UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

Report No. 50-302/80-24

Docket No. 50-302

License No. DPR-72

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

Facility Name: Crystal River Unit 3 Nuclear Generating Plant

Inspection at Crystal River, Florida

Inspectors: DR Quick for T. F. Stetka, Senior Resident Inspector	9-2-80 Date Signed
DR Quick for B. W. Smith, Resident Inspector	9-2-80 Date Signed
MDHunt M. D. Hunt, Reactor Inspector	9-3-80 Date Signed
Approved by: DR Quick for R. D. Martin, Section Chief, Reactor Operations and Nuclear Support Branch	9-2-80 Date Signed

Inspection Summary

Inspection on June 2 through July 31, 1980 (Report No. 50-302/80-24)

Areas Inspected

Routine inspection by the resident inspectors of plant operations, security, radiological controls, Licensee Event Reports (LER's), licensee action on IE Bulletins and Circulars, review of ILRT data, reactor vessel thermal shock issue, FPC organization changes, nonroutine events, and licensee action on previous inspection items. Numerous facility tours were conducted and some facility operations were observed on the back shift. This report also includes a review of items identified in NUREG 0578 Short Term Lessons Learned Recommendations and the FPC Safety Task Force. The inspection involved 500 hours onsite by two resident inspectors and one region based inspector.

Results

Two items of noncompliance were identified (Infraction - Failure to obtain and analyze samples from test well number 4 and failure to obtain Commission approval prior to changing Environmental Technical Specification 3.1.5 - Paragraph 5; Infraction - Failure to have adequate procedures which caused an operational event resulting in loss of pressurizer level-Paragraph 9c).

DETAILS

1. Persons Contacted

*D. Poole, Nuclear Plant Manager
P. Baynard, Nuclear Support Services Manager
*J. Bufe, Compliance Auditor
M. Collins, Reactor Specialist
*J. Cooper, QA/QC Compliance Manager
W. Cross, Operations Engineer
*S. Johnson, Maintenance Staff Engineer
J. Hancock, Assistant Vice President, Nuclear Operations
*W. Kemper, Plant Training Manager
*K. Lancaster, Compliance Supervisor
*T. Lutkehaus, Technical Services Superintendent
*P. McKee, Operations Superintendent
*G. Perkins, Health Physics Supervisor
G. Ruszala, Chem/Rad Protection Engineer
D. Smith, Technical Support Engineer
*G. Westafer, Maintenance Superintendent
D. Wilder, Health Physics Supervisor
G. Williams, QA/QC Supervisor

Other personnel contacted included office, operations, engineering, chem/rad, and corporate personnel.

*Present at the exit interviews

2. License Action on Previous Inspection Items

(Closed) Inspector Followup Item (302/80-23-01): The licensee has revised Procedure AI-500 as Revision 33 to include the requirement for the CNO and NO to review the shift supervisor log. Review of the shift supervisor log and discussions with operators indicate that the reading is being accomplished.

(Closed) Inspector Followup Item (302/80-23-10): The licensee's investigation of the guide bushing cracks indicate the cracks were caused during casting of the guide bushing and will not propagate with relief valve usage. In addition, guide bushing wall thickness is considerably thicker than the minimum required thickness and therefore the cracks will not affect guide bushing integrity. As a result of these findings, the licensee will re-install the guide bushings. The inspectors reviewed the licensee's findings and the results of the findings by Battelle Memorial Institute and Dresser Industries and have no further questions on this item.

(Closed) Inspector Followup Item (302/80-20-04): This item has been reviewed by the inspectors and was considered to be resolved in IE Inspection Report 50-302/80-23, Paragraph 2.b.(7)ii with respect to drumming of contaminated clothing. In addition the inspectors' review of radiation work permits (RWPs) indicate that the licensee's greater vigilance in this area and the use of the roving Chem-Rad Technician has minimized RWP errors. The use of a frisker when exiting the radiation control area (RCA) was identified as an item of noncompliance in IE Inspection Report 50-302/80-23.

(Closed) Inspector Followup Item (302/80-20-05): The licensee has revised their training tape inventory to include training tapes that cover site emergency and evacuation alarms. The inspector noted the use of these tapes during a new employee training session conducted on July 24, 1980.

(Open) Inspector Followup Item (302/80-23-09): The licensee has completed lubrication of all safety-related pumps and motors. The inspectors reviewed the records of 10 randomly selected pumps and motors for verification of this item. No discrepancies were found. This item remains open pending completion of the formal PM program.

(Closed) Inspector Followup Item (302/80-23-07): The licensee has completed maintenance on all safety-related swing disc check valves of similar design to FWV-45 and FWV-46. The inspectors reviewed several work documents to verify completion of this item. No discrepancies were found.

(Closed) Inspector Followup Item (302/79-25-01): The licensee has responded to this item in a letter dated April 23, 1980. The inspectors have reviewed this response and concur with the licensee's position that additional weld examinations are not required. Action on this item is considered complete.

(Closed) Inspector Followup Item (302/80-23-12): The licensee has performed visual inspections of all reactor coolant pump stud bolts and has found no corrosion with the exception of light surface rust on the studs. The inspectors performed their own inspections on these studs and concur with the licensee's findings. The licensee plans to perform visual inspections of these studs during the next refueling outage to assure that the stud condition has not changed.

(Closed) Inspector Followup Item (302/80-20-02): The crack identified on the Decay Heat Removal System "A" pump shaft was determined to be surface and not volumetric and was not considered to be a problem; however, the licensee did replace the pump casing, shaft and impeller on this pump at the recommendation of Babcock and Wilcox Company. This recommendation was based on the fact that several maintenance operations performed on this pump that required pump disassembly resulted in a reduction in the tolerances on the keyway on the pump shaft. In addition, the pump casing was found to be slightly distorted. The inspectors have reviewed the licensee's action on this item and have no further questions at this time.

3. Review of Plant Operations

The facility is currently shutdown and is presently in Mode 4 operations. The facility is making preparation for plant startup following a refueling outage. Facility startup is planned for early August, 1980.

As a result of the facilities' shutdown status, particular inspection emphasis was placed on radiological controls, plant housekeeping conditions, fire protection controls, and refueling and maintenance operations.

a. Shift Logs and Facility Records

The inspectors reviewed the records listed below and discussed various entries with operations and chem/rad personnel to verify compliance with Technical Specifications (TS) and the licensee's administrative procedures.

(1) Shift Supervisor's Log
(2) Operator's Log
(3) Equipment Out-of-Service Log
(4) Equipment Clearance Order Log
(5) Shift Relief Checklists
(6) Control Center Status Board
(7) Health Physics (HP) Supervisor's Log
(8) Roving HP Log

In addition to these record reviews, the inspectors independently verified selected clearance order tag-outs.

b. Facility Tours and Observations

Throughout this inspection period, facility tours were conducted to observe operations and maintenance activities in progress. These observations included the setting of the reactor vessel head on June 17. In addition the inspectors attended numerous licensee planning meetings to keep abreast of plant activities in progress and planned, and attended Plant Review Committee (PRC) meetings held as a result of nonroutine events (see Paragraph 9 following).

During the facility tours the following observations were made:

(1) Equipment Status: The following items were observed to verify that indicated parameters were in accordance with the Technical Specifications for the current operational mode:

-Equipment Operating Status;
-Area Radiation Monitors;
-Decay Heat Removal System Lineup; and,
-Electrical System Lineup.

(2) Shift Staffing: The inspector verified by spot checks that the operating shift composition was in accordance with Technical Specification requirements. This verification also included a check of the number of licensed operators within the containment building during core alterations.

(3) Seismic Restraints: Numerous seismic restraints (snubbers) on safety-related systems were observed. Several snubbers were observed to contain less than the required hydraulic fluid in the sight glasses. The inspectors discussed this item with the licensee and received a commitment that all safety-related snubbers would be inspected prior to facility startup. This action was verified complete by the inspectors on July 31, 1980. The inspectors are satisfied that the licensee's action on this item is complete.

(4) Plant Housekeeping and Conditions: Storage of materials and components and the cleanliness condition of various areas throughout the facility were observed to determine whether safety and/or fire hazards exist.

The facilities' refueling/maintenance outage status was considered in these observations. The following areas were identified to the licensee as areas that require increased emphasis on housekeeping.

- Boric Acid Storage Tank Areas; and
- Reactor coolant (RC) and miscellaneous waste (MW) evaporator rooms.

The licensee has begun cleanup in these and other areas throughout the facility as work is completed. Housekeeping will continue to be examined during subsequent inspections.

(5) Fire Protection. Fire extinguishers and fire fighting equipment were observed to be unobstructed and inspected for operability.

(6) Radiation Areas. Radiation control zones were observed to verify proper identification and implementation. These observations included review of step-off pad conditions, disposal of contamination clothing and area posting. The inspector also observed personnel activity at the radiation check point to verify the effectiveness of the licensee's controls. It was observed that the licensee is still having problems with radiological rope barriers being left down at the entrance to several contaminated and radiation areas. Effective access controls do not appear to be a problem with high radiation areas. The licensee is continuing with their effort to minimize this problem. The inspectors will continue monitoring this item during routine tours conducted in these areas.

(7) Security Controls. Security controls were observed to verify that security barriers are intact, guard forces are on duty and access to the protected area is controlled in accordance with the Security Plan.

4. Review of IE Bulletins and Circulars

The following IE Bulletins (IEB) and Circulars (IEC) were reviewed to verify adequacy of the licensee's actions.

a. IEB 79-13, Rev. 2 - Cracking in Feedwater System Piping

IE has reviewed the licensee's responses dated July 16, 1979, December 13, 1979 and April 23, 1980 as well as inspections performed by IE documented in Report No. (50-302/79-25). Based on the above, the inspectors are satisfied that the licensee has completed all required actions associated with this bulletin.

b. IEB 80-10, Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to the Environment

The inspectors have reviewed the licensee's response to this bulletin and determined it to be inadequate based on the following:

(1) The nitrogen and hydrogen systems were not included in the list of nonradioactive systems that could potentially become radioactive through interfaces with radioactive systems.

(2) The nitrogen system was contaminated on July 28, 1980 with a resultant unmonitored, uncontrolled release to the environment. The radioactive contamination of the nitrogen system was due to interfacing of the nitrogen system (nonradioactive) with the core flood system (radioactive).

The licensee will submit a new response for this Bulletin by August 31. This Bulletin remains open.

c. IEC 80-11, Emergency Diesel Generator (EDG) Lube Oil Cooler Failures

The inspectors reviewed the licensee's action with respect to this Circular. The licensee held discussions with Colt Industries, the emergency diesel generator (EDG) vendor concerning this problem, and was told that the lube oil coolers used on the licensee's EDG's did not employ soft solder construction; however, a potential problem was identified with the EDG radiators due to the presence of CS inhibitor and soft solder construction. An internal inspection of the 3B EDG was performed with the results indicating little or no corrosion. The licensee is planning to take the following action with regard to the potential corrosion problem in the EDG radiators:

(1) Remove the present inhibitor and replace it with an approved inhibitor;

(2) Check for lead in the system using Surveillance Procedure SP-713, RC Support Systems' Chemistry Surveillance Program, to develop a lead history in the system;

(3) Change Surveillance Procedure SP-354, Emergency Diesel Fuel Oil Quality and Diesel Generator Monthly Test, to check the radiator system for leaks after each test; and

(4) Retain Colt Industries to do an in-depth study of the cooling system and provide the licensee with a list of all components that have materials that are not compatible with any inhibitor used.

The licensee has committed to completing items (1), (2) and (3) above by August 31, 1980. The date for completion of item (4) is still under review.

This item will be Inspector Followup Item (302/80-24-01).

5. Review of Licensee Event Reports

The inspector reviewed Licensee Event Reports (LERs) to verify that:

--The reports accurately described the events;
--The safety significance is as reported;
--The report is accurate as to cause;
--The report satisfies requirements with respect to information provided and timing of submittal;
--Corrective action is appropriate; and,
--Action has been taken.

LER's 79-102, 80-11 and 80-22 were reviewed. This review identified the following items.

a. LER 79-102 reported the failure of core flood tank (CFT) sample isolation valve CFV-12 to close following a sampling evolution. The licensee's troubleshooting of this valve has not identified the cause of this failure and this is the second occurrence of CFV-12's failure to close. The inspector discussed these occurrences with licensee representatives and stated that additional troubleshooting on this valve appeared to be necessary. The licensee acknowledged the inspector's remarks and performed additional examination of the valve during this outage. In addition the licensee will cycle the valve after the plant has returned to normal operating pressure to insure proper operation. Review of the licensee's action with respect to CFV-12 is an Inspector Followup Ite (302/80-24-02).

b. LERs 80-11 and 80-22 reported a failure to take representative samples from test wells as required by the Environmental Technical Specificiations (ETS). The cause of these events was the destruction of the wells due to construction activity to modify the intake canal. As a result, the licensee decided in LER 80-11 not to restore test well 4 (the destroyed well) and to eliminate test well 1.

c. In LER 80-22, however, the license decided to restore test well 1 but continued their decision not to restore test well 4. In addition, the licensee submitted on ETS Change Request (#62) to remove these surveillance requirements, however, this change has not been approved. ETS 3.1.5 requires samples from test wells 1 and 4 on a monthly basis. Modification to the facility that involves a change to the ETS cannot be made without prior NRC approval and is considered to be contrary to the requirements of 10 CFR 50.59. Failure to perform the monthly analysis of test well 4 is contrary to the requirements of ETS 3.1.5. These items, taken collectively, are considered to be an item of noncompliance (302/80-24-03).

d. During the review of the preceding LER's the inspector noted that the Plant Review Committee (PRC) was considerably behind in its review of Nonconforming Operating Reports (NCOR's). There were several reports from 1979 and several from early 1980 that had not been reviewed. This issue was discussed with licensee representatives. As a result of these discussions, the licensee investigated the reason for these delays, and has taken action to rectify the problem. The majority of the overdue report reviews has been completed, and the review procedure, CP-111, "Procedure For Documenting The Reporting and Review of Nonconforming Operations", is being revised to steamline the review process.

The effectiveness of the licensee's actions will be reviewed during subsequent inspection. This item is an unresolved item (302/80-24-04).

6. Followup Review of Integrated Leak Rate Test

Containment integrated leak rate test (ILRT) data were reviewed by an inspector subsequent to the performance of this test. (Refer to inspection report 50-302/80-26.) During this review it was noted that containment penetration #113, #202 and #439 still required local leak rate testing. Penetration #113 needed testing due to unacceptable leakage from an air handling valve located in the penetration line. Penetration #202 needed testing as this was the pressure input point for the ILRT. Penetration #439 needed testing due to valve maintenance being performed on several valves in this line. The inspectors have reviewed the results of the leak rate tests performed on these penetrations to verify satisfactory data. The inspectors have no additional questions on this item.

7. Reactor Vessel Thermal Shock Possibility From Borated Water Storage Tank (BWST) Water

On May 11, 1980 the Oconee Nuclear Plant licensee, Duke Power Company, was notified by Backcock and Wilcox (B&W) that a small break analysis indicates that high pressure injection of BWST water into the primary system would thermally shock the reactor vessel. B&W recommended to the licensee that the BWST minimum temperature setpoint be raised to 90°F.

As a result of this finding the inspector requested the licensee to determine the applicability of these findings to the Crystal River plant. The licensee received a copy of a B&W letter dated May 10, 1980 which further discusses this issue. The letter stated that the feasibility of maintaining the BWST water temperature near 90°F is being investigated. This item will be further reviewed and is an inspector followup item (302/80-24-05).

8. Florida Power Corporation (FPC) Organizational Changes

FPC has made both corporate and site organization changes to provide a more responsive chain of command for nuclear plant activities. Technical Specification Amendments numbers 51 and 55 have been submitted to the Office of Nuclear Reactor Regulation (NRR) reflecting these changes and the approval of these amendments is imminent.

The inspector had no further questions on this item.

9. Nonroutine Events

The following events occurred during this inspection period:

- Identification of Contaminated Storm Drains;
- Inadvertant Engineered Safeguards Actuation;
- Operational Transient Resulting in a Loss of Pressurizer Level Indication;
- Unmonitored Release of Radioactivity;
- Power Operated Relief Valve Block Valve Sticking; and,
- Inadvertant Dropping of Control Rod Drive Lead Screws.

These events are described in detail in the following paragraphs:

a. Identification of Contaminated Storm Drain

On June 12, while performing a routine survey of the south berm storm drain, licensee personnel noted increased activity in the sludge laying in the drain. Since this drain is located outside of the radiation control area (RCA), the licensee took immediate actions to contain the area and make the necessary notifications.

Approximately six weeks prior to the above survey, a hose carrying radioactive fluid to a disposal trailer ruptured in the vicinity of this drain. As a part of the clean-up effort, the licensee was removing contaminated asphalt. It was postulated that a rainfall had caused water to run over the contaminated area and carry the contamination into the storm drain. Prior to this event, no activity above background was identified in the storm chain.

To determine the extent of the spread of contamination the licensee took smears of adjoining areas and samples of the setting ponds to which the drains run, and no additional contamination was detected.

The licensee has removed the contaminated sludge and completed removal of the contaminated asphalt.

The inspectors reviewed this event and observed the licensee's actions. The licensee's actions appear to be adequate and the inspectors have no further questions on this item at this time.

b. Review of Inadvertant Engineered Safeguards Actuation

On July 16, 1980, at approximately 10:10 a.m. an inadvertant engineered safeguards (ES) actuation occurred. The plant was in Mode 5 operation (cold shutdown). The ES actuation signal started a high pressure injection (HPI) pump, realigned the running decay heat removal (DHR) pump to the injection mode, started the emergency diesel generators, and injected about 2023 gallons of water into the reactor coolant system. There were no changes to pressure (approximately 40 psig) or temperature (approximately 120°F) of the primary system; however, pressurizer level increased and pressurizer temperature decreased as a result of the injection. The cause of the injection was the failure to reset one ES channel prior to tripping another ES channel while conducting Surveillance Procedure SP-132, "Engineered Safeguards Channel Calibration." Procedure SP-132 was found to be deficient in that it did not direct re-setting of the channel coming out of test prior to tripping another channel. A temporary procedure change was made to SP-132 to correct the problem prior to re-performance of the procedure. Emergency phone notification was made by the licensee. The inspectors have reviewed the licensee's corrective action and have no further questions on this occurrence.

c. Reactor Plant Operational Transient

On July 18, 1980, while the plant was cooling down an event occurred which involved a loss of pressurizer level indication. The plant conditions just prior to the event were:

- Reactor Coolant System (RCS) Temperature - 172°F
- RCS Pressure - 320 psig
- Pressurizer (PZR) Level - 80 inches
- Make-up Tank (MUT) Level - 54 inches
- Two Reactor Coolant Pumps (RCPs) running

At approximately 0430 hours the packing in the cross-tie valve (MUV-4) between makeup pumps (MUPs) B and C blew out of the valve and caused about 200 gallons of reactor coolant water to spill into the MUP valve gallery area of the auxiliary building. Most of the water was captured by auxiliary building drains and there was no indication of airborne activity. The remaining water was cleaned up.

To repair MUV-4, the running makeup pump (MUP-B) had to be secured and the reactor coolant system (RCS) pressure reduced to enable the plant to be placed on the decay heat removal (DHR) system. As DHR valves were aligned for operation; pressurizer level was noted to decrease from 50 inches to 15 inches and MUT level decreased from 64 inches to 14 inches (an expected transient due to the addition of relatively cooler water into the RCS). After observing PZR level increasing back to the initial conditions, decay heat pump (DHP) A was started and the two running RCPs secured. When this evolution was completed, the operators noted a rapid decrease in PZR and MUT levels. With PZR and MUT levels decreasing, the operators took the following actions:

(1) Attempted to feed MUT from the reactor coolant bleed tank (RCBT); however, soon after starting the waste transfer pump, the pump breaker tripped. The operators then immediately attempted to lineup the running MUP-B to the borated water storage tank (BWST).

(2) Other operators were dispatched throughout the plant to check for leaks and to determine the cause of an apparent rapid RCS cool-down. No leaks were identified and control room indications did not indicate level increases in the building drain sumps. An operator in the Auxiliary Building observed that the decay heat closed cycle cooling (DHCCC) heat exchanger "A" inlet valve (DHV-177) was open and the bypass valve (DHV-17) was closed. The operator than immediately proceeded to the instrument air valves controlling these valves to check the lineup. The operator determined that an incorrect instrument air valve lineup had been made which prevented control of these valves from the control room. Note: With the DHCCC valves aligned in this manner, the full cooling capacity of the decay heat exchanger w available and therefore caused excessive cooling of RCS water passing through the DH system.

(3) As discussed in Paragraph (1) preceding, the operators were attempting to lineup MUP-B to the BWST. After the operators had aligned the valves into what they thought to be the correct lineup, the operators did not note any injection into the RCS and also noted cavitation on MUP-B. After securing MUP-B, the operators began de-pressurization of the RCS and aligned the BWST to the running DHP to enable low pressure injection (LPI) to occur. As RCS pressure reached 18 psig, the operators noted that no LPI was occurring. At about the same time, the operators realized the error with the MUP supply lineup from the BWST, corrected the lineup, and began injection into the RCS with MUP-B. With evidence that MUP-B was injecting into the RCS and also with evidence that there were no RCS leaks caused by the rapid RCS cooldown, the operators continued to use MUP-B to restore both PZR level and MUT levels.

The inspectors reviewed this event and identified the following items:

(1) The valve lineup error with the MUP supply was caused by the operators misinterpreting the control board marking for the MUP cross-tie valve control switches. These switches were just recently installed this outage and, due to the IEEE-273 separa-

tion requirements, were located such that the switches for the A-B MUP's were on the B-C pump board and the switches for the B-C MUP's were on the A MUP pump board.

To insure that this modification is understood by all operators and to insure that there are no other switches installed on the main control board (MCB), Waste Disposal Panel, or Condensate Polishing System Panels that could cause operator confusion due to recent plant modifications, the inspectors determined that the following actions should be completed prior to proceeding to Mode 4:

- Conduct a review of all switches installed this outage as a result of plant modifications to determine what switches will cause confusion; and,

- If any such switches are identified, perform an engineering check to determine if a re-design could be made to eliminate the confusion. If such re-design cannot be performed, then ensure that such switches are clearly marked as to their function and conduct additional traning to assure that operators understand switch operation.

These actions were completed by the licensee on July 20 and were subsequently verified complete by the inspectors.

(2) In addition to the confusion over the MUP cross-tie valves, the inspectors noted that even though LPI was aligned to the BWST, injection did not occur due to RCS pressure of 18 psig plus system head pressure (approximately 50 psig) being greater than BWST head pressure (approximately 45 psig) which kept the check valve between the RCS, BWST and the DHP from opening. This lineup was due to the fact that the plant had re-entered Mode 5 and the DH system (LPI) was re-aligned for normal DH cooling. To enable the lineup from the BWST the operator had to shut a RCS loop suction valve thus removing RCS head pressure from the check valve.

To assure that all operators are aware of all the events as they occurred and that all operators are aware of the valve manipulations required to obtain flow from the BWST while in Mode 5, the inspectors determined that all operators should receive additional training on this event.

This training was completed by the licensee on July 20 and was subsequently verified complete by the inspectors.

(3) The main cause of this event was an incorrect instrument air valve lineup which disabled the DHCCC flow control valves and prevented their operation from the control room. The inspectors' review indicate that this error occurred due to the use of an

inadequate operations procedure. Procedure OP-404, Decay Heat Removal System, Paragraphs 8.1.5.1 and 8.1.5.2, specified the instrument air valve lineup and required a signoff for completion of the steps. Review of these paragraphs and observation of the system layout by the inspectors indicate a high probability of operator confusion since the procedure did not designate the valve numbers of the valves to be manipulated.

To prevent re-occurrence, the licensee has revised OP-404 to designate the required valves by valve number, added a diagram of the valve layout to the procedure, and provided additional training to the operators.

The approval and implementation of inadequate procedures is contrary to the requirements of 10 CFR Appendix B, Criterion V and is an item of noncompliance (302/80-24-06).

Review of these licensee's actions as delineated above is considered to be adequate, and therefore no additional response to this item of noncompliance is required.

d. Unmonitored Release of Radioactivity Through N_2 Supply Header to N_2 Tank Farm

On July 28, 1980, while attempting to perform agitation on the core flood tanks (CFT's) for the purpose of chemical mixing, an event occurred which resulted in contamination of the low pressure nitrogen (N_2) system and an unmonitored release of radioactivity. The plant was in cold shutdown (Mode 5) operations during this event. The following is a discussion of the operations conducted prior to, during, and subsequent to the event.

The CFT's were drained during the refueling outage for valve work and post maintenance hydrostatic testing. A temporary procedure change was in effect to OP-401, Core Flooding System, in order to recharge the CFT's in preparation for plant heatup. On or about July 13, 1980 the low pressure N_2 system was lined up via NGV-4 (low pressure nitrogen header isolation to core flood tanks) to precharge the CFT's to approximately 150 psig. Subsequent to the charging operation, NGV-4 was directed to be closed (not locked) by the procedure and the procedural step had been initialed indicating this action had been performed.

During the period of July 25 to July 27, 1980, the CFT's were charged with concentrated boric acid from the boric acid storage tank and then were filled to the technical specification level of approximately thirteen (13) feet from RCS letdown.

On July 28, 1980 at approximately 0425, a lineup was made from the high pressure nitrogen system to CFT "A" for the purpose of bubbling nitrogen through CFT "A" to mix the contents from chemistry sampling.

At approximately 0510 the plant operator notified the control room of an approximate 300 psig decrease in high pressure nitrogen header pressure. The control room operator then notified a level decrease in "A" CFT of about one foot and a pressure decrease of about 30 psig. The valve lineup was immediately verified and it was discovered that NGV-4 was open. The plant operator immediately closed NGV-4. At approximately 0520 a nitrogen blowdown valve on the upstream side of NGV-4 was opened and found to contain water. An initial smear survey and subsequent detailed analysis of this water indicated the presence of radioactivity (1.4E-2 μci/ml) and boron (2450 ppm). At approximately 0630 a check was made of the nitrogen "TANK FARM" area which located outside the restricted area but on the licensee's property. Relief valve NGV-215 located in the "tank farm" area had lifted and deposited approximately 20 gallons of radioactive water on the ground. Detailed analysis of the free standing water and soil below NGV-215 indicated radioactivity levels of 9.4E-3 μci/ml and 1.2E-2 μci/gm respectively.

The contaminated area below NGV-215 was limited to an area approximately fifteen (15) feet in diameter. At approximately 0730 a Class "A" Emergency was declared based on radioactive contamination of >1000 dpm/100ft^2 in an unrestricted area. The tank farm area and applicable areas in the turbine building were immediately roped off and posted as contaminated areas.

Emergency phone notification was made to the NRC at 0745 and to the State at 0757. Actions were then initiated to: determine the extent of radioactive contamination in the nitrogen system; remove the contaminated soil at the Nitrogen "Tank Farm" area; and to proceed with decontamination of the nitrogen system.

On July 30 at approximately 0830 the licensee secured from the Class "A" Emergency due to soil removal at the "Tank Farm" reducing radioative contamination to less than 1000 dpm/100ft^2. Decontamination of the nitrogen system is still in progress.

The inspectors were onsite just prior to the Class "A" Emergency being declared and monitored the licensee's actions. The inspectors did a review of this event to verify the licensee's actions were adequate. The following items were identified by the licensee:

(1) NGV-4 is a normally "Locked Closed" valve as specified in OP-414, Nitrogen and Hydrogen Systems. NGV-4 was directed to be closed per the temporary change to OP-401 and the procedural step had been initialled indicating this action had been performed. NGV-4 was found in the open position which resulted in contamination of the nitrogen system.

A new step has been added to OP-401 to verify NGV-4 is closed just prior to lining up the Nitrogen System to the Core Flood System. In addition, the licensee reperformed valve lineups required by OP-202, Plant Heatup, OP-419, Liquid Sampling System, and motor control center breaker lineups to ensure there were no additional valves and breakers out of their required position. OP-414, Nitrogen and Hydrogen System, valve lineups will be performed at the completion of nitrogen flushing operations. This will be Unresolved Item (302/80-24-07) pending further inspector review.

(2) Even with NGV-4 open, installed check valve CFV-79 (Check valve inline with nitrogen system and core flood system) should have prevented back flow into the nitrogen header. This valve apparently malfunctioned.

The licensee is planning to replace the internals of DFV-79. This action will be completed prior to going into Mode 3. This item is identified as Inspector Followup Item (302/80-24-08).

(3) The licensee will require independent verification of system valve lineup prior to placing a system in service following refueling ou'ages or extended shutdowns. This will be Unresolved Item (302/80-24-09).

(4) Locks will be provided with a unique key for each valve required to be in a locked position. A Master Key would be maintained in the possession of the Shift Supervisor, Assistant Nuclear Shift Supervisor, and Chief Nuclear Operator. The locks should have the capability for changeout of the locking core. This will be Inspector Followup Item (302/80-24-10).

In addition to the licensee's proposed corrective actions, the inspectors received committments from the licensee to do the following:

- A pressure drop test will be performed on the underground run of nitrogen system piping from the turbine building to the tank farm area to verify that no radioactive water leaked from this piping during the event. This action will be completed subsequent to nitrogen flushing operations. This will be Inspector Followup Item (302/80-24-11).

- If the nitrogen system is not returned to service prior to entering Mode 2 operations, procedures will be developed to ensure adequate surveillance is performed on the temporary nitrogen bottles installed in place of the nitrogen system to assure nitrogen reliability. This will be Unresolved Item (302/80-24-12).

The inspectors have reviewed the licensee's completed and proposed actions and consider them to be adequate.

e. Power Operated Relief Valve (PORV) Block Valve Sticking

On July 22 the licensee attempted to cycle the PORV block valve, RCV-11 and determined that the valve disc had separated from the valve stem. The licensee postulates that since the valve was closed (as a result of the February 26 event) prior to plant cooldown, the contraction of the valve body caused the disc to sieze in place. When operation was attempted with the motor operator, the stem was pulled out of the disc.

The licensee has replaced the disc and the motor operator with a smaller unit to ensure that the motor operator will stall prior to breaking the disc if seizure should re-occur. The licensee's engineering staff and B&W are also reviewing this issue to determine the operational effects. This is an Inspector Followup Item (302/80-24-13).

f. Control Rod Drive Leadscrew Dropping Event

Prior to removal of the Control Rod Drive (CRD) motor tubes, the control rod leadscrews must be lifted up and latched in place within the motor tube. On three occassions, the "piggy-back" tool that is utilized to lift and latch the leadscrew released and dropped the leadscrew. In two instances, the leadscrew drop occurred as the leadscrew reached the end of its travel in the motor tube (G-9 and K-3). The third instance occurred after the leadscrew (M-7) was latched in the fully withdrawn position and the motor and leadscrew were suspended and being transported by the overhead crane.

Review of these events by the licensee and Babcock and Wilcox (B&W), the reactor vendor, indicate that the drops were caused by improper use and consequential overstressing of the piggy-back tool. The overstress condition occurred when the leadscrew reached the top position in the motor tube and the crane continued to pull on the piggy-back tool. The tool soon was overstressed thereby unlatching and dropping the leadscrew. In the event involving leadscrew M-7, the overstressing forces had already caused the piggy-back tool to partially lose its grip such that during the later motor tube lifting the piggy-back tool released.

To prevent further events from occurring, a procedural change was made to require the use of a spring scale between the piggy-back tool and the crane to prevent the tool from being overstressed. The inspectors discussed these events with the licensee and B&W and reviewed the events as discussed above. B&W stated that this is the first facility where leadscrew dropping events have occurred and that the piggy-back tool is a standard tool for the B&W plants. The licensee has submitted details of these events to their engineering offices to determine reportability under Part 21. The licensee's action with respect to issuance of a Part 21 report is an Inspector Followup Item (302/80-24-14).

10. Review of NUREG 0578, Category A, Short-Term Lessons Learned Recommendations

Nuclear Reactor Regulation (NRR) performed a safety evaluation of the licensee's compliance with category A items of NUREG 0578 and described the results of this evaluation in a letter to the licensee dated May 5, 1980. The inspectors verified the completion and adequacy of the following items.

Item 2.1.1 - Emergency Power Supply Requirements Pressurizer Heaters

Emergency Procedure-101, Unit Blackout has been reviewed by the inspectors and determined to contain adequate instructions for the connection of preselected pressurizer heaters to an emergency power supply during a loss of offsite power. The inspectors have no further questions on this item at this time.

Item 2.1.3.a. - Direct Indication of Power Operated Relief Valve and Safety Valve Position

The inspectors reviewed completed Modifications Approval Record (MAR) 79-6-6, and revisions A and B of this MAR. The inspectors performed direct observation of the installation of the accelerometers on the power operated relief valve and code safety valves, the valve position monitoring panel, and the valve position indicators. In addition, the inspectors reviewed the valve monitoring system installation and checkout procedures to verify satisfactory data. There is still a portion of the installation and checkout procedure that must be completed after the unit is returned to power operations. The at-power testing on the valve monitoring system will be reviewed for satisfactory data after completion of this testing and is considered to be an Inspector Follow-up Item (302/80-24-15). The inspectors have no further questions on this item at this time.

Item 2.1.3.b. - Subcooling Meter Installation and Procedures

The inspectors reviewed completed Modification Approval Record (MAR) 79-9-8, TSAT Modification, and revisions A and B of this MAR. The inspectors were present and observed the major portions of Surveillance Procedure SP-122, TSAT meter calibration; and the Functional Checkout Procedure (string calibration checkout sheets) which are included as part of the MAR package. SP-122 and the Functional Checkout Procedure were reviewed to verify satisfactory data. Based on the above review, the inspectors have concluded the installation of the TSAT monitor appears to be adequate. The inspectors have no further questions on this item at this time.

Item 2.1.4 - Diverse Containment Isolation

Completed Modification Approval Record (MAR) 80-1-72, modification of the engineering safeguards actuation system to initiate the closure of the reactor building isolation valves (not required for emergency core cooling system) on high pressure injection signal, and revisions A through I of this MAR have been reviewed by the inspectors for adequacy of installation.

In addition, completed MAR test procedure 80-1-72 and revisions A through I of this test procedure have been reviewed by the inspectors to verify satisfactory test results. The inspectors have no additional questions on this item at this time.

Item 2.2.1.a. - Shift Supervisor Responsibilities

Shift Supervisor responsibilities that provide direct management of ongoing safety related operation are delineated in procedures AI-200, Organization and Responsibility and AI-500, Conduct of Operations.

Item 2.2.1.b - Shift Technical Advisor

The functions of the Operations Technical Advisor (OTA) and the Shift Technical Advisor (STA) are delineated in procedures AI-200 and AI-500. The licensee has established interim measures (Enclosure 2 to AI-200) that address the use of temporary STA's until a permanent STA staff is established.

The inspector reviewed procedures AI-200 and AI-500 as applicable to the OTA and STA functions and also examined the manning roster. During this review the inspector noted that one interim STA did not hold a Senior Reactor Operators (SRO) license. This issue was discussed with licensee representatives since use of this individual (who holds a current Reactor Operators license) was not consistent with the Nuclear Reactor Regulation (NRR) staff evaluation of the implementation of Category "A" Lessons Learned requirements.

The licensee acknowledged the inspector's comments and stated that the individual would be removed from the manning roster and not utilized as an STA until NRR concurrence was obtained.

Based on these reviews and the licensee's actions the inspector has no further questions on this item at this time.

Item 2.2.1.c - Shift and Relief Turnover Procedures

Procedure AI-500 has been revised to assure an adequate and complete shift turnover. The licensee has implemented the use of various checklists (Shift Relief Checklist, Operational Status Checklist, and Critical Plant Equipment/ Parameters Checklist) to provide guidance for a complete and systematic turnover between the off-going and on-coming shift. Shift turnovers and the use of these checklists have been observed by the inspectors on numerous occasions.

Item 2.2.2.a - Control Center Access

The responsibility and authority for control center access is delineated in procedure AI-500. The information included in this procedure appears adequate to assure effective access control.

Items 2.2.2.b - Onsite Technical Support Center (TSC) and 2.2.2.c - Operational Support Center (OSC)

A description of, and the manning requirements for, the TSC and OSC are delineated in procedure EM-102, Staffing of Technical Support Center and Operational Support Center. In addition, the Emergency Plan Procedure, EM-100, has been revised to include reference to EM-102 as the implementing procedure and to include a description of the TSC and OSC. In addition, on May 29, the inspectors observed performance of an emergency plan drill (see NRC Inspection Report 50-302/80-23) which included the manning and operation of these support centers.

The inspector has no further questions on this item at this time.

11. Review of Florida Power Corporation (FPC) Safety Task Force Priority Items (Letter Dated May 2, 1980)

On May 2, 1980, Florida Power Corporation (FPC) submitted a letter to NRR listing the FPC Nuclear Safety Task Force Priority Items and describing the actions that would be taken by the licensee to resolve these items. This listing was reviewed by the Office of Inspection and Enforcement (IE) and it was determined which items should be completed prior to plant startup.

The following items were reviewed by the inspectors and verified to be complete and adequate to insure safe plant operation. These reviews included both document review and witnessing of selected testing.

Item 1 - Verify That NNI Testing Reviews Already Completed Are Adequate and Documented

The inspector was present when the following tests were conducted in accordance with Procedure PT-454:

(1) Verification of auctioneering diodes and power supplies capability to carry bus voltage.

(2) Determination of the AC voltage level (86.8VAC) that will produce the 22 VDC on the NNI internal buses and trip the power supplies through the power supply monitors.

(3) Verification that breakers S1 and S2 could be closed with the power supply monitor in service with a load on the NNI X bus. The timing of S1 and S2 was verified at 0.5 second by brush recorders.

(4) Verification that vital AC to breakers S1 and S2 could be interrupted and restored without tripping S1 and S2.

The inspector observed the data being taken and later reviewed the results of this test. During a followup inspection (IE:RII 50-302/80-16), the inspector examined the buffer card and buffer amplifier module which had malfunctioned due to pin misalignment. This module was found as a result of a complete module by module examination by the licensee.

The manufacturer has conducted tests which confirmed the cause of the loss of the +24DC bus voltage. As a result, the method for installing amplifier cards in the buffer amplifier module has been modified to provide for visual inspection of the pins to insure proper pin alignment.

The above actions appear to have isolated the problem, verified the extent of the problem, and provided a method for controlling the installation of the amplifier card in the buffer amplifier module. The inspector has no further questions at this time.

Item 2 - Procedural Controls Review Emergency, Abnormal, Surveillance and Functional Test Procedure

The inspectors have reviewed the following procedures and determined the licensee has met their commitment for establishing procedural controls of selectable sources for indication and control:

- Emergency Procedure - 114 - Loss of ICS and NNI Power;
- Surveillance Procedure - 505 - Operability and Functional Check of Events Recorder and Annunciator System;
- Surveillance Procedure - 300 - Operating Daily Surveillance Log;
- Operating Procedure - 501 - Reactor Non-Nuclear Instrumentation;
- Test Procedure 80-3-63G, H and I - Functional Testing of Vital Bus Failure, Loss of Off-Site Power, MAR Safety Circuits and to Verify All Emergency, Abnormal, and Operating Procedures are Adequate for Operator Responses; and
- Subcooling Monitor Functional Test Procedure (The Subcooling Monitor Functional Test Procedure is covered in detail under NUREG 0578, Item 2.1.3.b.).

The inspectors have no further questions on this item at this time.

Item 3 - Initiate a More Extensive Surveillance Program On the Events Recorder System

Surveillance Procedure - 505 - Operability and functional check of events recorder and annunciator system, has been reviewed by the inspectors and determined that the licensee has met their commitments on this item. The inspectors have no further questions on this item at this time.

Item 4 - Review Test Program and Results

The testing of the modifications was either witnessed by the inspectors or the test results reviewed in detail. Before a test was conducted, the test procedure contained in the MAR, or developed for the MAR, was reviewed by Plant Review Committee (PRC).

This method of review appears to control the functional testing of the modifications and the inspectors have no further questions on this item at this time.

Item 7 - Check of Control Rod Drive Mechanisms (CRDM) Insulation Resistance

The licensee has completed taking CRDM insulation resistance readings. The inspectors reviewed the results of the CRDM insulation resistance checks which are contained in Preventive Maintenance Procedure No. 114, CRDM Electrical Checks. No out-of-specification readings were found. The inspectors have no further questions on this item at this time.

Item 8.b. - Structural Stress Calculations

The inspector reviewed the calculations submitted by the licensee.

While the calculations concluded that no damage occurred to the pressurizer, reactor coolant drain tank, the safety valve discharge piping or the pipe supports, certain hanger/restraint inspections were required. The supports identified as RCH35 and RCH55 were inspected by the licensee to insure that no snubber damage had occurred. The snubbers were tested in accordance with Maintenance Procedure MP 130, Snubber Maintenance. A review of the test results was made and the records indicate that the two snubbers are acceptable. The inspector has no further questions in this area at this time.

Item 8.c. - Pressurizer PORV and Safety Valve Disassembly and Refurbishment

The licensee has installed a new Power Operated Relief Valve (PORV) and new safety valves on the pressurizer. The new PORV was installed after the licensee determined that a cracked seat existed on the installed valve and that the lead time for valve repair was excessive; however, during performance of work on the existing PORV, it was determined that the original valve was oversized (passed approximately 152,000 pounds per hour of steam vs. approximately 100,000 pounds per hour as stated in the FSAR). A safety evaluation was performed by B&W, stress calculations were performed by Gilbert Associates (GAI) (see Item 8.b. preceding), and an overall evaluation was performed by the licensee to verify that the plant systems could support this size valve and that plant operations would not have been adversely affected by its use during previous operation. The new PORV installed on the pressurizer has a capacity of approximately 118,000 pounds per hour. Since this capability is also in excess of the FSAR valve, an additional safety evaluation was performed to support use of this new valve.

The inspectors accomplished the following activities to verify the licensee's actions:

- Safety valve and PORV installations were observed for proper installation and valve nameplate data;
- Nameplate data from installed safety valves were compared with licensee records to insure traceability of the valves and to verify installation data and proper lift settings;
- Nameplate data from the installed PORV were compared with licensee records to verify valve sizing; and

- The PORV safety analysis performed by B&W, GIA, and the licensee and the Modification Approval Records (MARs) numbers M-80-7-65, M-80-7-65B involving installation of the oversized PORV were reviewed.

As a result of these observations and reviews, the licensee's valve installations appear adequate and the inspectors have no further questions at this time.

The inspectors' reviews of the licensee's paperwork indicate a possible problem with data documentation. This item is discussed in the following paragraph entitled "General Discussion".

Item 9 - HPI Termination Criteria Procedure Revision

Emergency Procedure-106, Loss of Reactor Coolant or Reactor Coolant Pressure, has been reviewed by the inspectors. The inspectors determined that the 20°F subcooling requirement for cutting back and securing HPI flow during a small break or overcooling transient has been incorporated into this procedure. The inspectors have no further questions on this item at this time.

Item 10 - Tube Rupture Procedure Revision

Emergency Procedure-104, Steam Generator Tube Failure, has been reviewed by the inspectors and determined to meet the licensee's commitment for incorporating Babcock and Wilcox tube rupture guidelines into this procedure. The inspectors have no further questions on this item at this time.

Item 11 - Proper OSTG Level Procedural Revisions

Emergency Procedure - 106, Loss of Reactor Coolant or Reactor Coolant Pressure, has been reviewed by the inspectors and determined to contain the revised B&W small break guidelines concerning proper OTSG level during HPI and manual RCP trip. The inspectors have no further questions on this item at this time.

Item 12 - Feed and Bleed Procedure Revisions

Emergency Procedure - 108, Loss of Steam Generator Feedwater, has been reviewed by the inspectors and determined to contain necessary instructions for the initiation of HPI cooling upon a total loss of secondary heat removal capability. The inspectors have no further questions on this item at this time.

Item 13 - OTSG Overfill Protection Procedure

Emergency Procedure - 105, Steam Supply System rupture, has been revised to include a statement that the main steam isolation vlaves will not automatically open. In addition, new valve closing instructions are included in this procedure for selected feedwater valves to prevent inadvertent opening on OTSG pressure recovery or actuation of bypass.

Emergency Procedure - 108, Loss of Steam Generator Feedwater, has been revised to include a section entitled "Recovery from Emergency Feedwater Actuation". In addition, the procedure has been revised to verify feedwater valve status.

Emergency Procedure - 113 - Plant Shutdown From Outside Control Room, has been revised to include operator verification of proper feedwater system status prior to leaving the control room.

Abnormal Procedure - 112, Loss of Electrical Supplies, has been revised to recognize a loss of the startup transformer as a unit blackout if the startup transformer is supplying all the unit's electrical loads.

The inspectors have no further questions with this item at this time.

Item 14 - Establish Minimum Conditions for Voluntarily Entering Degraded Modes of Operation

Administratve Instruction 500, Conduct of Operations, has been revised to include the guidelines listed under Item 14, Enclosure B, of the Florida Power letter dated May 2, 1980. The inspectors reviewed the revision to verify adequacy. The inspectors have no further questions on this item at this time.

Item 15 - ICS Rod Withdrawal Inhibit Reset Upon RPS Reset

Operating Procedure - 204 - Power Operations, has been revised to include instructions to reduce the Reactor Demand High Load Limiter Setpoint along with reduction in High Power RPS Trip Setpoint during operation with 3 reactor coolant pumps. The inspectors have no further questions on this item at this time.

Item 16 - Operator and Instrument and Control Technician Training

The licensee has completed the training of all the licensed operators and instrument and control technicians with respect to NNI/ICS operation, NNI/ICS modifications, available indications during system upsets and NNI/ICS power losses, and review of emergency and abnormal procedures necessary to shut down the plant, with emphasis on how to regain normal control during NNI/ICS power losses. The inspectors reviewed the training syllabus and the course attendance records of selected licensed operators and instrument and control technicians to verify completion of this training. The inspectors have no further questions on this item at this time.

Item 17 - Operator Training for CR-3 Event and for Plant Modifications Made As A Result of the Event

The licensee has completed the training of all the licensed operators with respect to the operational transient event of February 26 and for plant modifications made as a result of the event. The inspectors reviewed the training syllabus and the course attendance records of selected licensed operators to verify completion of this training. The inspectors have no further questions on this item at this time.

Item 20 - Secondary Steam Relief Blowdown Settings

The licensee has adjusted all the secondary steam relief valve blowdown rings such that blowdown will be limited to less than 5%. The blowdown ring settings were determined experimentally by sending a number of the relief valves to Wyle Laboratories for testing.

The inspector reviewed the following documentation associated with setting the relief valve blowdown settings:

- Procedure MP-109, OYSG Relief Valve Removal and Replacement, Revision 8, and the temporary changes associated with this procedure;
- All Work Requests involved with completion of this work;
- Blowdown test documentation from Wyle Laboratories;
- Dresser Industries letter which provided the licensee with valve blowdown ring settings; and
- Completed data sheets from procedure MP-109 documenting completion of blowdown ring setting and re-installation of the valve.

The inspector also examined the valve installation and verified that valves set by Wyle Laboratories and Dresser Industries were lockwired and sealed. The inspectors have no further questions on this item at this time.

Item 21 - I&C Technician 24 Hour Coverage

The licensee has revised procedure AI-500, Conduct of Operations, to require 24 hour coverage by a Nuclear Instrument and Control Technician during operations (Modes 1-4). The inspectors reviewed AI-500 and discussed planned shift coverage with licensee representatives. The licensee actions appear adequate and the inspectors have no further questions on this item.

Item 22 - Inspection and Replacement of Improperly Installed Fiber Clamps

As the result of notification from Babcock and Wilcox (B&W) concerning the possible shorting of wires in Reactor Protection System (RPS) instrumentation due to improperly installed fiber wire clamps, the licensee inspected all identified instrument modules. The inspection was accomplished in accordance with Work Request (WR) 10433 and the B&W letter and was completed on June 2, 1980. The licensee determined that they did not have the fiber clamps identified in the letter.

The inspector reviewed WR10433, the B&W implementing letter, and examined a typical instrument module. The inspector has no further questions on this item.

Item 23 - Provide Temporary Backup Air System for Main Feedwater Startup Control Valves

A diesel-driven air compressor is now located outside the turbine building and is tied into the service air system and provides a backup sypply to the instrument air system which will provide greater instrument air system reliability for controlling of main feedwater (MFW) and emergency feedwater (EFW) valves during loss of offsite power.

Emergency Procedure - 101 - Unit Blackout, has been revised to include general instructions for starting the diesel driven air compressor upon a loss of offsite power.

Abnormal Procedure - 118 - Loss of Instrument Air, has been revised to include the necessary starting instructions for placing the diesel driven air compressor on the line.

The inspectors have no further questions about this item at this time.

Item 24 - Procedure Revisions for Decay Heat Pump Tripping

Operating Procedure - 404 - Decay Heat Removal System and Emergency Procedure - 112, Loss of Decay Heat Removal System, have been revised to require the decay heat pumps to be tripped anytime their suction supply valves are not in the open position. The inspectors have no further questions on this item at this time.

Item 26 - Establish Administrative Controls to Minimize Containment Access in Mode 1

Operating Procedure - 417, Containment Operating Procedure, has been revised to minimize containment access during Mode 1 operations. The inspectors have no further questions on this item at this time.

Item 27 - Isolation of Letdown and Makeup

Emergency Procedure - 114, Loss of ICS and NNI Power, has been revised to include instructions for isolating letdown in the event of loss of ICS or NNI Power supplies. In addition, the licensee has completed the training of all operators on this item. The inspectors reviewed the procedure, training syllabus and selected course attendance records to verify completion of this item. The inspectors have no further questions on this item at this time.

Item 28 - Reactor Coolant Pump Restart Interlock

The inspector reviewed the Modification Approval Record (MAR) 80-5-62 which provides instructions for the installation and testing of a keyhole switch on the control board. This switch provides a bypass of permissive interlocks to permit the starting of any RCP in an emergency situation. One keylock switch with ganged contacts is provided rather than an individual switch for each RCP as stated in the licensee's May 2, 1980 letter. The key is under administrative control. The modification appears adequate and the inspectors have no further questions in this area at this time.

Item 31 - Field Changes to NNI/ICS Systems should be performed in accordance with design control requirements

Item 32 - NNI/ICS Changes should include specific reference(s) to installation and maintenance precautions identified by the equipment supplier

The inspector reviewed an engineering directive issued regarding modification to the NNI/ICS system. All modifications were to be performed in a manner that is equivalent to the requirements of Safety Related Engineering Procedures with minor exceptions.

The inspector reviewed the following Safety Engineering and Administrative Procedures.

a. Safety Engineering Procedure No. 2, Design Control

   This procedure provides for the translation of design requirements and QA requirements into specifications, drawings, and design documents.

b. Safety Engineering Procedure No. 5, Document Approval and Control

   This procedure provides the necessary reviews and approvals for engineering documents.

c. Safety Engineering Procedure No. 6, Design Control to Modification Approval Record (MAR) Control

   This procedure directs the flow path for approval of MARs.

d. Administrative Procedure No. 8, Interim Drawing Control

   This procedures provides directions for drawing revisions as a result of changes relating to MARs.

The procedures appear adequate to control items 31 and 32. The inspector has no further questions in this area at this time.

Item 33 - Bailey 820 Signal Monitor Output Seal-In Problems

The signal monitor seal-in problems were caused by the misalignment of pins in a buffer amplifier module. It was determined that the possibility of electrical shorting exists when a "too wide module", which utilizes only one set of the two sets of plug pins available at the back of the cabinet, is inserted into the cabinet frame. This shorting could result in the loss of one or both 24 volt power supplies.

To prevent this occurrence, an insulating material was installed in the back of the "too wide module" frame to prevent shorting of the back of the module with the unused plug pins. In addition a warning card has been posted inside the cabinet door stressing the importance of the insulating material.

The inspector examined the corrective actions taken by the licensee which included observations of insulating material installation and the insertion and installation of the two wide modules. The inspector has no further questions on this item at this time.

Item 34 - Main Plant Parameter Redundant Indications

This modification was performed in accordance with MAR 80-3-64I and tested in accordance with T/P 80-3-64I. The inspector reviewed the MAR and observed the functional testing.

During the test, readings were recorded from the indicators both before and after various bus breakers were opened. When the NNI(x) power feed was removed, the newly installed (y) powered indicators continued to function. The only exception was the operation of the B loop OTSG level indicators. The wiring for these indicators (x and y) appeared to be reversed because the (x) indicator continued to function and the (y) indicator failed upon loss of the (x) power. When the NNI(y) power feed was removed the opposite effect occurred with these instruments. All other indicators, power available lights, and power failure annunciators operated as expected. The licensee has corrected and rechecked the OTSG "B" level indicators.

There appears to be adequate redundant indication to provide the operator with the needed operating information should the (x) or (y) power be lost to the NNI system. Based on this review and test observations, the inspectors have no further questions on this item at this time.

Item 35 - Atmospheric Dump Valve Closure Upon Loss of ICS

The inspector reviewed MAR 80-4-96 which modified the atmospheric dump valve circuit to insure the valves close upon loss of ICS power. The test of this modification, which involved placing the dump valves at 50% open and verifying valve closure upon loss of ICS power, was observed by the inspector.

This modification appears to perform as intended and the inspector has no further questions on this item at this time.

Item 36 - NNI(x) and ICS(AC) Power Automatic Transfer

The addition of automatic switches to transfer the power feeds to the NNI and ICS systems from the vital to the regulated instrument bus was installed under MAR 79-11-9B. The testing was accomplished under Test Procedure TP 79-11-9B. The inspector reviewed the test procedure and the results. The procedure appeared adequate and the test results were in accordance with the requirements of the test procedure. The inspector has no further questions at this time in this area.

Item 40 - Emergency Feedwater Pump Automatic Start Circuit Reliability

The licensee has modified the emergency feedwater pump auto start and anticipating reactor trip circuitry in accordance with MAR 79-11-67G to assure that no single power failure will disable these functions.

The inspectors reviewed MAR 79-11-67G and test procedure 79-11-67L, Functional Test of Emergency Feedwater Automatic Actuation and Anticipating Reactor Trip to verify the licensee's actions. In addition the inspectors witnessed the performance and successful completion of the functional test.

The inspectors have no further questions on this item.

Item 45 - Modification of EFW Pump Auto Start Logic Circuit

The inspectors reviewed Modification Approval Record (MAR) 79-11-82A through 82G (Modification of EFW Pump Auto Start Logic Circuit) and Test Procedure 79-11-82 (Motor-Driven Emergency Feedwater Pump 3A (EFP-1) Auto Circuit Electrical Test). In addition, the inspectors observed the performance of Test Procedure 79-11-82. The licensee actions on this item appear adequate and the inspectors have no further questions on this item.

Item 46 - PORV And PORV Block Valve Logic

The inspector reviewed the modifications made to the logic circuitry for the Power Operated Relief Valve (PORV) in accordance with MAR 80-3-62 and witnessed the performance of the functional test. The test verified that the PORV will close when either NNI(x) or NNI(y) power fails and that the override switch functioned as intended.

This modification appears to have corrected the problem that occurred when the NNI power was interrupted on February 26, 1980. Since the block valve logic is already supplied froman independent vital power supply, no modification to this circuitry was required. The inspector has no further questions on this item at this time.

Item 47 - Pressurizer Spray Valve Logic Modification

The pressurizer spray valve logic was modified in accordance with MAR 80-3-62A, B, and C. The inspector reviewed this MAR and witnessed the functional testing of the modified logic circuitry. This testing verified that the pressurize spray valve would close upon loss of any 24 volt power supply.

This inspector has no further questions on this item at this time.

Item 48 - ICS(x) Power Automatic Transfer

The inspector reviewed the drawings required for the modifications made under MAR 79-11-9B. Test Procedure TP 79-11-9B was developed to functionally test the transfer of 120VAC power from the vital bus to the regulated instrument bus for the NNI/ICS systems.

The results of the test were reviewed by the inspector and appeared satisfactory. The inspector has no further questions at this time in this area.

Item 49 - Loss of ICS and NNI Power Annunciators

The inspector reviewed MAR 80-3-65 that added separate annunciator windows to inform operators of the loss of NNI(x), NNI(y) or ICS power sources. During the redundant instrument testing (Item 34) the inspector observed that the window annunciation and power available lights funtioned as designed.

The inspector has no further questions on this item at this time.

General Discussion

The licensee has made several extensive modifications to provide redundant indication and control of various operating parameters. The inspectors have reviewed drawings, observed work in progress and observed functional testing.

The importance of drawing control was stressed in view of the many modifications that have been made. This aspect was stressed when the licensee advised that a Field Change request which was approved and implemented in 1975 had not been added to design drawings. This modification was minor and did not impact on safe operation of the plant, but did indicate a need to review the Field Change Notices (FCN's) to assure that drawings have been updated to reflect any changes in plant configuration. The inspector has identified this need as an Unresolved Item 302/80-24-16.

12. Unresolved Items

Unresolved items are those items for which further information is required to determine whether they are acceptable or items of noncompliance. Unresolved items are contained in paragraphs 5.d., 9.d., and 10.

13. Exit Interviews

The inspectors met with licensee representatives (denoted in paragraph 1) on a weekly basis and at the conclusion of the inspection on July 31, 1980. During these meetings the inspectors summarized the scope and findings of the inspection as they were detailed in this report. During these meetings the items of noncompliance, unresolved items and inspector followup items were discussed.