve the issue. The inspector noted no obvious errors or oversights in the design approach being considered. The licensee's schedule was to have modification packages completed by mid-January and to have installation and testing completed by mid-February.

6) Emergency Diesel Generator (EGDG) Loading

This issue involved challenges to the rated capacity of EGDG-1A by the continuous, automatically connected loads as well as the loads that are manually connected in the later stages of accident mitigation. The licensee was pursuing increasing the EGDG load capacity by (1) power upgrade to increase one or more of the load ratings, (2) removal and/or reduction of connected loads, and (3) improving the accuracy of the kilowatt (KW) meters used to display the EGDG's output.

- The EGDG power upgrade was to be accomplished by MAR 96-10-05-01, which involved modifications to the turbocharger, the combustion air intercooler, and intercooler piping. The modifications would upgrade the 2000 hour cumulative rating from 2851-3000 KW to 2851-3200 KW and the 200 hour cumulative rating from 3001-3250 KW to 3201-3400 KW. The 0-2850 KW continuous and the 3251-3500 KW 30 minute cumulative ratings would not be affected. Identical modifications have been successfully installed at two other plants with similar EGDGs manufactured by the same vendor (Coltec).

  The inspector reviewed the preliminary DRB package, including the Board meeting minutes and discussed the design with Engineering personnel. The Preliminary DRB package included a MAR Project Description, preliminary Design Input Record, preliminary 50.59 Screening and Evaluation documents, and Coltec Engineering Report. The inspector noted no obvious errors or oversights in the design approach being considered. Engineering personnel indicated that the MAR was targeted for issue by mid-December with installation for EGDG-1A in December. Installation for EGDG-1B was scheduled for February, 1997.

- The removal/reduction or connected loads involved review to determine if some loads could be removed and review of system flow calculations for major pump loads to determine if the calculations could be refined to allow reducing some load values. This review was in process and was being completed in conjunction with the overall EGDG loading calculations being performed as a result of re-rating the EGDGs and modifications to connected equipment, such as adding flow venturis to the EFW pumps. At the close of the inspection, Engineering was still evaluating the schedule for completion of the EGDG loading calculation. The latest draft of the schedule indicated a completion date of March 12, 1997.

- Improving the accuracy of the KW meters involved removing unnecessary burden on the Potential Transformers (PTs) and the Current Transformers (CTs). This was to be accomplished under MAR 96-03-12-01 by eliminating a watt transducer for each EGDG and utilizing the spare conductors from the CTs to the watt/var transducer installed as part of the MAR. Improving the accuracy of the KW meters would increase the EGDG load capability by reducing the instrument error that must be taken into account in calculating EGDG loading.

The inspectors reviewed the preliminary DRB package, including the Board meeting minutes, and discussed the status of the design with Engineering personnel. The preliminary DRB package included a MAR Project Description, preliminary Design Input Record, and preliminary 50.59 Screening and Evaluation documents. The inspector noted no obvious errors or oversights in the design approach being considered. Engineering personnel indicated that the MAR was targeted for issue by early December with necessary calculations complete by mid-December and MAR installation by the end of January.

The inspectors noted that a detailed EGDG loading calculation supporting the above modifications and resolution of EGDG loading questions was in process. This calculation was to support activities necessary to be completed prior to plant restart, including TS and Final Safety Analysis Report (FSAR) changes as well as verifying acceptability of the EGDG loading effects of modifications installed during the outage. The draft schedule for the loading calculation showed completion by March 14, 1997.

7) Failure Modes and Effects of Loss of Direct Current (DC) Power

The licensee planned to perform a Failure Modes and Effects Analysis (FMEA) on the DC power system to address the extent of the condition related to design and operating vulnerabilities that have been identified by postulating the effects of a loss of DC power. The licensee had contracted with two different outside engineering companies to perform and independently review the analysis. The licensee also planned to have the analysis reviewed by their personnel who were familiar with plant design, operation, and operating procedures. The licensee's schedule was to start the project in November 1996 and to complete the analysis by March 4, 1997.

8) Generic Letter 96-06

The NRC issued Generic Letter (GL) 96-06 on September 30, 1996. This GL identified a number of questions relative to equipment operability and containment integrity during design-basis accident conditions. One concern involved over pressurization of containment penetration piping due to elevated temperatures following a postulated design basis accident. To address this concern, the licensee has designed and plans to connect expansion chambers to a number of containment penetrations.

The inspectors reviewed the preliminary DRB package, including the Board meeting minutes, and discussed the status of the design with Engineering personnel. The Preliminary DRB package included a MAR Project Description, preliminary Design Input Record, and preliminary 50.59 Screening and Evaluation documents. Engineering personnel indicated design calculations are complete and in the process of being verified. The MAR was targeted for issue by early December with installation to be complete by early February, 1997.

c. Conclusions

None of the licensee's modifications or test activities to address the eight design issues were completed; all were in various stages of design or testing. In the five Preliminary DRB modification packages and 50.59 safety evaluations reviewed, the inspectors noted no obvious errors or oversights in the design approaches being considered. The Preliminary DRB packages were detailed, and there appeared to be a multi-disciplined composition of personnel on the Boards. The inspectors also observed that the preliminary 50.59 safety evaluations were more complete than most previous safety evaluations. The inspectors questioned two of the analyses and conclusions in the draft 50.59 safety evaluations and found that the new 50.59 Review Group was already reviewing the preliminary 50.59 safety evaluations and was questioning those same items.

One Inspector Follow-up Item (IFI 50-302/96-17-02) was identified for the potential overflow of the make-up tank to the auxiliary building while the HPI/LPI piggyback mode is in recirculation.

E1.2 Extent of Condition Reviews for Design Errors and Design Margin

a. Inspection Scope (37550)

In a meeting with the NRC on November 14, 1996, the licensee stated that they would perform several extent of condition reviews to address identified configuration management problems (i.e., design errors and design margin). The inspector reviewed the licensee's status and schedule for completing those reviews and discussed them with licensee personnel.

b. Observations and Findings

The licensee's extent of condition reviews and their status and schedules were as follows:

1) Develop Timelines for Five Systems

The licensee planned to describe and analyze changes to five systems over the time from initial design through current design, including all significant modifications, FSAR changes, and licensing changes. The timelines for the EFW and EGDG systems were about 90% completed, and the timelines for the other three systems selected (building spray, decay heat, and make-up) were not yet started. The licensee had not yet planned or scheduled these efforts.

2) Review a Sample of Past Modification 50.59's

The licensee planned to review 50.59 evaluations for at least 20 past modifications to assure that they did not erroneously overlook any unreviewed safety questions. At the time of this inspection, the licensee's new 50.59 Review Group had already selected over 20 past modifications for review, had begun the reviews, and was scheduled to complete them by the end of December 1996. The sample size was based upon a total of approximately 120 modifications conducted during the last outage.

3) Perform an FMEA of LOCA, Loss of Offsite Power (LOOP), Loss of DC Power

This item was already discussed in paragraph E.1.1.7 above.

4) Perform an Integrated Safety Assessment of Outage Modifications

The licensee planned to perform an integrated assessment of all modifications installed during this current outage. They planned to do this toward the end of the outage, after all of the modification design packages and related 50.59 safety evaluations were completed.

c. Conclusions

On November 14, 1996, the licensee had described several extent of condition reviews, for engineering design issues, that they planned to perform. At the time of this inspection, none of these reviews were completed. All were in a conceptual, planning, or partially completed condition.

E1.3 New Engineering Design Standards and Oversight

a. Inspection Scope (37550)

As part of the corrective actions for Engineering Performance and Management Oversight and Involvement under the licensee's Management Corrective Action Plan (MCAP), a new "Conduct of Engineering" procedure was to be issued. In addition, some management oversight groups were to be re-organized and strengthened and others were to be formed. The inspectors reviewed the status of these corrective actions and discussed them with licensee personnel.

b. Observations and Findings

1) Conduct of Engineering Procedure

A draft Administrative Procedure had been written and was ready for issue. The inspectors reviewed the draft procedure and found that it was similar in scope to licensee procedures AI-500, Conduct of Operations and AI-600, Conduct of Nuclear Plant

Maintenance. The procedure included responsibilities, "Expectations for Engineering", "Nuclear Engineering Code of Ethics", department goals and objectives, and monitoring and assessment of performance.

2) Review and Oversight Groups

The inspectors reviewed the status of the various oversight groups. The following summarizes the inspection activities:

- Nuclear Safety Assessment Team (NSAT)

  A new manager was assigned in October, 1996, and a new draft organization chart issued. Three new full time positions have been added to the group. The procedure covering the group's activities, AI-512, Conduct of Nuclear Safety Assessment Team, had not been revised to reflect the new organization, responsibilities, and how the group will function. A procedure revision was expected by early 1997.

- Nuclear General Review Committee (NGRC)

  A new chairman was in place, and the committee was to be restructured. The chairman had previously served on a number of plant safety committees and oversight committees at other sites. The chairman planned to restructure the subcommittees so that each will have a chairman from outside Florida Power Corporation (FPC), a vice-chairman from within FPC at the director or assistant director level, two experienced people from inside FPC at the supervisor or higher level, and one person who can represent the functional area. Ad-hoc committees will be established as needed to aid in restart reviews. A special committee of outsiders will be established to review and make recommendations relative to plant restart. An NGRC Charter and Procedure Manual were in place. The chairman indicated that the procedures would require revision to cover the planned changes in the committee structure and activities.

  In addition to review of the above activities, the inspectors attended two NGRC subcommittee meetings. In the Engineering and Technical Support subcommittee meeting, all members were present for most of the meeting, and all participated in the questions and discussions. The inspector noted that in some areas the subcommittee members were not getting a correct understanding of issues, and there was room for more depth and breadth in their questions and in their value added. In the Quality and Regulatory Verification subcommittee meeting, the chairman of the Plant

Review Committee (PRC) presented the results of an on-going assessment of the PRC. A good exchange of information occurred in the meeting with a number of recommendations for improvement in the PRC.

- Design Review Board (DRB)

The DRB was created to take the place of the Design Engineering Review Board, which was made up of essentially Engineering personnel. The licensee had determined that participation of organizations outside of Engineering was essential to implementation of successful plant modifications. The current DRB included personnel from all affected departments. The procedure governing the DRB was not expected to be issued until early 1997. The DRB was operating under "Design Review Board Expectations" issued by interoffice correspondence (IOC). The inspectors reviewed the expectations document and a number of Preliminary DRB packages for MARs in preparation, including meeting minutes for the preliminary DRB meeting. (See paragraph E1.1 above for preliminary DRB packages reviewed.)

The inspectors noted that the Expectations document was not clear relative to who could serve as chairman of the board and who made up the board core. Also, preliminary packages contained different versions of the Expectations document. However, based on review of the meeting minutes, it appeared that all meetings had a good representation of personnel from the affected plant organizations. The licensee stated that the definition of the core members and the designation of chairmen will be refined as more experience is gained and the requirements are proceduralized.

The inspectors attended several DRB meetings. During the meeting to review the conceptual modifications to the EGDG kilowatt meter (to improve accuracy) the inspectors noted that the total core membership required by the expectations IOC was not present. When questioned, the licensee decided that mechanical and structural engineers were not needed for this DRB meeting. The expectations IOC stated that if all core members of the DRB are not present, the meeting would not be held. Discussions with the licensee revealed the belief that since the IOC was not a procedure, that strict adherence was not required. The lack of procedural controls for the DRB process and the lack of adherence to the DRB expectations IOC is a weakness.

- Corrective Action Review Board (CARB)

The CARB was being established by the new Corrective Action Procedure, CP-111, which was scheduled for issue on November 22, 1996. The CARB will replace the Management

Review Panel and will be made up of the direct reports to the Senior Vice President, Nuclear Operations. The primary responsibility of the Board will be to review Root Cause Analysis and appropriateness of corrective actions for all Level A and B Precursor Cards (PCs). The first meeting for the CARB was scheduled for the week of November 25, 1996.

- Precursor Card Screening Committee (PCSC)

  The PCSC was also being established by CP-111. This committee will be primarily responsible for screening and grading all PCs as to their significance. The inspectors noted that CP-111 did not specify who makes up the PCSC. Although CP-111 had not been issued, the PCSC was meeting every day using the guidelines of CP-111 to grade all PCs.

- 50.59 Review Group

  This group was being established as part of the Fuel Management and Safety Analysis Group to review all 50.59 Screening/Evaluations for plant modifications. A preliminary scoping plan and staffing request had been issued. In addition, an Engineering Stand-down meeting had been conducted to apprise all personnel of problems with 50.59 Screening/Evaluations. Staffing of the group was in process. Two full time contract personnel, each with 30 plus years of experience and one FPC employee were working in the group. Hiring of two permanent employees for the group was in process. The implementation plan was not yet issued. The plan was expected by mid-December, with Nuclear Engineering Procedures (NEPs) planned for revision by the end of December. The group was already reviewing 50.59 Screening/Evaluations. As noted in paragraph E1.1 above, the inspectors reviewed preliminary 50.59 Screening/Evaluations for the modifications being currently planned. The inspectors questioned two of the analyses and conclusions in the preliminary 50.59 safety evaluations and found that the new 50.59 Review Group was questioning those same items.

- Plant Review Committee (PRC)

  As noted under the NGRC paragraph above, the PRC was in the process of a self assessment. A number of weaknesses and areas for improvement had been identified and procedure changes were planned. Significant changes were being implemented in the areas of required attendees and the required quorum members in light of recent problems with issues approved by the PRC (e.g. the EFW modification).

c. Conclusions

The draft of the new Conduct of Engineering administrative procedure documented expectations and goals for the department, and it was almost ready to be issued. The various oversight groups were all in some state of change to improve management oversight. The changes were potentially positive, but it will take time to determine the net effect. Most of the oversight organizations were working to expectation documents issued by interoffice correspondence. Implementing procedures had not yet been issued. The lack of procedural controls for the Design Review Board process and the lack of adherence to the Design Review Board expectations Interoffice Correspondence is a weakness.

E2 Engineering Support of Facilities and Equipment

E2.1 Testing of Safety Related Logic Circuits (37551, 92902, 92903)

On April 12, 1996, the licensee, during the initial review of plant conditions in response to GL 96-01, Testing of Safety Related Logic Circuits, identified two circuits which were not being appropriately tested in accordance with TS requirements. These circuits were the auto reset of ES blocks 4 and 6 load sequencing relays and the load shed circuit that trips EFP-1 when EGDG-1A is supplying the ES bus and an HPI signal is received.

TS Surveillance Requirement (SR) 3.3.5.2, Engineered Safeguards Actuation System (ESAS) Instrumentation requires that a channel function test of ESAS circuitry be performed once every 31 days. The failure to test the auto reset for ES blocks 4 and 6 does not comply with the TS SR requirements.

TS SR 3.8.1.10, Alternating Current (AC) Sources - Operating, requires testing of load shedding from emergency buses on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ES actuation signal once every 24 months. The failure to test the load shedding of EFP-1 does not comply with the TS SR requirements.

Licensee Event Report (LER) 96-11 was issued to address this issue on May 13, 1996. Corrective actions included performing the required surveillances and revising the surveillance procedures to correctly test the circuitry in the future. Additional corrective actions included continuing the review for GL 96-01 required review.

On October 22, 1996, while continuing the review for the GL 96-01 corrective actions, 12 contacts were identified in the ESAS logic which were not being tested in accordance with TS. These 12 contacts were all part of the ESAS Reactor Coolant System (RCS) Pressure - Low and Low Low actuation circuits.

Each RCS Pressure bistable has two contacts which provide an actuation signal to the A and B ESAS actuation matrices. When either matrix senses that two of the three RCS Pressure - Low bistables indicates a low RCS pressure, the matrix will actuate ES equipment needed to mitigate the effects of a SBLOCA. When either matrix senses that two of the three RCS Pressure - Low Low ESAS bistables indicate a low RCS pressure, the matrix will actuate ES equipment needed to mitigate the effects of a Large Break LOCA. The licensee determined that two contacts on each of the three RCS Pressure - Low and Low Low RCS Pressure - were not tested by the licensee procedure, SP-130, Engineered Safeguards Monthly Functional Test.

TS SR 3.3.5.2, requires a Channel Functional Test be performed once every 31 days. TS SR 3.3.5.3 requires that a Channel Calibration be performed once every 24 months. The failure to test the 12 contacts did not comply with the TS SR requirements.

LER 96-25 was issued on November 21, 1996 to address this issue. Planned corrective actions included revising the test procedure prior to startup from the current outage and completion of the surveillances prior to escalation to Mode 4. Additional corrective actions included continuing to address the GL 96-01 concerns. Until the review is completed, this issue is identified as Unresolved Item (URI) 50-302/96-17-03, Failure to conduct required Technical Specification surveillance testing on safety related circuitry (GL 96-01).

E8 Miscellaneous Engineering Issues

E8.1 <u>Changes in Quality Assurance (QA), Assessment, and Corrective Action Processes</u>

a. Inspection Scope (40500)

The inspectors reviewed recent audits and self assessments; resulting changes in the QA, assessment, and corrective action processes, and discussed them with licensee personnel.

b. Observations and Findings

1) Recent Cooperative Management Audit Program (CMAP) Audit and Licensee Self Assessment of Audit Process

The CMAP Audit was conducted in August 1996 by personnel from four other utilities. It included an assessment of the effectiveness of internal audit and monitoring programs, corrective action and trending programs, vendor audits and inspection programs, and the interface of Quality Assurance and station organizations. The inspector reviewed the CMAP Audit Report and noted that it contained many findings and recommendations that were both detailed and candid.

Two Audit Program Self Assessments were conducted in October 1996 by licensee personnel. They assessed audit program procedures, planning, and scheduling. These self assessments contained many recommendations for changes in the audit program.

At the time of this inspection, the licensee was making changes in response to the findings and recommendations of the CMAP Audit and the Self Assessments as well as NRC inspection findings. They had revised two procedures to improve the audit program, particularly in the areas of planning and scheduling. The licensee also was in the process of revising another procedure to improve self assessments. The inspector noted that the draft Rev. 5 to NOD-45, Management Self Assessments and Performance Monitoring, included new guidance for line organizations in conducting their own self assessments. The licensee had no plans for conducting training for line organizations on the new self assessment guidance but, after discussing the subject with the inspector, decided to conduct such training.

2) New Procedure for Corrective Action

The inspectors reviewed the new draft corrective action procedure, CP-111, Processing of Precursor Cards for Corrective Action Program. The procedure was scheduled to be approved and issued on November 22, 1996. The procedure was a comprehensive revision of the corrective action procedure implementing the licensee's new graded approach to the corrective action process. New levels of review, with CARB and PCSC reviews as described in paragraph E1.3 above, were included. The inspectors raised a few questions about the content in certain areas and the licensee took the questions for consideration.

c. Conclusions

The inspectors found that the licensee was in the process of making substantive changes to improve the QA audit, self assessment, and corrective action processes.

IV. Plant Support

F2 Status of Fire Protection Facilities and Equipment

F2.1 Appendix R Fire Study Documentation (71750)

The licensee opened Problem Report (PR) 96-401 to address a number of issues, including PRs, PCs, and other items identified with the 10 CFR 50, Appendix R Fire Study and supporting documentation. In the PR, the licensee stated that these issues appeared to be establishing a trend that questioned the adequacy of the documentation and administration controls necessary to support the Appendix R Fire Study. The original Fire Study was not required to be performed under a QA program and as a result was not verified prior to being implemented.

As a result of these concerns, the licensee has initiated a third-party, independent review of the Appendix R program. Pending the completion of the independent review, this item will be identified as an Inspector Follow-up Item (IFI 50-302/96-17-04), Adequacy of 10 CFR 50 Appendix R Fire Study and Documentation.

V. Management Meetings

X1 Exit Meeting Summary

The inspection scope and findings were summarized on December 5, 1996. 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.

X3 Management Meeting Summary

X3.1 On November 14, 1996 a management meeting was held in Atlanta to review the licensee's Corrective Action Plan (CAP). A meeting summary will be issued separately.

PARTIAL LIST OF PERSONS CONTACTED

Licensees

K. Baker, Manager, Nuclear Configuration Management
P. Beard, Senior Vice President, Nuclear Operations
G. Boldt, Vice President, Nuclear Production
J. Campbell, Assistant Plant Director, Maintenance and Radiation Protection
W. Conklin, Jr., Director, Nuclear Operations Materials and Controls
R. Davis, Assistant Plant Director, Operations and Chemistry
D. DeMontfort, Manager, Nuclear Operations
M. Donovan, Supervisor, Rapid Engineering Response Team
R. Fuller, Manager, Nuclear Chemistry
B. Gutherman, Manager, Nuclear Licensing
G. Halnon, Assistant Director, Nuclear Operations Site Support
B. Hickle, Director, Nuclear Plant Operations
L. Kelley, Director, Nuclear Operations Site Support
H. Koon, Manager, Nuclear Production and Nuclear Outage
K. Lancaster, Manager, Nuclear Projects
J. Maseda, Manager, Engineering Programs
P. McKee, Manager, Nuclear Plant Operations Support
R. McLaughlin, Nuclear Regulatory Specialist
W. Rossfeld, Manager, Site Nuclear Services
J. Stephenson, Manager, Radiological Emergency Planning
F. Sullivan, Manager, Nuclear Engineering Design
J. Terry, Manager, Nuclear Plant Technical Support
D. Watson, Manager, Nuclear Security
R. Widell, Director, Nuclear Operations Training
D. Wilder, Manager, Safety Assessment Team

NRC

B. Crowley, Reactor Inspector, Region II (November 18 through 22, 1996)
J. Jaudon, Deputy Director, Division of Reactor Safety, Region II (November 22, 1996)
K. Landis, Branch Chief, Region II (November 7 through 8, November 15, November 20 through 21, 1996)
L. Raghavan, Project Manager, NRR (November 5 through 8, 1996)
R. Schin, Reactor Inspector, Region II (November 18 through 22, 1996)
M. Thomas, Reactor Inspector, Region II (November 5 through 8, 1996)

INSPECTION PROCEDURES USED

IP 37550:	Engineering
IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving and Preventing Problems
IP 61726:	Surveillance Operations
IP 62707:	Maintenance Observations
IP 71750:	Plant Support Activities
IP 82301:	Evaluation of Exercises for Power Reactors

IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
IFI	50-302/96-17-02	Open	Potential for HPI/LPI recirculation resulting in make-up tank overflow. (paragraph E1.1)
URI	50-302/96-17-03	Open	Failure to conduct required Technical Specification surveillance testing on safety related circuitry (GL 96-01). (paragraph E2.1)
IFI	50-302/96-17-04	Open	Adequacy of 10 CFR 50 Appendix R Fire Study and Documentation. (paragraph F2.1)

Closed

Type	Item Number	Status	Description and Reference
NCV	50-302/96-17-01	Closed	Failure to comply with Technical Specification 3.3.1 Condition A action statement. (paragraph O2.1)

Discussed

Type	Item Number	Status	Description and Reference
LER	50-302/96-011	Open	Personnel error causes testing deficiency resulting in condition prohibited by improved TS. (paragraph E2.1)
LER	50-302/96-025	Open	Personnel error causes testing deficiency resulting in condition prohibited by TS. (paragraph E2.1)

LIST OF ACRONYMS USED

AC - Alternating Current
AI - Administrative Instruction
ASV - Auxiliary Steam Valve
B&W - Babcock & Wilcox
CAP - Corrective Action Plan
CARB - Corrective Action Review Board
CFR - Code of Federal Regulations
CMAP - Cooperative Management Audit Program
CRD - Control Rod Drive
CT - Current Transformers
DC - Direct Current
DH - Decay Heat
DRB - Design Review Board
DSS - Diverse Scram System
ECCS - Emergency Core Cooling System(s)
EFIC - Emergency Feedwater Initiation and Control
EFP - Emergency Feedwater Pump
EFW - Emergency Feedwater
EGDG - Emergency Diesel Generators
EOP - Emergency Operating Procedure
ES - Engineered Safeguards
ESAS - Engineered Safety Actuation System
FMEA - Failure Modes and Effects Analysis
FPC - Florida Power Corporation
FSAR - Final Safety Analysis Report
GL - Generic Letter
gpm - Gallons Per Minute
HPI - High Pressure Injection
IFI - Inspection Followup Item
IOC - Interoffice Correspondence
IR - Inspection Report
JCO - Justification for Continued Operation
KW - Kilowatt
LER - Licensee Event Report
LOCA - Loss of Coolant Accident
LOOP - Loss of Offsite Power
LPI - Low Pressure Injection
MAR - Modification Approval Record
MCAP - Management Corrective Action Plan
MU - Make Up
MUT - Make-up Tank
NCV - Non-cited Violation
NEP - Nuclear Engineering Procedure
NGRC - Nuclear General Review Committee
NI - Neutron Instrumentation
NOD - Nuclear Operations Department
NPSH - Net Positive Suction Head
NRC - Nuclear Regulatory Commission
NSAT - Nuclear Safety Assessment Team
OI - Operations Instructions

PC - Precursor Card
PCSC - Precursor Card Screening Committee
PR - Problem Report
PRC - Plant Review Committee
PT - Potential Transformers
QA - Quality Assurance
RB - Reactor Building
RC - Reactor Coolant
RCS - Reactor Coolant System
REA - Request for Engineering Assistance
RPS - Reactor Protection System
SBLOCA - Small Break Loss of Coolant Accident
SR - Surveillance Requirement
SSOD - Shift Supervisor on Duty
TS - Technical Specification
URI - Unresolved Item
VIO - Violation
WR - Work Request