U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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

Report No: 50-302/97-19

Licensee: Florida Power Corporation

Facility: Crystal River 3 Nuclear Station

Location: 15760 West Power Line Street
Crystal River, FL 34428-6708

Dates: October 26 through November 29, 1997

Inspectors:
S. Cahill, Senior Resident Inspector
T. [illegible]per, Resident Inspector
S. [illegible]hez, Resident Inspector
P. F[illegible], Reactor Inspector, sections E1.2, E8.5, E8.17
T. Johnson, Senior Resident Inspector, Turkey Point, Sections O8.1, E8.1
E. Lea, Project Engineer, sections O8.2, M8.1, E8.21
L. Mellen, Reactor Engineer, section E8.22
M. Miller, Reactor Inspector, sections E1.1, E8.2 - E8.4
R. Reyes, Resident Inspector, Turkey Point, sections E8.14 - E8.15
G. Salyers, Emergency Preparedness Specialist, section P8.1
R. Schin, Reactor Inspector, sections O7.1, E8.6 - E8.10, E8.16
M. Thomas, Reactor Inspector, sections E8.11 - E8.13

Approved by: K. Landis, Chief, Projects Branch 3
Division of Reactor Projects

Enclosure 2

EXECUTIVE SUMMARY

Crystal River 3 Nuclear Station
NRC Inspection Report 50-302/97-19

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 5-week period of resident inspection; in addition, it includes the resu s of announced inspections of open restart items by regional inspectors.

Operations

The licensee's operational actions to respond to the perturbation in RWP-3A performance were very good. They displayed an excellent sensitivity to preserving several options of decay heat removal, took prompt interim corrective actions and developed a methodical and appropriate action plan to secure the pump and remove the suction blockage (Section O1.2).

The evolution to establish a pressurizer steam bubble was well controlled and displayed good coordination among operators (Section O1.3).

The inspectors concluded that even though the number of identified clearance errors has been increasing, the impact of the errors has been reduced and that the licensee has been identifying the errors mainly through review programs intended for that function (Section O2.1).

Although Operations' performance problems remained, they were generally minor and promptly and appropriately corrected. The inspectors have observed that the licensee has committed to continue to focus on improving performance in this area long beyond restart of the plant. Additionally, the licensee consistently encouraged self-identification of problems by all groups. Several of their initiatives required long time frames and were not deterrents to restart of the plant. The inspectors concluded that overall Operational performance was adequate (Section O4.1).

The inspector concluded the licensee was performing adequate training on the new Emergency Operating Procedures and had diligently tracked and responded to operator problems and questions. (Section O5.1).

The inspectors concluded that the licensee's progress to date on the Management Corrective Action Plan (MCAP II) continued to be satisfactory, and there was one item related to licensing basis information which remained to be accomplished prior to plant restart (Section O7.1).

The inspectors observed that Nuclear Quality Assessment (NQA) activities continued to be appropriately focused. Licensee line management continued to utilize NQA to follow up on suspected problems within the various departments, giving management an independent assessment of performance. The licensee devoted appropriate management attention to screening problems in the corrective action system. No recent, notable errors were observed (Section O7.2).

The inspector determined the licensee's corrective actions appeared appropriate to address configuration control problems. Recent trends have been positive; however, continuing monitoring for effectiveness and management oversight is warranted (Section O8.1).

Maintenance

An instrument technician displayed a poor level of skepticism and self-checking when electing not to perform steps in a procedure. This resulted in an inadvertent actuation of the pressurizer spray valve that fortuitously had minimal consequences. However, the licensee recognized the implications of the problem and took appropriate corrective actions (Section M1.1).

The control of work in the protected trains of engineered safeguards systems was weak, because it lacked adequate procedural guidance and there was not a clear definition as to what constituted a protected train. A weakness was also noted in changes to modification scope being issued directly to the field, bypassing the operations shift supervisor on duty (Section M2.1).

Engineering

The inspectors concluded that the licensee had implemented and completed superior programs for Generic Letter 96-01 and Decay Heat Closed Cycle System Failure Modes and Effects Analysis. Both programs reviewed were technically adequate and were implemented in accordance with licensee requirements, commitments and NRC regulations (Section E1.1).

The inspector concluded that the revised emergency diesel generator loading calculations demonstrated that the generators have the capacity and capability to accept the design basis loads as required by 10 CFR 50, Appendix A, Criterion 17 (Section E1.2).

The inspector concluded that the licensee had performed a very good investigation and root cause determination, although the documentation was difficult to follow and the corrective actions didn't clearly match the identified causes (Section E1.3).

The inspector concluded that the licensee's final assessment was thorough and adequately resolved any concerns with the potential for loose parts damage in the RCS (Section E1.4).

Several concerns were noted during the functional testing for the B emergency diesel. Differences existed between the test limits and precautions and approved annunciator response procedures. The test allowed generator stator temperature levels above the installed meter's maximum reading capability. The vendor reference manual for the temperature relay for generator stator temperature was not updated after a change in the installed relay in 1979.

The test logs were weak and did not provide sufficient detail to reproduce actions taken during the testing. (Section E2.1).

A Violation (VIO 50-302/97-17-01) of 10 CFR 50.59 requirements was identified for an inadequate safety evaluation of the modification functional test procedure for the B emergency diesel generator (Section E2.1).

A Non-Cited Violation (NCV 50-302/97-17-02) was identified for performing work on safety related diesel generator clutch pads without approved procedures or work instructions (Section E2.1).

The licensee's actions to correct numerous tank parameters had appropriately addressed the difficulties associated with the engineering structure, work prioritization, and available resources to perform the work. The tank calculations were thorough and the results had been appropriately incorporated into the required procedures (Section E8.15).

The inspectors concluded that the licensee's Control Complex Habitability Envelope (CCHE) leakage analysis, described in their November 10, 1997 letter to the NRC, failed to recognize the potential for Control Rcom Emergency Ventilation System (CREVS) fans to cause a substantial amount of CCHE leakage during accident conditions. Licensee personnel stated that they would address the CCHE leakage due to CREVS fans in a revised submittal to the NRC (Section E8.16).

The inspector agreed that the electrical cable operability evaluation had a sound basis. An Inspector Follow-up item (IFI 50-302/97-17-03) was established to ensure NRC review of the final or long term resolution of the cable ampacity issue (Section E8.17).

A Violation (VIO 50-302/97-17-04) was identified for inadequate design control related to thermal relief valves on various heat exchangers (Section E8.20).

An Inspector Follow-Up Item (IFI 50-302/97-17-05) was established to track the resolution of improved Technical Specification setpoint program deficiencies prior to entry into Mode 4 (Section E8.22).

Plant Support

A review of an open item on inconsistent Emergency Action Level classification determined that the licensee had developed diverse and challenging training scenarios and had adequately addressed the original concerns (Section P8.1).

The inspectors assessed the licensee's performance in the five areas of continuing NRC concern in the following sections: the assessments are limited to the specific issues addressed in the respective sections.

NRC AREA OF CONCERN	ASSESSMENT SECTION																										
	O4.1	O7.2	O8.1	O8.2	O8.3	M8.1	E1.1	E1.2	E8.1	E8.2	E8.3	E8.4	E8.5	E8.6	E8.7	E8.8	E8.10	E8.11	E8.12	E8.13	E8.14	E8.15	E8.17	E8.18	E8.19	E8.21	E8.22	P8.1
Management Oversight	G	G	G	G	A	G	S	G	A	S	G	G	G	G	G	A	A	G	G	G	G	G	G	G	G	A	G	G
Engineering Effectiveness						G	S	G	G	S	G	G	G	G	A	A	A	G	G	G	A	G	G	A	G	A	S	
Knowledge of Design Basis			A	G		G	G	G	A	G	G	A	G	G	A	A	A	G	G	G		A	G	A	A	A	S	G
Compliance With Regulations	A	G	A	G	A	G	G	G	G	G	G	G	G	G	G	A	A	G	G	G	A	A	G	A	A	A	S	G
Operator Performance	A		A	G	A	G								G		A	G			G								

S = Superior G = Good A = Adequate/Acceptable I = Inadequate
Blank = Not Evaluated/Insufficient Information

O4.1 Operator Performance and Communication Observations

O7.2 Licensee Self-Assessment Activities

O8.1 (Closed) VIO 50-302/97-02-01; Failure to Follow Equipment Control Procedure Requirements (FPC Restart Issue O-13A)

O8.2 (Closed) LER 50-302/96-21-00; Delayed Entry Into Technical Specification Required Action Caused by Inadequate Documentation of Out-of Service Equipment Requirements for a Modification

O8.3 Reportability Program
(Closed) EA 97-094 (4 examples: 01013, 01023, 01033, and 01043); Repeat Failure to Make Timely Reports to the NRC
(Closed) VIO 97-08-01; Inadequate Corrective Action and Procedure for External Reporting Requirements

M8.1 (Closed) LER 97-002-01; Out of Calibration Fuel Pool Water Level Transmitters

(Closed) VIO 97-01-04; Failure to Perform Technical Specification Surveillance for Spent Fuel Level

E1.1 Design Control Process

E1.2 Emergency Diesel Generator Loading Calculations

E8.1 (Closed) VIO 50-302/96-08-01; Failure to Take Timely Corrective Action to Address Issues and Actions For Makeup System Audit Findings and Excessive Vibration on a Spent Fuel Pool (SFP) Pump Fan Motor (FPC Restart Issue OP-24)

E8.2 (Closed) LER 96-011-00, LER 96-025-00, and LER 97-003-00 through 005; Personnel Errors Caused Testing Deficiencies (GL 96-01)

E8.3 (Closed) VIO 50-302/97-05-03; Incorrect Information in Annunciator Response Procedure for Inverters.

E8.4 (Closed) VIO 50-302/97-07-01; Failure to Follow Procedure CP-111 for the Processing of Precursor Cards (PC)

E8.5 (Closed) URI 50-302/96-201-07; EDG Not Protected Against Water Spray from the Fire Protection System Sprinkler

E8.6 (Closed) EA 95-126, VIO I.C.2 (04013); Corrective Actions for an Inadequate Curve 8 (Two STI's and a Revised Curve 8A and 8B) were Also Incorrect

E8.7 (Closed) EA 96-365, C (03013): Inadequate Corrective Actions for 10 CFR 50.59 Evaluation Errors for Inadequate Containment Peneration Surveillance

E8.8 (Closed) EA 97-162 (01013); Inadequate Safety Evaluations for Added Operator Actions for Design Basis SBLOCA Mitigation

E8.10 (Closed) LER 96-24-01; Plant Modification Causes Unanalyzed Condition Regarding Emergency Feedwater

E8.11 (Closed) EA 96-365, EA 96-465, EA 96-527, VIO B (Example 1) (02013): Failure to Update Applicable Design Documents to Incorporate Design Information

E8.12 (Closed) EA 96-365, EA 96-465, EA 96-527, VIO B (Example 2) (02013): Failure to Include Applicable Design Information in the Design Input Requirements for a Modification

E8.13 Followup on Restart Issue Resolution - BWST NPSH Concern (FPC Restart Issue D-18)

E8.14 (Closed) LER 50-302/97-017-00: Personnel Error Caused Inadequate Electrical Separation Of High Pressure Flow Indicators (FPC Restart Issue D53A)

E8.15 (Closed) VIO 50-302/EA 95-126 NOV II.B: Failure to take adequate corrective action for required tank volumes, level, and suction points. (FPC Restart Issue OP-12)

E8.17 (Closed) VIO 50-302/97-01-09: Inadequate Corrective Actions for Cable Ampacity

(Closed) LER 50-302/97-31-00: Inadequate Cable Sizing Due to Nonconservative De-rating Factors Could Reduce the Cable Remaining Qualified Life

E8.18 (Closed) IFI 50-302/97-02-05: Outstanding Issues Associated with the Emergency Diesel Generator Power Upgrade Modification

E8.19 (Closed) VIO 50-302/97-11-06: Failure to Follow Licensee Procedure NEP-254

E8.21 (Closed) VIO 50-302/96-09-06, Erroneous Calculation Inputs and Inservice Inspection Boundary

(Open) LER 50-302/97-038: Engineering Oversight Resulted in Operation Outside Design Basis of Waste Disposal System

E8.22 (Closed) EA 95-16: Use of Nonconservative Trip Setpoints for Safety-Related Equipment

(Closed) LER 50-302/94-006-00 through LER 50-302/94-006-06; Deficiency in Understanding of Technical Requirements Leads to Nonconservative Safety Systems Setpoint and Violations of Improved Technical Specifications.

P8.1 (Closed) IFI 50-302/97-08-03: Variations in the Classification and Interpretation of the EALs by the Emergency Coordinators.

Report Details

Summary of Plant Status

The unit remained in Mode 5 through the inspection period, continuing in the outage that began on September 2, 1996. The reactor coolant system (RCS) started the period filled to a normal pressurizer level with a nitrogen over pressure of approximately 40 psig. Train "A" of forced decay heat removal system flow was operable and in service to support train "B" maintenance, modifications, and testing for the B Emergency Diesel Generator (EDG) radiator upgrade and other routine emergency equipment train-related work. Both once-through steam generators (OTSG) remained filled to a normal inventory with a nitrogen blanket, and one was always preserved as available to support use as a backup decay heat sink, if needed. On November 18, 1997 a vacuum was established in the main condenser using auxiliary steam. On November 20, 1997 a pressurizer steam bubble was established to control RCS pressure.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707 the inspectors performed routine reviews of plant operations which included shift turnovers, response to emergent problems, log reviews, coordination meetings, and restart activities. Significant observations are discussed in the following paragraphs.

O1.2 Raw Water Pump Discharge Pressure Perturbation (71707)

On October 26, 1997, an intermittent low suction alarm was received on the operating Nuclear Services and Decay Heat Sea Water (RW) pump RWP-3A. The alarm cleared, but the pump discharge pressure stabilized at a lower than expected value. The operators questioned this, and the licensee promptly initiated a thorough investigation. They verified that vibration levels were normal; temperatures of components cooled by RW were normal, but the discharge pressure was below the low acceptance limit for the RW pump surveillance. Consequently they declared the pump inoperable per Technical Specification 3.4.6 and complied with the Limiting Condition for Operation (LCO). The corresponding B train RWP and Decay Heat Removal System (DH) components were unavailable due to modification and maintenance work so the licensee technically did not have a fully operable train of decay heat removal. However, they developed a comprehensive action plan with contingencies for other acceptable methods of decay heat removal and developed appropriate procedural guidance for each of their contingency and action plan items. They also frequently monitored RWP-3A and verified its parameters were not degrading and that it was fulfilling its core cooling function. The licensee immediately suspended the B train work, significantly perturbating their outage schedule, and initiated actions to restore RWP-3B and the B train DH equipment to service. On October 30, 1997, after verifying alternate decay heat removal options were available, they expeditiously secured RWP-3A, sent divers into the suction pit to investigate the source of the low discharge pressure, and restored the pump to service. Discharge pressure increased to the expected value and the pump operated normally after removal of a wooden shim with an attached lanyard that was partially blocking the suction of the RWP. The shim had been used earlier in the month for maintenance on the seawater intake structure that eventually feeds the RWP suction. The licensee also determined that an identical piece of wood had been noted floating in the intake during that work, but no action to account for it had been taken. This raised significant concerns with the inspector and licensee management regarding the adequacy of foreign material exclusion practices used for that work and why the wood had not been retrieved. The licensee initiated a root cause investigation which was not completed at the end of this report period. The inspector will review the completed investigation to disposition the above concerns. Regardless, the licensee's operational actions to respond to the perturbation in RWP-3A performance were very good. The licensee displayed an excellent sensitivity to preserving several options of decay heat removal, took prompt interim corrective actions and developed a methodical and appropriate action plan to secure the pump and remove the suction blockage.

O1.3 <u>Establishment of a Steam Bubble Inside the Pressurizer (71707)</u>

On November 20, 1997, the inspectors observed activities in the control room associated with the establishment of a steam bubble inside the pressurizer using the pressurizer heaters. The evolution observed by the inspector was slow and well controlled, in part due to several pressurizer heater groups being out of service for maintenance. These out of service heaters limited the rate at which the operators could heat up the water inside the pressurizer. Once the steam bubble was formed, the operators established saturation conditions inside the pressurizer by utilizing local vent valve RCV-227 to the reactor coolant drain tank. Good communication was observed between the control room operators and the operators at the vent valve. Saturation conditions were maintained at approximately 50 pounds per square inch gauge (psig) RCS pressure and 297 degrees Fahrenheit pressurizer temperature. The bulk RCS temperature during the pressurizer bubble evolution remained constant at approximately 81 degrees Fahrenheit. The inspectors concluded that the overall pressurizer bubble evolution was performed effectively.

O2 Operational Status of Facilities and Equipment

O2.1 Use of Clearances and Tagging Orders

a. Inspection Scope (71707)

The inspectors performed a follow-up of several recent errors in the control of clearances in the plant. In addition, the inspectors reviewed the licensee's clearance error trend analysis performed for the period from May 1997 through October 1997.

b. Observations and Findings

On November 3, 1997, Precursor Card (PC) 97-7568 was written to document that the C main condensor water box discharge Amertap screen was found open, contrary to the required position on an active clearance. Investigations revealed that the clearance required the screens to be tagged in the "open" position, but the controls for the screens had two labeled positions: "Operate" and "Backwash". The clearance tag terminology and the in-plant control labels did not match, leading to confusion on the part of the plant operator. The operator tagged the control switch in the Backwash position, with the screen open, rather than in the Operate position, with the screen closed. The licensee repositioned the screen to the correct position. New tags have been manufactured to state screen positions in Operate and Backwash modes. The licensee verified that no manways were open on the condensor outlet waterboxes, preventing a security breach. There was no impact on either safe operation of the plant, security integrity or personal safety.

On November 7, 1997, PC 97-7641 was written to document an error on clearance 97-10-017 for work on the B building spray (BS) system. The licensee discovered that valve BSV-99 was tagged closed on the A BS train instead of BSV-98 on the B BS train. The licensee verified that BSV-98 was locked closed during this period, but was not tagged. This error in the development of the clearance order was missed by all reviewers. At the time of discovery, the licensee was in the process of releasing the clearance. The licensee verified that the clearance error did not compromise the safety of any work performed on the system.

The licensee conducted a trend analysis of clearance errors from May 1997 through October 1997. The analysis revealed a slowly increasing trend in errors during that period. The licensee concluded that approximately 60 percent of the clearance errors during that period were attributable to administrative preparation, approval, and release of clearances. The inspectors reviewed the trend analysis and determined that the licensee's conclusions and proposed corrective actions were appropriate.

c. Conclusions

The number of identified clearance errors is slowly increasing. The licensee has identified this trend and is taking actions in an attempt to correct this trend. The inspectors concluded that even though the number of identified errors is increasing, the impact of the errors has been reduced and that the licensee is identifying the errors mainly through program barriers intended for that function. The errors that are occurring appear to be mainly personnel attention to detail types of problems and not a programmatic issue. The inspectors will periodically monitor the licensee's program to assess the effects of the proposed improvement initiative.

O4 Operator Knowledge and Performance

O4.1 Operator Performance and Communication Observations

a. Inspection Scope (71707)

The inspectors continued to assess examples of Operations performance for improvement in operator's questioning attitudes and communications practices. Operations Readiness is a restart restraint item on the NRC Restart List.

b. Observations and Findings

As discussed in several previous reports, minor problems continued to occur, indicative of weaknesses in Operations' communications with other departments and inconsistent questioning attitudes. One example was a late Surveillance Procedure (SP)-157A completion on November 13, 1997 when a manual calculation, necessary to compensate for an out of service meteorological tower instrument, was informally turned over to an oncoming shift and not completed within the required surveillance periodicity. Another example was inappropriate procedural guidance observed by an inspector in entry 9711.06 to the Operations Study Book (OSB). An Operability Concerns Report (OCR) had been completed by engineering to address concerns with spurious trips of molded case circuit breakers from PC 97-6906. Part of engineering's conclusion was recommended actions to take for a single spurious trip. Operations excerpted this guidance and issued it as an OSB entry to direct operator action for a trip. The OSB is a tool to promulgate information of interest to operators but not procedural direction. The licensee has other mechanisms with appropriate reviews to issue direction, such as Short Term Instructions. Operations management recognized the error and removed the inappropriate guidance.

An example of poor operator awareness and questioning attitude involved a malfunction of the control room display computer that caused it to stop processing data. The only outward indication of this was that the digital clock on the display monitors did not advance. The malfunction occurred at approximately 2 a.m. but was not detected until after shift turnover at 7:30 a.m. by the oncoming crew. The safety significance of this oversight was minimal because plant conditions had not changed so the data displayed on the monitors was still relatively accurate and main control board instrument indications were available and functioning for all the computer parameters. While reviewing this problem, the inspector observed that several systems in the main control board contain internal clocks, none of which were synchronized to indicate the same time. The clocks on the plant display computer, annunciator response computer, and a parameter trending computer all differed by several minutes. A prominently displayed digital clock on a monitor displaying video camera coverage of main steam relief valve tailpipes was over 20 minutes incorrect. The Safety Parameter Display System (SPDS) monitor did not indicate a time, but the inspector considered it's internal clock likely to be inaccurate also because the operators indicated that computer clocks were rarely synchronized and there was not a requirement to periodically do so. The inspector considered this problem to have minimal safety significance in the current plant conditions, but if operating and a plant trip occurred, the licensee would experience significant difficulty assembling a valid sequence of events using the data from these systems. This would hinder their investigation and correction of the cause of the trip. The licensee recognized the implications of this problem, initiated PC 97-8007 for corrective action, and was evaluating requirements to synchronize the clocks automatically or perform a periodic manual synchronization.

Another example involved a site drain system (SD) valve left out of position on November 1, 1997, due to a skipped procedure step following a tank release. However, this example was identified soon thereafter by an oncoming Turbine Building operator, who recognized the incorrect position on his rounds. The valve was a cross-tie for release radiation monitors so its incorrect position did not result in any notable problem. The operator generated PC 97-7539 to investigate and correct the cause of the error, and Operations Management completed a thorough incident investigation. The second operator's observation was a good example of a questioning attitude, but the original operator's error was an example of poor work practices.

A positive example involved questioning of operators regarding an inspector-identified test deficiency that created conflicting guidance between the test procedure and the operator's annunciator response (AR) procedures as discussed in Section E2.1. The operators consistently responded that they would follow their AR procedure in the absence of specific guidance in the test that allowed exceeding the AR limits. This was also Operations management's expectation. The consistent response indicated to the inspector that Operations' management expectations were being translated and understood by their personnel.

A last example involved an inadvertent positioning of a switch for emergency diesel voltage control by an operator during testing on November 13, 1997. The significance of the misposition was negligible because it transferred control to a local station and was immediately recognized by the operator (due to an alarm) and corrected. The operator informed his supervision of the error and the licensee took appropriate action considering the operator self-identified the deficiency. The inspector observed that the licensee consistently encouraged self-identification of problems and focused on the solution to prevent recurrence of the problems, versus punitive discipline. The inspector considered this appropriate and supportive of their goal to improve overall performance.

c. Conclusions

The inspectors concluded that these examples indicate that Operations' performance problems remain but were generally minor and were being promptly and appropriately corrected. Examples of poor communications between Operations and other organizations continued to occur, but the licensee was identifying them and addressing the causes appropriately. The inspectors have observed that the licensee consistently encouraged self-identification of problems by all groups. The inspectors have observed that the licensee has committed to continue to focus on improving performance in this area long beyond restart of the plant. Several of their initiatives required long time frames and were not deterrents to restart of the plant. The inspectors concluded that overall Operational performance was adequate.

The inspector assessed the licensee's performance, with respect to this restart-related issue, in the five NRC continuing areas of concern:

- Management Oversight - Good
- Engineering Effectiveness - N/A
- Knowledge of the Design Basis - N/A
- Compliance with Regulations - Adequate
- Operator Performance - Adequate

O5 Operator Training and Qualification

O5.1 Emergency Operating Procedure (EOP) and Requalification Training

a. Inspection Scope (71707)

The inspectors observed a session of the licensee's training on EOPs done as part of licensed operator requalification training. The EOPs have undergone significant overall revision, and all operators were being trained on the draft procedures, prior to finalizing and implementing them. The inspector assessed the training for adequacy.

b. Observations and Findings

The inspector observed strong oversight and ownership of his crew's performance by the Shift Supervisor. Several malfunctions, both minor and significant, were run on the simulator to assess the crew performance. The crew responded capably and the inspector observed that they were very proficient at troubleshooting instrument and control malfunctions. The training staff emphasized a logical approach to diagnosing the failures, which the crew readily implemented. The inspector observed some minor individual performance and communication deficiencies which were also noted by the training staff and shift supervisor and appropriately addressed with the individual. The inspector also observed some minor uncertainty with the intent and minor difficulty implementing some of the new EOP steps. However, these were diligently recorded by the training staff for resolution. Based on interviews with several of the operators, the inspector determined that their concerns and questions had been consistently recorded, and answers were always provided to them in a timely manner. The inspector did not have any notable concerns with the observed items.

c. Conclusions

The inspector concluded the licensee was performing adequate training on the new EOPs and had diligently tracked and responded to operator problems and questions. No concerns were identified.

O6 Operations Organization and Administration

O6.1 On October 28, 1997, the licensee announced that the Energy Supply Strategic Business Unit (SBU), along with the company's Power Marketing Group and Purchased Power Resources, will be combined with Nuclear Operations to form one SBU focused on the company's generation assets. Senior Vice President of Nuclear Operations Roy Anderson will head the new organization as Senior Vice President of Energy Supply. The licensee expected the change to be fully implemented in January 1998.

O7 Quality Assurance in Operations

O7.1 Management Corrective Action Plan (MCAP II)

a. Inspection Scope (40500)

The NRC Confirmatory Action Letter to Crystal River of March 4, 1997, required that FPC achieve satisfactory progress on MCAP II before restart of Unit 3. In September 1997, an NRC inspection of MCAP II concluded that the licensee had twelve MCAP II items on which additional progress was needed prior to restart. The results of that inspection were documented in NRC Inspection Report (IR) 50-302/97-13. During this inspection, the inspectors followed up on the status of those twelve items.

b. Observations and Findings

The inspectors reviewed the licensee's MCAP II files and discussed certain items with licensee personnel. The inspectors found that the licensee had made satisfactory progress for restart on eleven of the twelve items. The one remaining item was to provide adequate licensing basis information and training to the plant staff to support operability evaluations, Technical Specifications (TS) interpretations, and 10 CFR 50.59 evaluations. Licensee personnel described plans to accomplish that item prior to plant restart.

c. Conclusions

The inspectors concluded that the licensee's progress to date on MCAP II continued to be satisfactory, and there was one item related to licensing basis information which remained to be accomplished prior to plant restart.

The inspectors had previously assessed the licensee's performance as adequate, relative to MCAP II, in IR 97-13. That assessment was not affected by this followup inspection.

O7.2 Licensee Self-Assessment Activities (71707, 40500)

The inspectors reviewed various licensee self-assessment activities and corrective action processes which included:

- Routine reviews of Nuclear Quality Assessments (NQA) activities and surveillance report findings
- Reviews of precursor cards entered in to the corrective action system

The inspectors observed that NQA activities continued to be appropriately focused on fulfilling audit requirements and using discretionary time to inspect suspected problem areas. Licensee line management continued to utilize NQA to follow up on suspected problems within departments to giving management an independent assessment of performance.

The inspectors continued to review PCs entered in the corrective action system to verify the licensee had addressed problems with screening significance levels. These problems were discussed in previous reports including IR 50-302/97-16. The inspectors have not identified any significant errors in screening PCs since those observations. Minor discrepancies still occurred but were recognized and corrected by licensee management in their daily reviews of all the screening committee decisions. The inspectors did not identify any notable concerns and concluded the licensee is devoting appropriate attention to the problem.

The inspector assessed the licensee's performance, with respect to this restart-related issue, in the five NRC continuing areas of concern:

- Management Oversight - Good
- Engineering Effectiveness - N/A
- Knowledge of the Design Basis - N/A
- Compliance with Regulations - Good
- Operator Performance - N/A

O8 Miscellaneous Operations Issues

O8.1 (Closed) VIO 50-302/97-02-01: Failure to Follow Equipment Control Procedure Requirements (FPC Restart Issue O-13A)

a. Inspection Scope (92901)

This item addressed multiple examples of equipment incorrectly positioned or configured, which revealed inadequate controls to maintain the appropriate status of operational configuration management.

The inspector reviewed the licensee's root cause and corrective actions, the violation response, and recent self-assessments to monitor the effectiveness of corrective actions. The inspector also independently verified a sampling of the corrective actions and assessed their effectiveness.

b. Observations and Findings

The licensee responded to the violation in a letter dated May 23, 1997 (3F0597-29). The licensee agreed with the violation, and stated the reasons to be: poor work practices, insufficient field supervision, and program and procedure deficiencies. Additional causal factors assessed by the licensee included the following: a lack of self checking; at times, a false sense of urgency, and an observed preoccupation by and resultant distractions of personnel; supervision not spending adequate time in the field; weaknesses in compliance procedures (CP)-115, Nuclear Plant Tags and Tagging Orders, and in CP-113A, Work Request Initiation and Work Package Control; some operating procedure deficiencies; plant labeling weaknesses; needed Work Control Center (WCC) facility upgrades and staffing increases; poor verbal and written communications; and, inadequate training of operations personnel.

Licensee corrective actions completed and planned included the following items:

- Returned the equipment that was found out-of-position to its required position as stated in the clearance or procedure in effect

- Developed a configuration control improvement program directed towards operations performance improvements
- Conducted training sessions during a site-wide stand down to address configuration control issues and recent errors
- Conducted classroom and on the job training for operations
- Issued a required reading study book item
- Issued a night order book item addressing supervisory expectations in the field
- Increased supervisory time spent in the field monitoring and coaching
- Evaluated operator administrative duties, and eliminated those which could be performed by clerical assistance or WCC personnel
- Added configuration control to operator continuing training
- Revised Procedures CP-113A and CP-115
- Revised those operating procedures which were identified to be in error
- Performed an assessment of the effectiveness of the corrective actions
- Plans to upgrade plant labeling
- Plans to upgrade the WCC facility (work in progress) and to increase staffing

The inspector verified a sampling of the above stated corrective actions. Observations of the tagout and clearance process were made from the control room, the WCC, and the field. Selected clearances were reviewed and walked down in the field. The inspector noted that the labelling related corrective action has a long lead time, and is not due for completion until December 1998. The inspector discussed configuration issues and the above corrective actions with plant and Operations management, operators, and licensing personnel.

The inspector also reviewed recent line management and independent assessments of the program and related enhancements. Recently performed Nuclear Quality Assurance audits and surveillances concluded that the configuration control program performance was adequate and recent trends were improving. This was verified by the inspector's review of the 1997 Nuclear Quality Assurance assessments.

c. Conclusions

The inspector determined the licensee's corrective actions appeared appropriate to address configuration control problems. Recent trends have been positive; however, continuing monitoring for effectiveness and management oversight is warranted. Based on the above reviews and inspections, and on assessments of recent performance, the violation was closed.

The inspector assessed the licensee's corrective action performance, with respect to this restart-related issue, in the five NRC continuing areas of concern:

- Management Oversight - Good
- Engineering Effectiveness - N/A
- Knowledge of the Design Basis - Adequate
- Compliance with Regulations - Adequate
- Operator Performance - Adequate

O8.2 (Closed) LER 50-302/96-21-00: Delayed Entry Into Technical Specification Required Action Caused by Inadequate Documentation of Out-of Service Equipment Requirements for a Modification

a. Inspection Scope (92901)

This issue involved the licensee's delayed entry into a technical specification required action involving an inoperable reactor protection system (RPS) channel. The inspector reviewed Licensee Event Report (LER) 96-21-00, which was issued in response to the delayed technical specification entry. The inspector reviewed associated documentation, and interviewed licensee personnel to determined the adequacy of the licensee's response to the issue identified in the LER.

b. Observations and Findings

The inspector noted that resolution of this LER was being tracked under licensee Restart Issue O-15. The inspector reviewed the documentation associated with the LER and restart package O-15. Included in the documentation reviewed were the root cause of why the delayed entry into the TS occurred and the corrective actions intended to prevent recurrence. Following a review of associated documentation, the inspector concluded that the licensee had identified the root cause and contributing factors, and that the corrective actions were adequate. The inspector verified that all corrective actions were completed. The corrective actions included a detailed incident investigation per Operations Instruction (OI)-12, operations department study book entry, and procedure revisions to OI-7, Control of Equipment and System, Operating Procedure (OP)-502, Control Rod Drive System and Preventive Maintenance Procedure (PM)-114, Control Rod Drive Mechanisms - Electrical Checks.

Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 97-21, the licensee meets the criteria for enforcement discretion per Section VII.B.2 of the NRC Enforcement Policy as described in NUREG-1600. Consequently this item is closed and is identified as another example cf Non-cited Violation NCV 50-302/97-21-01, Examples of Noncompliances in Design Control, 10 CFR 50.59 Evaluations, Procedure Adequacy/Adherence, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

c. Conclusions

The inspector concluded that the licensee's corrective actions for this LER were satisfactory. This item is closed

The inspector assessed the licensee's performance, relative to the corrective actions for this LER, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - N/A
- Knowledge of the Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - Good

O8.3 Reportability Program

(Closed) EA 97-094 (4 examples: 01013, 01023, 01033, and 01043); Repeat Failure to Make Timely Reports to the NRC

(Closed) VIO 50-302/97-08-01; Inadequate Corrective Action and Procedure for External Reporting Requirements

a. Inspection Scope (92901, 40500)

To close these items, the inspector reviewed the licensee's open item closure packages for Restart Items OP-4 and OP-4A, which included their response to the items and their program for reporting events and conditions to the NRC as required by 10 CFR 50.72 and 50.73. This included review of the licensee's procedures and processes, review of recent reportability determination problems, and discussions with operations personnel. The first violation was unsuccessfully inspected for closure in Inspection Report 50-302/97-08. A significant deficiency was found in the procedure which resulted in the opening of the second violation.

b. Observations and Findings

The inspector reviewed the licensee's current procedures for implementing the reporting requirements of 10 CFR 50.72 and 50.73: CP-151, External Reporting Requirements, Rev. 3, dated October 6, 1997 and Nuclear Operations Directive (NOD)-3, Reporting Requirements Program.

Rev. 13, dated August 19, 1995. The inspector noted that the CP-151 procedure definition of discovery time for determining the time limits for reporting events had been corrected. The term "Discovery Time" was eliminated and replaced by "Time Limits for Reporting" which clearly delineated the regulatory requirements. The inspectors interviewed the Nuclear Shift Managers (NSM) responsible for implementing the procedure and determined they understood the revised definition. The inspector also determined CP-151 was generally clearly written and was an acceptable procedure to support accurate and timely reportability determinations. Procedure NOD-3 was primarily a matrix summarizing numerous routinely written reporting regulatory requirements and responsible licensee personnel for each report. The inspector did not review the adequacy of the guidance for other reports, only 10 CFR 50.72 and 50.73. The inspector noted one administrative discrepancy in that LERs were required to be issued by the Licensing Manager and approved by the Vice President for Nuclear Production. The licensee's current practice is to have accountable department managers issue the LERs. Based on the extended time since the last revision to NOD-3 and numerous recent licensee organizational changes, the inspector considered it appropriate to review and revise NOD-3. The licensee agreed and was evaluating needed revisions.

Numerous reportability determinations have been performed since the identification of the second violation. These have generally been consistently timely and technically valid. The inspector has observed that licensee personnel almost always document concerns via the PC process and forward them to the NSM for reportability review in the same shift. However, the inspector's review revealed several recent deficiencies:

- IR 97-08 noted that CP-151 contained a new requirement for tracking the outstanding reportability evaluations by the NSM. The process prescribed by CP-151 required PCs classified as potentially reportable items to be tracked, and a final independent reportability determination to be made by the NSM after receipt of a written technical evaluation from an assigned Reportability Review Owner. If determined to be reportable, this removed the PC from the potentially reportable tracking list and placed it on the pending LER list. PC 97-2055 was originally screened as a potentially reportable item per 10 CFR 50.73 by the NSM on May 27, 1997. After receipt of the written technical evaluation, the NSM upgraded the PC to reportable on June 10, 1997, with May 27, 1997 as the discovery date. However, the inspector observed that a LER was not issued 30 days from the discovery date of May 27, 1997, as required per 10 CFR 50.73. Further investigation revealed that shortly after June 10, 1997 the Licensing and Engineering groups had determined that PC 97-2055 was not reportable because the issue was covered as part of the extent of condition for a previously reported LER. Although this determination was correct, this knowledge caused Licensing to remove it from the pending LER list without the PC being rescreened by the NSM. The last reportability determination posted against the PC and signed by the NSM was that it was reportable. The licensee generated PC 97-7565 when the inspector identified this deficiency, and processed a reportability evaluation through the NSM on November 6, 1997 to recategorize PC 97-2055 as not reportable. The licensee's follow-up to PC 97-7565 determined that the guidance in CP-151 was adequate but that this deficiency was due to a personnel error. The inspector considered this an accurate assessment.

- PC 97-2485 was first screened by the NSM on April 6, 1997 as not reportable. An Engineering self-assessment caused the PC to be re-evaluated and on October 8, 1997, it was determined to be reportable. However, contrary to CP-151 requirements, the NSM recorded the "discovery date" as October 8, 1997, and not April 6, 1997. Although the correct discovery date was used in the LER, PC 97-7751 was issued to correct the NSM misconception.

- PC 97-4530 was determined to be reportable per 10 CFR 50.72 and 50.73 on July 7, 1997. A four hour phone report per 50.72 was initiated. However, the PC was not tracked for reportability and the LER was not issued within 30 days as required. On August 29, 1997 the licensee became aware of the oversight and initiated actions to process the report. However, LER 97-27 was issued October 3, 1997, which again didn't make the 30 day limit from August 29, 1997. The licensee issued PC 97-6256 to document and correct the failure. Their apparent cause determination concluded that their tracking methods were weak and periodic cross-checks with the NSM logs were non-existent. Corrective actions included improvements to their tracking process and a weekly review by Licensing of all items in various stages of reportability determinations, which were verified against NSM logs to ensure all items were contained in the database. The inspector verified these actions and has not noted any similar problems since these actions were implemented.

- PC 97-6224 was initiated and screened for reportability on October 5, 1997. Although subsequently determined to be not reportable, the licensee's Engineering Manager recognized that the technical concern of this PC had been identified approximately a month earlier, but a PC had not been initiated. Therefore the reportability evaluation was late due to the delay in issuance of the PC. The licensee has identified several examples of untimely PC generation and has vigorously attempted to correct the problem and the resultant delay in reportability evaluations. PC 97-6950 was generated by the licensee to capture these efforts.

- For a potentially reportable PC that has been evaluated as not reportable by the Reportability Review Owner, CP-151, section

4.4.5 required the NSM to review the recommendation, make an independent reportability determination, and re-perform sections 4.2 and 4.3, which would entail completing a new Enclosure 3, Reportability Evaluation Worksheet. However, the inspector noted that several recent PCs that were determined not reportable did not have new Enclosure 3 forms completed. The NSM indicated his concurrence with the reportability recommendation by signing the memo received from the Reportability Review Owner. The inspector also observed this practice on a "Reportable" PC that was subsequently determined to be "Not Reportable." When questioned by the inspector, a NSM indicated that signing the memo was allowed by the procedure and would eliminate confusion that could result from having two Enclosure 3 forms in the PC file. The inspector did not agree with this reasoning because the last Enclosure 3 in the file would be the original "Potentially Reportable" or "Reportable" decision, and an auditor would have to determine the non-reportable decision from reviewing the details of the Reportability Review Owner memo. Also, when the NSM signed the memo this diluted the apparent independence of the NSM's reportability review. Per CP-151, the memo was supposed to contain a recommendation, but the NSM was responsible for making an independent determination on reportability. His signature, indicating concurrence with the memo, gave the appearance that he was part of the recommendation process and not independent. Lastly, CP-151 did not allow the signing of the memo as the NSM indicated was allowed. It required another Enc. 3 form to be completed. The inspector discussed these observations with the Licensing department, who was pursuing corrective action to ensure the NSMs complied with the CP-151 requirements. The inspector considered this problem to be administrative with no safety significance. However, it did indicate a potential disregard for strict procedural compliance among members of shift management.

The inspector reviewed the licensee's root cause and corrective actions for PCs 97-0724 and 0841, which were written for the examples cited in EA 97-094. The inspector verified that the corrective actions for these PCs were appropriate and completed. The licensee's package also contained a common cause analysis under PC 97-2089 for reportability errors. The licensee determined that the primary common root cause for their numerous problems was the lack of ownership of the reportability process. The inspector's review of their corrective actions revealed that the Operations department has been clearly delineated as the process owner, and that Licensing was established as a consistent source of guidance. The inspector considered the licensee's conclusions and actions appropriate and the common cause analysis thorough.

The inspector verified the licensee's closure actions for VIO 97-08-01 that were done under PC 97-4918. This review entailed verifying CP-151 was corrected as discussed earlier. No discrepancies were noted with the licensee's actions for this item. The inspector also reviewed the licensee's overall reportability closure package, OP-4, which was initiated in 1996 before the open items were identified. It adequately addressed the overall programmatic issues the licensee had with reportability.

The overall licensee reportability process has significantly improved since the current plant shutdown was initiated in September 1996. A team was established by the licensee in November 1996 to develop an integrated plant approach to reportability. Improvements in the corrective action process have ensured that it is the source of virtually all reportable items, have resulted in better information provided to the NSM making the reportability determination, and have improved the timeliness of prompt concern identification and classification. The inspectors have also observed that in the spring of 1997, a dedicated, point-of-contact person was designated in the Licensing organization. An individual with detailed knowledge of reportability requirements was not previously available to the NSMs for reference and guidance. The result has been an improvement in reportability determination consistency and tracking. Although the process improvements have been significant, deficiencies with reportability still occur due to failures to follow the process. The inspector concluded that these were primarily administrative errors of minimal safety significance, but the licensee was still challenged to ensure their process was correctly implemented.

c. Conclusions

The inspector determined the licensee's actions were good improvements to their reportability process and addressed the programmatic and specific causes of the open items. Consequently both of the violations are closed. However, the inspector and licensee have identified several examples of personnel errors that can potentially circumvent the improvements in the process. The inspector concluded the licensee's reportability program was adequate and acceptable for restart and has been functioning well during this report period. However, licensee management attention in this area needs to continue to ensure procedure compliance and that personnel errors are minimized.

The inspectors assessed the licensee's performance, relative to the Reportability Program, in the five areas of continuing NRC concern:

- Management Oversight - Adequate
- Engineering Effectiveness - N/A
- Knowledge of the Design Basis - N/A
- Compliance with Regulations - Adequate
- Operator Performance - Adequate

II. Maintenance

M1 Conduct of Maintenance

M1.1 Incorrect Calibration Performance Causes Spray Valve Opening

a. Inspection Scope (62707)

On November 5, 1997, a licensee instrument technician did not perform a procedural step which caused an inadvertent opening of the pressurizer spray valve. The inspector reviewed the licensee's response to the problem and their corrective action plan.

b. Observations and Findings

Surveillance Procedure 112, Calibration of the Reactor Protection System, Rev. 57, was being performed on November 5, 1997 to calibrate RCS pressure transmitters (PT). Steps 4.3.2.6 and 4.3.2.7 required the instrument technician to have Operations bypass the Smart Analog Signal Select system (SASS) channel being tested and select the channel not under test for control. The steps were allowed to be marked not applicable (N/A) only if certain instruments were being tested. The technician was not testing those annotated instruments but he inappropriately marked these steps N/A, continued on with the procedure, and did not consult with the control room operator. This resulted in an inadvertent high pressure signal being sent to the pressurizer spray valve, causing it to open. The impact of this on the RCS was inconsequential because the pressurizer spray block valve was closed so the nitrogen over pressure blanket on the pressurizer was not affected. The licensee initiated PC 97-7638 for corrective action and assigned it a grade of "B" which requires a formal root cause be completed. The licensee assigned the higher grade because they recognized the potential implications of the procedure error if plant conditions had been different. The root cause was not completed at the and of the inspection period, but the inspector discussed the preliminary cause and planned corrective actions with the instrument technician shop supervisor. The licensee determined the primary causes were personnel performance and a lack of attention to the details of the procedure. They were developing appropriate corrective actions. They also noted some ambiguity in the wording of the skipped steps and were evaluating enhanced wording.

c. Conclusions

The inspector concluded the instrument technician displayed a poor level of skepticism and self-checking when electing not to perform steps in a procedure. This resulted in an inadvertent actuation of plant equipment that fortuitously had minimal consequences. However, the licensee recognized the implications of the problem and took appropriate corrective actions.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Maintenance Activities on Protected Engineered Safeguards (ES) Trains

a. Inspection Scope (62707)

The inspectors investigated the licensee work controls for performing work on the protected engineered safeguard trains. The circumstances involved in the issuance of PC 97-7622, involving unauthorized work performed on a protected ES train observed by a licensee auditor, were reviewed by the inspectors.

b. Observations and Findings

On November 6, 1997, PC 97-7622 was issued to document that work authorized on the protected A train of 480 V ES motor control center (MCC) cubicles had been expanded to another cubicle without approval from the Shift Supervisor on Duty (SSOD). Craft personnel were performing Work Request (WR) 348172 to implement Modification Approval Record (MAR) 97-06-13-01, to replace a nine point terminal block with two four point blocks. The personnel had notified the Engineering department that there was insufficient space in the specified cubicle for the two blocks. Engineering provided a field change notice to the modification specifying that the terminal blocks be mounted in a different, spare cubicle. The craft supervision opened the spare cubicle and were scoping out the work when a technician expressed concern that the work was being performed on the protected train. At the time of the original work, the A train was considered the protected and operable ES equipment train. The SSOD had approved work in the original cubicle but was not aware of the change notice, which expanded the work outside the bounds of his original approval.

The inspector reviewed the licensee's administrative controls for work control and modifications. Even though the licensee designated a train of ES systems as protected, there was neither a procedural definition of protected nor requirements for controlling activities on these systems. This weakness created the potential for a threat to the systems that the licensee was maintaining operable for Technical Specification requirements.

The inspector reviewed licensee Procedure NEP-251, Preparation, Review, and Approval of Field Change Notices (FCN). This procedure details the necessary approvals for development of a FCN, including requiring Director of Nuclear Plant Operations approval for implementation. The SSOD was not required to be notified prior to a FCN being issued to the field for installation. This weakness allowed the SSOD to be bypassed and unaware of changes to work scope that had been previously approved.

The licensee took immediate corrective actions by suspending the ongoing work under the MAR. A night order was issued to the SSOD requiring that, before a SSOD allows work in the protected train, an assessment of the possible consequences of this work must be performed and approved by the SSOD. The information for the evaluation must be supplied to the SSOD, in writing, by the individual wanting to do the work. This information was to include, as a minimum, scope of work, reasons why work must be done during the requested window, contingency plans regarding work, duration of work activity, scaffolding plans, plans for any required barriers between trains, pre-job briefing requirements, methods of communication, a single point of accountability at the job site, and the project manager's name. The forms to perform this work have been approved since the issuance of the night order and were reviewed by the inspector. Some of the information was observed to be ambiguous on the form, but the SSOD was aware of the conditions existing and was able to answer all questions from the inspector.

c. Conclusions

The control of work in the protected trains of engineered safeguards systems was weak, because it lacked adequate procedural guidance and there was not a clear definition as to what constituted a protected train. A weakness was also noted in changes to modification scope being issued directly to the field, with review and approval of scope changes not being communicated to the operations shift supervisor on duty.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) LER 97-002-01: Out of Calibration Fuel Pool Water Level Transmitters

(Closed) VIO 97-01-04: Failure to Perform Technical Specification Surveillance for Spent Fuel Level

a. Inspection Scope (92903)

This violation and LER involved the licensee's failure to calibrate Spent Fuel Pool Level Transmitters within their required calibration intervals. The inspector reviewed documentation which was generated as a result of the subject LER and violation, and interviewed licensee personnel to assess the adequacy of the licensee's corrective actions.

b. Observations and Findings

The inspector noted that the resolution of the violation and the LER was being tracked under licensee restart Item M-9. The inspector verified that the licensee had completed the corrective actions which were identified as a result of the violation and the LER. The licensee identified the corrective actions in a letter dated June 16, 1997, in the LER, and in restart issue package M-9. The corrective actions included:

- Develop a root cause determination
- Develop formal expectation/duties for daily scheduled surveillances
- Revise appropriate procedures
- Develop new procedures
- Perform an extent of conduction inspection
- Provide training to appropriate personnel

Following the review of documentation associated with the violation and the LER, and interviews of licensee personnel, the inspector concluded that the licensee had completed the corrective actions identified. The effectiveness of the corrective actions implemented by the licensee will be evaluated during future routine inspections.

c. Conclusions

The inspector concluded that the licensee's corrective actions for the violation and the LER were satisfactory. These items are closed.

The inspector assessed the licensee's performance, relative to corrective actions for the violation and the LER, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of the Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - Good

III. Engineering

E1 Conduct of Engineering

E1.1 Design Control Process

a. Inspection Scope (37550)

The inspectors reviewed two NRC restart items that were identified as engineering programs, GL 96-01, Testing of Safety Related Logic Circuits and DC Failure Modes and Effects of Loss of DC Power (FMEA) to verify their completion. These two programs were inspected to verify they were technically adequate and were implemented and completed in accordance with the licensee's commitments and NRC regulations.

b. Observations and Findings

b.1 (Closed) GL 96-01, Testing of Safety-Related Logic Circuits (MPA #L601)
IR 50-302/97-11 (May 5-9, 1997) referenced IR 50-302/97-07 and identified that the licensee had completed all the requirements for GL 96-01 except for the following items: 1) reactor protection system validation; 2) closure of eight open PCs; 3) contractor's final submittal; and 4) final review, approval and closure by the licensee of all GL 96-01 documents. The final documents for closure were identified as R01 through R01G.

The inspectors verified that all open items and documents were satisfactorily completed by the licensee. These items included the revision of 11 surveillance procedures and implementing corrective action for the eight open PCs. The licensee opened LER 97-003-001 through 005, regarding GL 96-01 testing deficiencies and closure. LER 97-003-005 encompassed and superseded LER 96-025-00 and LER 96-011-00, which also were concerned with logic testing deficiencies and GL 96-01.

The licensee sent a Conformation of Completion (TAC 94668) letter to the NRC for GL 96-01, Testing of Safety related Logic Circuits, dated September 5, 1997. The NRC replied by letter dated September 22, 1997 that the licensee had provided the required submittals and responses for their commitments for GL 96-01 and therefore TAC 94668 was closed. The inspectors concluded the licensee had implemented a superior program to meet the requirements in GL 96-01.

b.2 (Closed) Failure Modes and Effects Analysis (FMEA) for Loss of DC Power (CR3 D.I.7)
IR 50-302/97-11 (May 5-9, 1997) referenced IR 50-302/97-07 and identified that the licensee had completed all their commitments except for the following items: 1) address and closeout the 12 open PCs; 2) complete final review of contractors work; and 3) final review, approval, and closure of all FMEA documentation. The inspectors verified that all open items and documents were satisfactorily addressed and completed by the licensee.

The licensee Conformation of Completion letter dated September 24, 1997, was in response to the requirement in NRC's Confirmatory Action Letter dated March 4, 1997. The licensee's letter stated that the FMEA program was completed except for one recently identified item. A problem with a potential 3A battery failure was identified and was being tracked as a new licensee restart item D-07A. This potential 3A battery failure condition was described in PC 97-4354 and LER 97-21 as "Loss of Class 1E Battery A With LOOP/LOCA Will Result in Failure of EDG-3A to Load ES AC Buses." The licensee opening of this new restart item for the potential failure of the 3A battery was being adequately addressed, and closure of the FMEA program was appropriate. The inspectors concluded the licensee had implemented a superior FMEA program to meet their requirements, commitments and NRC regulations.

c. Conclusions

The inspectors concluded that the licensee had implemented and completed superior programs for GL 96-01 and DC FMEA. Both programs reviewed were technically adequate and were implemented in accordance with licensee requirements, commitments and NRC regulations.

The inspectors assessed the licensee's performance, relative to the design control process, in the five areas of continuing NRC concern:

- Management Oversight - Superior
- Engineering Effectiveness - Superior
- Knowledge of the Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - N/A

E1.2 Emergency Diesel Generator Loading Calculations

a. Inspection Scope (92903)

The inspector reviewed the licensee's emergency diesel generator loading calculations. The requirement to have emergency diesel generators and requirements for the performance of those generators are contained in 10 CFR 50, Appendix A, Criterion 17 - Electric Power Systems. Criterion 17 states that an onsite electric power system shall be provided to permit functioning of systems important to safety. The safety function of the onsite power system (emergency diesel generators) shall be to provide sufficient capacity and capability to assure that the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. The onsite electric power supplies shall have sufficient independence and redundancy to perform their safety functions assuming a single failure.

b. Observations and Findings

The licensee's emergency diesel generator loading analysis was contained in Calculation E-91-0026, EGDG-1A Scenario Based Loading, Voltage Dip, Frequency Dip and Transient Motor Starting Analysis, Revision 3, dated October 17, 1997. Calculation E-91-0027 covered the B train emergency diesel generator. A major supporting calculation was Calculation M-96-0069, ES Pump Maximum Flow for EDG Loading, Revision 0, dated September 15, 1997. As indicated by the title, these calculations determined loading for nine accidents together with the single failure of either a diesel generator, the turbine driven emergency feedwater pump or a battery (control power). The inspector found that the calculations contained the input data to demonstrate that the requirements stated in the scope section above were met. The inspector observed that suitable design control measures were employed in development of the calculation.

In the short term, i.e., less than one hour following an accident, the worst case scenario was a steam line break inside containment with failure of the B train diesel generator. The calculated load included all loads powered by the diesel generator in their respective maximum flow condition, and the total load was within the diesel generator 30-minute and 200-hour ratings with about 6 percent margin. In the long term, i.e., more than one hour following an accident, the worst case was a small break LOCA with failure of the A train diesel generator and with recirculation in piggy-back mode. This scenario involved operator action to remove and add loads. The inspector observed that the new operating procedures contained simple, clear instructions to guide the operator in removing and adding loads in the context of diesel generator loading. The calculated worst case long term load was within the 200-hour rating.

The ability of the diesel generator to accept the programmed load sequence was demonstrated with a dynamic type computer program. The results of this program showed that peak transient loading was kept below 3910 kW, the maximum diesel generator power output capability, which would minimize frequency drop-off associated with suddenly applied relatively large loads. The results of the dynamic program showed that the voltage and frequency excursions and recovery times remained within the limits recommended in Regulatory Guide 1.9, Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants, Revision 3, dated 1993. As an independent check of these computer results, the inspector reviewed results from the integrated diesel generator sequencing test performed under Surveillance Procedure SP-417 on April 25, 1996. The voltage and frequency traces recorded during this test were consistent with the results of the E-91-0026 calculation.

The inspector observed that the licensee calculated two sets of load values within the diesel generator loading calculation; a value based entirely on manufacturer, originally supplied pump performance curves (calculated value) and a value determined by recent onsite measurements of electric power and system parameters made by the licensee for each of the major loads. The inspector observed that for some loads the measured value was higher than the calculated value, and for some loads the measured value was less than the calculated value. The two values for each load matched within an acceptable deviation for comparison of test and calculated values. The licensee determined two total loads: one using all calculated values and one using all measured values. The higher of these two total loads (usually the calculated value) was taken as the calculation result and compared to the diesel generator ratings. Having two values for each load raised the question of which was the correct load to use in the calculation. The inspector calculated a total load using the higher of calculated and measured for each load, and found that the total load calculated in this manner was about 50 kW higher than the licensee's calculation result. However, the inspector observed that even with this additional 50 kW the 30-minute and 200-hour ratings were not exceeded.

The diesel generators had been modified recently to upgrade the 200-hour and 2000-hour ratings. The results of the calculations were compared to the new ratings. The Updated Final Safety Analysis Report (UFSAR) would have to be revised to reflect new ratings and higher load values. The licensee provided copies of the revised UFSAR pages which would be submitted to the NRC.

c. Conclusions

The inspector concluded that the revised emergency diesel generator loading calculations demonstrated that the generators have the capacity and capability to accept the design basis loads as required by 10 CFR 50, Appendix A, Criterion 17.

The inspector assessed the licensee's performance relative to the revision to the diesel generator loading calculations in the five areas of continuing NRC concern.

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of the Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - N/A

E1.3 Non-use of T-MARs on Secondary Systems (37551)

a. Inspection Scope

In July of 1997, during repairs to leaking pipes in the site drain (SD) system, the inspector identified a concern with the potential for the licensee to make modifications to a system and restore it to service outside of the normal modification approval process. Specifically, repairs to one of the pipes were delayed and use of the pipe was urgently needed to support a discharge of the Turbine Building sump drain tank, SDT-1, which was full and causing the sump to fill to capacity.

b. Observations and Findings

Operations and maintenance shift management elected to place a temporary patch on the pipe by initiating but not fully documenting a WR, and then placed the SD system in service to discharge SDT-1. They based their decision on a statement in CP-113A, Work Request Initiation and Work Package Control, that allowed the use of a maintenance activity in an emergency without having a documented WR. However, they were not specifically allowed to restore a system to service per this process, were supposed to document the change in the Temporary Alteration Log which was not done, and were required to document the work afterwards which should receive the same degree of review as a fully planned WR. The inspector was concerned that this patch constituted a temporary modification to the SD system which was not processed under the Temporary Modification Approval Record (TMAR) process. Licensee management had identified this problem and also had the same concern. They initiated an appropriate investigation and corrective action. The inspector reviewed the results of their root cause determination in PC 97-5076. The licensee identified another patch on the same line that had been installed in 1993 under similar circumstances. The PC discussed that it had been added by improperly altering the scope of a WR and had been left in place without any subsequent documentation. As an immediate corrective action, the licensee performed an engineering evaluation of both patches and a Commercial Grade Work Request was developed to document the adequacy of both patches. They determined that their work process procedural guidance was weak in involving Engineering in potential modifications to plant configuration. They identified appropriate corrective actions to correct the guidance.

c. Conclusions

The inspector concluded that the licensee had performed a very good investigation and root cause determination, although the documentation was difficult to follow and the corrective actions didn't clearly match the identified causes. The inspector did not identify any safety concerns with the use of the temporary patches and determined the original concerns were adequately resolved by the corrective actions of the PC. The inspector concluded the licensee's threshold for identifying temporary modifications was appropriate.

E1.4 Loose Part on A OTSG Upper Tubesheet (37551)

In IR 50-302/97-11, the inspector discussed the licensee's discovery of a loose part on the A OTSG tube sheet. The damage from this part significantly expanded the scope of repair efforts for damaged tube ends and delayed the licensee's schedule. The part was determined to be half of a 3/4 inch hex nut and caused the licensee to have to repair over 10,500 of the 15,531 tube ends on the A OTSG in order to complete eddy current inspections. The licensee initiated a root cause investigation to determine the source of the part, address deficiencies with the Loose Parts Monitoring System (LPMS) that failed to identify the loose part in the OTSG, and assess the impact of other potential loose parts in the RCS. The licensee's effort for this investigation was disjointed because it was tracked by several precursor cards. In September 1997, the inspector reviewed the completion of PC 97-4269 which addressed the LPMS problems. The inspector noted that the licensee determined the LPMS detector for the A OTSG was inoperable following their startup from the last refueling outage, even though it had been tested acceptable just prior to startup. This contributed to the lack of knowledge of the loose part until the OTSG was opened for inspection. The licensee proposed appropriate corrective actions for the LPMS, but the PC left numerous questions unanswered such as the source of the hex nut, the potential for the other half of the hex nut to be in the RCS, and the potential for other known loose parts in the RCS to cause further damage to the repaired tube ends. Licensee Engineering management was unable to resolve these questions when asked by the inspector, although it was later determined a separate investigation under PC 97-4440 would resolve the various issues. The inspector deferred further review of this problem until that PC was completed.

In November 1997, PC 97-4440 was completed. It encompassed PC 97-4269 and PC 97-5041 on the OTSG tube sheet damage, to be a final integrated assessment of the loose part issue. The licensee's final determination concluded significant deficiencies in the foreign material exclusion (FME) program over several years had contributed to the loose part in the RCS. The inspector verified a significant upgrade was in process for the licensee's FME program in response to this and other recently identified problems. The licensee determined that the loose part most likely originated from a refueling bridge crane and adequately resolved the source and status of other loose parts. One small loose part was known to remain in the bottom of the reactor vessel. This was analyzed as an acceptable condition in 1994 due to the small size of the part and remote chance of it relocating. The licensee reviewed that justification to ensure it remained valid and bounded the current situation. The inspector did not identify any further concerns with the current status of loose parts in the RCS.

c. Conclusions

Although the final assessment was completed well after the part was identified in June 1997, the inspector concluded the licensee's final assessment was thorough and adequately resolved any concerns with the potential for loose parts damage in the RCS.

E2 Engineering Support of Facilities and Equipment

E2.1 Emergency Diesel Generator 1B MAR Functional Test

a. Inspection Scope (61726, 62707, 92903)

The inspectors observed activities associated with the MAR functional test (FT) of the radiator replacement and power upgrade of emergency diesel generator (EGDG) 1B. The modifications were performed under MAR 97-05-15-01 and 97-05-15-02, radiator replacement, MAR 97-05-15-05 for the radiator fan drive upgrade, MAR 96-10-[illegible]5-01 for the generator 150 kw upgrade, and MAR 97-04-03-02 for the EDG building ventilation system modification. The inspectors observed the performance of several of the prejob briefings, test runs of the EGDG-1B, and troubleshooting activities associated with the MAR FT. A review of the MAR FT procedure and associated documents was conducted by the inspectors.

b. Observations and Findings

The dates, times, and other information for each of the diesel starts are described in the attachment at the end of the inspection report.

Prior to the beginning of the MAR functional test, the inspectors reviewed the test procedure, MAR 97-05-15-01 TP 2. Several areas of concern were noted. Section 5.0, Limits and Precautions, Step 5.14 stated that generator stator temperature and not exceed 150°C. The step states that if the generator high temperature alarm actuated at 135°C, the generator stator temperatures should be closely monitored to ensure 150°C was not exceeded on the highest reading Resistor Temperature Detector (RTD). The inspector reviewed licensee Procedure AR-902, DGB Annunciator Response, which stated that for a valid alarm, the operator shall check for adequate generator airflow, reduce load on the diesel generator, and inform the electrical supervision of protective relay actuation. Discussions with various operations personnel determined that if the alarm were received, the personnel would follow the more restrictive annunciator response procedure requirements. The developers of the MAR FT failed to realize that the operators would follow the annunciator response procedure and make allowances or provide justifications in the MAR FT for the operators deviating from the AR requirements.

The inspector performed a field walkdown of the EDG. The meter used to monitor generator stator temperature has a maximum reading of 140 degrees C. The licensee procedure allowance of 150 degrees C could not be monitored with installed instrumentation, and the procedure developers failed to validate this change. The licensee changed the limit to 135 degrees C, after notification by the inspector of the discrepancy.

The inspector also identified that referenced temperature relay for generator stator temperature was a model IRT51A. The vendor information in the controlled copy of the EDG manual states that this relay was rated for 80-120°C. The setpoint on this relay was 135°C. PM-102, Calibration of Protective Electrical Relays, stated that the range for this relay was 100-160°C. The licensee located a letter from the vendor, dated June 26, 1979, which stated that the original relay could not support raising the setpoint to 135°C, so an extended range model, with revised operator manual, was being supplied. The licensee had not updated the diesel vendor manual to reflect this change.

Further review of the MAR FT procedure development documentation, CP-134, Preparation and Approval of MAR Functional Test Procedures, Enclosure 4 disclosed that the test was to be performed in conjunction with Attachment A, SP-354B, Monthly Functional Test of the Emergency

Diesel Generator EGDG-1B. The safety assessment performed for the MAR FT procedure stated that the operation of the EGDG for this test would be the same as that performed for normal periodic surveillances and functional verification, with the additional load performance verification for the modifications. The inspector reviewed Attachment A to the procedure and determined that it was not the approved revision of SP-354B but was a proposed revision that had not been reviewed and approved by the licensee. The safety analysis did not address the differences in the acceptance criteria contained in the attachment from the approved revision of SP-354B. The licensee's screening evaluation concluded that an unreviewed safety question determination and Plant Review Committee review were not needed. The inspectors reviewed the safety evaluation and determined that the safety analysis for the EGDG-1B procedure was, with minor changes, identical to the safety analysis for the previously performed EGDG-1A MAR FT procedure. However, the A diesel procedure used the approved revision of SP-354A as Attachment A. The B diesel procedure used an unapproved revision to SP-354B, but this was not addressed in the safety analysis.

The Regulation, 10 CFR 50.59, state that a licensee may make changes in procedures as described in the safety analysis report or may conduct tests not described in the safety analysis report, without prior NRC approval, unless the proposed change or test involves a change in the technical specifications incorporated in the license or an unreviewed safety question. The licensee must maintain records of changes to procedures or of tests conducted, including a written safety evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question. The licensee's failure to recognize that the incorporation of new acceptance criteria into the attachment to the MAR FT procedure in the safety analysis resulted in a failure to assess that the changes did not constitute an unreviewed safety question. This failure to conduct an adequate unreviewed safety question determination was identified as a violation, VIO 50-302/97-17-01, Failure to Conduct an Adequate Unreviewed Safety Question Evaluation for a Modification Functional Test.

When the inspector identified that the licensee had incorporated an unreviewed revision to SP-354B into the MAR FT procedure, several licensee representatives were notified. However, the licensee failed to take prompt corrective actions and the test was begun the following day, without changes to the safety evaluation or procedure. At that time, the inspector notified Operations management of the situation, who stopped the performance of the procedure until the appropriate changes had been made.

The inspectors witnessed portions of a number of the diesel starts and runs. The initial start of EGDG-1B occurred on November 9, 1997, using Maintenance Procedure (MP)-499, Emergency Diesel Generator Engine Inspection/Maintenance. This was a slow start with an unloaded run to allow for air flow measurements and visual inspection for maintenance check-out, including an examination for coolant and oil leaks. A review of the Test Log for the MAR FT procedure revealed that at 1:15 am on November 9, 1997, Operations was nearly ready to start the 1B diesel for the MP-499 run. The next log entry was at 3:55 am, which stated that while attempting to complete the MP-499 unloaded run, the diesel (EGDG-1B) developed a large jacket coolant leak and testing was stopped. The inspector reviewed the reactor operator log and determined that the diesel had been started and stopped three times during that time period: the first three starts and stops due to fan drive clutch slippage and on the fourth run the diesel was stopped for the jacket coolant leak. The test log did not reflect start or stop times, or that the diesel was started and stopped four times for the initial MP-499 runs. Overspeed trip testing was performed during a test run on November 10, 1997. The test log states at 4:30 am that the engine was warm enough to start the MP-499 run. At 6:30 am, the test log states that the unloaded run was completed satisfactorily. The Reactor Operator (RO) log states that the diesel was started at 5:35 am and was shutdown at 6:06 am on overspeed trip testing. The test log did not state the start or stop times of the November 10, 1997 run, nor did it state that overspeed trip testing was completed as part of the run. Licensee procedure CP-134A, Performance of MAR Functional Test Procedures, Section 4.1.5.1, Test Log, provided detailed instructions as to the type of entries to be entered into the test log. However, the detail provided did not include specific directions as to entering such data as diesel start and stop times during this MAR FT procedure. Even though not specifically called for, the procedure stated that the test log should contain, but was not limited to, the information contained in section 4.1.5.1. The test log for the EGDG-1B MAR FT exhibited weaknesses, resulting in a failure to record accurately actions taken during performance of the test.

At 5:04 pm on November 10, 1997, the B diesel was started for a test run. At 5:43 pm, the diesel was shut down for high vibrations. A troubleshooting run was conducted November 12, 1997 in an attempt to determine the cause of the vibration. At the time, it was identified that the new clutch pads on the fan drive were not balanced. Engineering directed maintenance personnel to balance the pads without providing the technicians with written procedure or work instructions. The technician assumed that in order to accomplish the work, he should grind all of the pads to the size of the smallest. When this work had been accomplished, a maintenance supervisor discovered what was taking place. The work was stopped and installation of the pads was not allowed. The original clutch pads, removed prior to the modifications, were reinstalled. The licensee issued PC 97-7600 to document the event, and an investigation was initiated to determine the cause and take additional corrective actions.

The licensee's investigation determined that the engineering personnel inappropriately provided verbal instructions to the craft personnel. The craft personnel did not follow the verbal instructions and did not question the lack of written, approved work instructions. Corrective actions have been developed but not yet implemented to addresses these areas.

10 CFR 50, Appendix B, Criterion V states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances, and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to these requirements, the Engineering personnel involved in the diesel engine troubleshooting verbally directed the mechanical maintenance personnel to perform maintenance on the clutch pads and the work was accomplished without any written, reviewed, and approved instructions or procedures. However, consistent with Section VII.B.1 of the NRC Enforcement Policy, this licensee identified and corrected violation treated as a Non-Cited Violation. This issue was identified as NCV 50-302/97-17-02, Maintenance Performed on Safety Related Components Without Approved Procedures or Work Instructions.

The inspectors observed completion of a selection of the testing performed as part of the MAR FT. The final loaded test run was concluded at 6:40 am on November 24, 1997. Inspections were performed on the diesel during the next several days and an unloaded run was performed on November 27, 1997 as part of PM-123, Periodic Electrical Checks of Emergency Diesel Generators. On November 29, 1997, SP-354B was performed and EGDG-1B was declared operable for mode 5 conditions.

c. Conclusions

The inspectors concluded that the development of the MAR functional test procedure was weak, including inadequate safety analysis and the failure to recognize conflicts between the existing annunciator response procedure and the functional test procedure. The Operations staff, however, were consistent in stating that the annunciator response procedure, being more conservative, would be followed, if the alarm was received. The implementation of the test log was weak, omitting detail necessary to reconstruct the sequence of events during the test.

The inspectors were concerned with the actions taken by the Engineering and Maintenance personnel with the fan clutch pads. However, Maintenance supervision appropriately identified the problem and prevented the installation of the clutch pads.

The inspectors reviewed the completed tests and determined that the acceptance criteria and commitments made to the NRC for EGDG-1B testing were successfully completed.

The inspector assessed the licensee's performance, relative to this violation, in the five areas of continuing NRC concern:

. Management Oversight - Inadequate
. Engineering Effectiveness - Inadequate
. Knowledge of Design Basis - Inadequate
. Compliance with Regulation - Inadequate
. Operator Performance - N/A

E8 Miscellaneous Engineering Issues

E8.1 (Closed) VIO 50-302/96-08-01: Failure to Take Timely Corrective Action to Address Issues and Actions For Makeup System Audit Findings and Excessive Vibration on a Spent Fuel Pool (SFP) Pump Fan Motor (FPC Restart Issue OP-24)

a. Inspection Scope (92903)

This item addressed two examples where the licensee had failed to take timely corrective action for self-initiated make-up system audit findings and excessive vibration on a SFP pump fan motor. These degraded equipment conditions were self-identified; however, the licensee had not pursued prompt action to correct the deficient conditions.

b. Observations and Findings

The SFP pump fan (AHF-8A) was operated with higher than normal and increasing vibration levels. The apparent cause of the vibration was a lack of stiffness in the fan housing and a mismatch between the fan drive belt and the drive sheave. A licensee make-up system audit had identified discrepancies during piping walk downs, including drawing errors.

The licensee responded to and accepted the violation in a letter dated October 14, 1996 (Letter 3F1096-07). The licensee concluded that the following causal factors resulted in the violation: personnel error on the part of a system engineer, a lack of management oversight and accountability, weaknesses in the deficiency reporting and tracking systems, and a lack of engineering sensitivity.

Corrective actions included the following:

- Audited the deficiency tracking system and found no other issues
- Reviewed the extent of the condition for vibration issues, and found two other examples. This included the decay heat closed cycle cooling pump fans (AHF-15A/B) and the non-safety related sump pumps (WDP-22A/B).

- Revised the problem and deficiency reporting and tracking systems

- Conducted training for engineering personnel relative to self-checking, timeliness, and sensitivity to degraded equipment conditions

- Enhanced the root cause analysis program

- Corrected the vibration issues on the fans and sump pumps

- Corrected the piping audit issues

The inspector reviewed documentation including applicable precursor cards, completed work orders, maintenance related information, revised procedure CP-111, Processing of Precursor Cards for Corrective Action Program, and the violation response. The inspector discussed these issues with licensee management and assessed recent performance in this area. The inspector also walked down selected equipment in the field and independently verified that the vibration issues were appropriately addressed.

c. Conclusions

The inspector determined the licensee's actions were appropriate to address the above degraded equipment conditions, including the extent of the related conditions. Recent performance in these areas has been noted to be acceptable, and the licensee has demonstrated improvements in these areas. Based on the above reviews, inspections and assessments, the violation was closed.

The inspector assessed the licensee's corrective action performance, with respect to this restart-related issue, in the five NRC continuing areas of concern:

- Management Oversight - Adequate
- Engineering Effectiveness - Good
- Knowledge of the Design Basis - Adequate
- Compliance with Regulations - Good
- Operator Performance - N/A
- Operator Performance - N/A

E8.2 (Closed) LER 96-011-00: Personnel Error Causes Testing Deficiency Resulting in Condition Prohibited by Improved TS

(Closed) LER 96-025-00: Personnel Error Causes Testing Deficiency Resulting in Condition Prohibited by TS

(Closed) LER 97-003-00 through Rev. 05: Personnel Errors Caused Testing Deficiencies (GL 96-01)

a. Inspection Scope (92903)

The three LERs identified testing deficiencies that did not meet the requirements in the technical specification. The licensee committed to implement the requirements in GL 96-01, Testing of Safety-Related Logic Circuits, as corrective action for the LERs. The inspectors reviewed the GL 96-01 program to verify corrective action was implemented.

b. Observations and Findings

The GL 96-01 program discussed in Section E1.1 addressed the deficiencies in the three LERs listed above. The inspectors verified that the licensee had implemented and completed a superior program to meet their commitments and the requirements in GL 96-01. The completion of GL 96-01 by the licensee was satisfactory corrective action to close these LERs.

c. Conclusions

The inspectors concluded the licensee's corrective action (GL 96-01) had been implemented. LERs 96-011-00, 96-025-00, and 97-003-00 through 05 are closed.

The inspectors assessed the licensee's performance, relative to corrective actions for these LERs, in the five areas of continuing concern:

- Management Oversight - Superior
- Engineering Effectiveness - Superior
- Knowledge of Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - N/A

E8.3 (Closed) VIO 50-302/97-05-03: Incorrect Information in Annunciator Response Procedure for Inverters.

a. Inspection Scope (92903)

Annunciator response procedure AR-701, SSF P Annunciator Response, was not changed to identify the location of the sensor input for "Battery Supplying Load Alarm" after modification MAR 93-05-07-03 was implemented. The licensee moved the location of the sensor from monitoring the battery input current shunt to monitoring the inverter's rectifier DC input voltage. The inspectors reviewed the licensee's response to verify corrective action was implemented.

b. Observations and Findings

The licensee corrected Procedure AR-701 to provide the correct location of the sensor. However, the alarm did not function as expected; it alarmed during momentary voltage transients. The licensee determined it was necessary to relocate the sensor back to its original location. The inspectors verified that plant modification MAR 93-05-07-04 was being implemented to relocate the sensor back to monitoring the battery input current shunt, and AR-701 was revised accordingly. This violation was closed.

c. Conclusions

The inspectors concluded that satisfactory corrective action was implemented to close VIO 50-302/97-05-03

The inspectors assessed the licensee's performance, relative to corrective actions for this violation, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - N/A

E8.4 (Closed) VIO 50-302/97-07-01: Failure to Follow Procedure CP-111 for the Processing of Precursor Cards (PC)

a. Inspection Scope (92903)

This violation involved three areas where the licensee did not follow Procedure CP-111: 1) the 20 day time limit for completion was exceeded; 2) non-qualified personnel were acting as the "Root Cause Team" leaders and the "Apparent Cause Evaluators"; and 3) precursor cards were improperly graded. The inspectors reviewed the licensee's corrective action to verify implementation was completed.

b. Observations and Findings

The inspectors reviewed the licensee's response to the violation stated in FPC's letter 3F0897-12 dated August 5, 1997. The corrective actions taken by the licensee and verified by the inspectors were: 1) CP-111 was revised to clarify management expectations regarding PC timeliness; 2) 30 additional personnel were trained during the week of July 28, 1997;

and 3) a followup assessment, Quality Programs Surveillance (QPS)-97-0129, dated September 22, 1997 was completed where all PCs were reviewed by quality assurance for improper grading. This assessment identified 43% of the PCs were improperly graded.

The licensee opened restart item OP-2B, dated September 22, 1997, to address the improperly graded PCs identified in QPS-97-0129.

c. Conclusions

The inspectors concluded satisfactory corrective actions were implemented to close VIO 50-302/97-07-01.

The inspectors assessed the licensee's performance, relative to corrective actions for this violation, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of Design Basis - Adequate
- Compliance with Regulations - Good
- Operator Performance - N/A

E8.5 (Closed) URI 50-302/96-201-07: EDG Not Protected Against Water Spray from the Fire Protection System Sprinkler

a. Inspection Scope (92903)

This unresolved item states: Should the deluge valves be disabled by a seismic event, water impingement could occur on both redundant diesel generators which could cause serious damage to both diesel generators. During this inspection, the inspector reviewed the design of the sprinkler system in the diesel generator rooms and the licensee's actions to resolve this issue. The inspector also determined whether any violations of NRC requirements had occurred.

b. Observations and Findings

The sprinkler system in the diesel generator area, which includes the engine room and the control panel air compressor room, is a dry, air pressurized system. A control valve, FSV-101, located in the diesel generator radiator room, receives electrical signals from heat detectors to actuate the system. The sprinkler piping downstream of Valve FSV-101 runs into both A train and B train diesel generator areas. The sprinkler piping was not originally designed with seismically qualified supports. In order to get spray down of both diesel generators as a result of a seismic event one would have to postulate failure of valve FSV-101 to maintain its pressure boundary and failure of certain branch piping in both diesel generator rooms.

The inspector determined that there was no specific requirement for the sprinkler piping to be seismically designed at Crystal River. The licensee presented evidence that the emergency diesel generators were designed to operate during actuation of the sprinkler system. The inspector observed that the diesel generators were constructed in such a way that water spray from above would be diverted away from openings to internal parts. The inspector also observed that the control panels had a drip hood installed similar to outdoor construction.

The licensee obtained documentation from the manufacturer that valve FSV-101, a Multimatic Model A-4 by Grinnell Co., was a seismically qualified style of valve. The documentation showed it was seismically qualified by test to remain closed during a seismic event, and to open upon receipt of a valid signal following the seismic event. The licensee implemented modification MAR 97-06-11-01, Fire Service Pipe Near FSV-101 Seismic Qualification, to harden the supports for the sprinkler piping in the immediate vicinity of the valve. The purpose of this modification was to prevent flooding of the diesel generator radiator room as a result of a postulated seismic event leading to failure of the sprinkler piping in that area. The inspector verified by inspection of the piping that the new supports were installed.

c. Conclusions

The inspectors determined that the original design of the sprinkler piping at the diesel generators met the requirements. The licensee took actions to improve the design in terms of postulated seismic events. Consequently this item is closed.

The inspector assessed the licensee's performance relative to this unresolved item in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of the Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - N/A

E8.6 (Closed) EA 95-126, VIO I.C.2 (04013): Corrective Actions for an Inadequate Curve 8 (Two STI's and a Revised Curve 8A and 8B) were Also Incorrect

a. Inspection Scope (92903)

This violation involved inadequate corrective action for an inadequate curve for maximum makeup tank operating pressure versus level. The inspectors followed up on the licensee's corrective actions for this violation.

b. Observations and Findings

The inspectors reviewed the licensee's response to this violation, interviewed licensee engineering and operations personnel, and reviewed the following procedures.

- OP-103B, Plant Operating Curves, Rev. 17, dated October 18, 1996; including:

- Curve 8A, Maximum MUT Operating Pressure vs. Level, Wide Range

- Curve 8B, Maximum MUT Operating Pressure vs. Level, Operating Range

- and Curve 8C, Maximum MUT Operating Pressure vs. Level, Preferred Range

- OI-6, Shift Orders, Rev. 3, dated November 14, 1996

- AI-400 C, New Procedures and Procedure Change Process, Rev. 22, dated July 22, 1997

- AI-400 F, New Procedures and Procedure Change Process for EOPs, APs, and Supporting documents, Rev. 4, dated June 19, 1997

- CP-213, Preparation of a Safety Assessment and Unreviewed Safety Question Determination (10 CFR 50.59 Safety Evaluation), Rev. 4, dated September 10, 1997

- NOD-45, Management Self Assessments and Performance Monitoring, Rev. 7, dated August 5, 1997

- AI-500, Conduct of Operations, Operations Department Organization and Administration, Rev. 94, dated April 30, 1997

- AI-1700, Conduct of Nuclear Engineering and Projects, Rev. 2, dated August 21, 1997

The inspectors verified that the revised Curve 8 in OP-103B, for maximum makeup tank overpressure, provided additional margin and clearly delineated acceptable and unacceptable operating regions. The inspectors also verified that the process dealing with procedure revisions was revised to require that engineering and other interdisciplinary reviews be performed. Also, the inspectors noted that the process for issuance of Short Term Instructions (STIs) was revised so that STIs would not be used in place of the normal procedure revision process.

The inspectors verified that management oversight of the operations and engineering interface had been strengthened through enhanced processes and additional personnel. Inspectors also verified that the licensee had established a single point of contact, called an issue manager, for important technical issues. Inspectors verified that a new operations management position had been established to provide increased oversight of plant operations and interface with operators. Also, inspectors verified that a special Rapid Engineering Response team had been established. In addition, inspectors verified that expectations for engineering and operations personnel had been strengthened and communicated.

c. Conclusions

The inspectors concluded that the licensee's corrective actions had been implemented, included actions to prevent recurrence of the violation, and were effective improvements. EA 95-126, VIO I.C.2 is closed.

The inspectors assessed the licensee's performance, relative to corrective actions for this violation, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of the Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - Good

E8.7 (Closed) EA 96-365, VIO C (03013): Inadequate Corrective Actions for 10 CFR 50.59 Evaluation Errors and for Inadequate Containment Penetration Surveillances

a. Inspection Scope (92903)

This violation included two examples where engineers were involved in inadequate corrective actions related to emergency diesel generator loading and one example where engineers were involved in inadequate corrective actions related to containment surveillance procedures. The inspectors reviewed the licensee's response to this violation and followed up on the licensee's corrective actions for this violation.

b. Observations and Findings

The inspectors verified that the licensee had implemented a revised corrective action program and a revised 10 CFR 50.59 program. The NRC had inspected those programs and documented the results in IRs 50-302/97-07, 97-08, 97-11, and 97-13. The NRC had also inspected the Quality Programs monitoring of the new corrective action process and documented the results of that inspection in IR 50-302/97-11. In addition, the NRC had inspected the licensee's MCAP II programmatic actions addressing engineering performance and documented the results of that inspection in IR 97-13. The previous NRC inspections determined that the revised 50.59 process was adequate. Also, the revised corrective action process for Grade A and B Precursor Cards was adequate. However, a determination on the effectiveness of the revised corrective action process, for Grade C and D Precursor Cards, was not made pending the results of further NRC followup of IFI 50-302/97-11-04, Corrective Actions for Approximately 4000 Precursor Cards Not Tracked to Completion.

c. Conclusions

The inspectors concluded that the licensee's corrective actions had been implemented and included actions to prevent recurrence of the violation. EA 95-365, VIO C is closed.

The inspector assessed the licensee's performance, relative to corrective actions for this violation, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - adequate
- Knowledge of Design Basis - adequate
- Compliance with Regulation - Goood
- Operator Performance - N/A

E8.8 (Closed) EA 97-162 (01013): Inadequate Safety Evaluations for A' J Operator Actions for Design Basis SBLOCA Mitigation

a. Inspection Scope (92903)

This violation involved inadequate 10 CFR 50.59 safety evaluations for procedure and FSAR changes that added four required operator actions, and changed one, for mitigation of a design basis small break loss of coolant accident (SBLOCA). The inspectors reviewed the licensee's corrective actions for this violation.

b. Observations and Findings

The inspectors reviewed the licensee's response to this violation, interviewed licensee engineering and operations personnel, and reviewed the following procedures and training records:

- CP-213, Preparation of a Safety Assessment and Unreviewed Safety Question Determination (10 CFR 50.59 Safety Evaluation), Rev. 4, dated September 10, 1997

- Nuclear Operations Engineering Standard AI-1700/OES-03, 10 CFR 50.59 SA/USQD Expectations, Rev. 1, dated May 9, 1997

- Nuclear Operations Training Department Lesson Plan, Special Technical Training, 10 CFR 50.59 Safety Evaluation, SA/USQD Lessons Learned Training, NUCST-0067LL, Two Hours, Rev. 0, dated July 23, 1997

The inspectors noted that the NRC had inspected the 10 CFR 50.59 program, including the Safety Analysis Group and training, and documented the results of that inspection in IR 97-08. During this inspection, the inspectors verified the licensee had again revised the 10 CFR 50.59 procedure and had conducted training on the "lessons learned" from this violation. The inspectors also verified that the licensee had submitted TS Change Request 210 to the NRC and that it adequately addressed the unreviewed safety questions of this violation. A scheduled NRC inspection of emergency operating procedures, prior to plant restart, will verify that the EOPs include operator actions as described by the licensee in TS Change Request 210 and that these actions are effective in mitigating an SBLOCA.

During this review, the inspectors had comments on information recently added to Procedure CP-213. Section 4.6.3.2 stated that only a Safety Assessment, and not an Unreviewed Safety Question Determination (USQD), is required for relocation of information from the FSAR into a referenced program (e.g., fire protection), because it already exists in an NRC reviewed/approved licensing basis document (e.g., relocation of existing information by reference to docketed material that has been reviewed/approved by the NRC). The inspector noted that this could improperly allow removal of significant information from the FSAR without having an NRC Safety Evaluation Report (SER) specifically approving the removal and without performing a USQD. Once the information was removed from the FSAR, subsequent modifications or procedure changes could potentially be made without performing a USQD, because they no longer represented changes to the plant or procedures <u>as described in the SAR</u>. CP-213 defines the SAR as the latest issued FSAR, pending FSAR changes not yet incorporated into the controlled FSAR, and NRC SERs.

Section 4.6.3.7 stated that only a Safety Assessment, and not a USQD, is required for organizational changes made as a specific NRC commitment (e.g., GL, BL, NOV, or LER response). The inspector noted that NRC review of an LER or NOV response does not constitute review and approval of a change to the licensing basis of the plant. Therefore, these documents were not an appropriate basis for bypassing the 10 CFR 50.59 requirement for performing a USQD. The licensee planned to review the inspectors' comments and revise CP-213 as needed.

c. <u>Conclusions</u>

The inspectors concluded that the licensee's corrective actions had been implemented and included actions to prevent recurrence of the violation. EA 97-162 is closed.

The inspectors assessed the licensee's performance, relative to corrective actions for this violation, in the five areas of continuing NRC concern:

- Management Oversight - Adequate
- Engineering Effectiveness - Adequate
- Knowledge of the Design Basis - Adequate
- Compliance with Regulations - Adequate
- Operator Performance - Adequate

E8.9 (Open) EA 97-330 (01013): Unreviewed Safety Question Involving Added EDG Protective Trips

a. Inspection Scope (92903)

This violation involved an inadequate 10 CFR 50.59 safety evaluation for a modification that had been made to the EDGs in 1987. The modification had added five protective trips to the control circuit for each EDG that were not bypassed during emergency operation and did not have two out of three coincidence logic. The inspectors reviewed the licensee's corrective actions for this violation.

b. Observations and Findings

The inspectors reviewed the licensee's response to this violation, interviewed licensee engineering and operations personnel, and reviewed the procedures and training records for improving the 10 CFR 50.59 process as listed above for violation EA 97-162.

In addition, the inspectors reviewed the modification package for MAR 97-08-01-01, EDG Protective Trips, and compared it with License Amendment Request 219: Emergency Diesel Generator Protective Relays Unreviewed Safety Question, dated September 12, 1997. The modification was to reroute the added EDG protective trips so that they would trip the EDG output breaker instead of tripping the EDG. The inspectors noted that the modification was appropriately described in the license amendment request.

The inspectors noted that the 10 CFR 50.59 safety evaluation for MAR 97-08-01-01 was for installing the modification while the affected EDG was out of service, and did not provide for returning the EDG to service. It stated that another 10 CFR 50.59 evaluation would have to be performed before returning the EDG to service. At the end of this inspection, the licensee had not received approval of the license amendment request from the NRC. However, the MAR was installed on the B EDG, which was scheduled to be returned to service in four days. In response to inspector questions, the licensee stated that they if they did not receive an NRC approval in time, they would complete the appropriate Justification for Continued Operation (JCO) evaluations to return the EDG to service when it was ready.

c. Conclusions

This violation remains open for further review of the licensee's resolution of the unreviewed safety question and installation of MAR 97-08-01-01.

E8.10 (Closed) LER 96-24-01: Plant Modification Causes Unanalyzed Condition Regarding Emergency Feedwater

a. Inspection Scope (92903)

This LER involved a modification that removed the automatic start signal from the A side of the Emergency Feedwater Initiation and Control (EFIC) system to the turbine-driven emergency feedwater pump. The modification failed to recognize that a previous modification, which had installed an automatic trip of the turbine-driven emergency feedwater pump at 500 psig RCS pressure (decreasing), had relied upon the automatic start of the emergency feedwater pump from the A side of the EFIC. The inspectors followed up on the licensee's corrective actions for this violation.

b. Observations and Findings

The inspectors reviewed the licensee's stated corrective actions in the LER and reviewed the following procedures and records:

- Nuclear Operations Engineering Standard OES-4, System Assignment Expectations, Rev. 0, dated January 22, 1997
- Nuclear Operations Engineering Standard OES-1, Design Review Board Expectations, Policies, and Practices, Rev. 0, dated January 22, 1997
- Operations Instruction OI-41, System Operator Program, Rev. 2, dated May 31, 1996
- Nuclear Plant Technical Support Manual, Rev. 9, dated December 1995
- NEP-104, Interface Design Control, Rev. 7, dated March 31, 1997
- NEP-210, Modification Approval Records, Rev. 16, dated March 31, 1997
- NEP-211, Commercial Grade Design Control, Rev. 17, dated March 31, 1997
- Nuclear Operations Department Manual NOD-38: Planning, Budgeting, and Scheduling Project Controls, Rev. 4
- Training Lesson Plan NUCST-2011, Solution Sets Bases for TS

Changes (Including SBLOCA Analysis), 6 hours

- Extent of Condition Review for Omission of Information from Design Basis Document Temporary Changes, dated March 14, 1997

The inspectors verified that nuclear engineering management had provided interim direction stating that all design modifications will receive a 10 CFR 50.59 evaluation, and that the 10 CFR 50.59 evaluations will be "stand-alone" documents. Also, the inspectors verified that a separate group had been established for reviewing 10 CFR 50.59 evaluations. Inspectors reviewed the extent of condition review for the omitted temporary change to the EFW/EFIC design basis document, and noted that the licensee found that one additional similar error had been made during the period of 1988 through 1993. Inspectors verified the issuance of expectations for and assignment of system ownership teams, expectations for project teams, and expectations for review of modification packages. In addition, inspectors verified the establishment of a Design Review Board. Inspectors verified that training had been provided to operators regarding SBLOCA design basis accident analysis and that engineers were scheduled to receive the training.

c. Conclusions

The inspectors concluded that the licensee's corrective actions had been almost all implemented and included actions to prevent recurrence of the event. LER 96-24-01 is closed.

The inspectors assessed the licensee's performance, relative to corrective actions for this violation, in the five areas of continuing NRC concern:

- Management Oversight - Adequate
- Engineering Effectiveness - Adequate
- Knowledge of the Design Basis - Adequate
- Compliance with Regulations - Adequate
- Operator Performance - Good

E8.11 (Closed) EA 96-365, EA 96-465, EA 96-527, VIO B (Example 1) (02013): Failure to Update Applicable Design Documents to Incorporate Design Information

a. Inspection Scope (37550, 92903)

This violation involved the licensee's failure to update the Final Safety Analysis Report (FSAR), the Enhanced Design Basis Document (EDBD), and the Improved Technical Specifications (ITS) Bases with regard to operation of the steam turbine driven auxiliary feedwater pump (EFP-2) for certain accident scenarios. The inspector followed up on the licensee's corrective actions by reviewing procedure changes, training records, internal licensee correspondence, and interviewing engineering personnel.

b. Observations and Findings

The inspector noted that resolution of this violation was being tracked under licensee restart item D-39. Related corrective actions implemented to address this issue were being tracked under MCAP II and other licensee restart items, which included D-40 and OP-6. The inspector reviewed some of the actions addressed under restart item OP-6 and documented the results in IR 50-302/97-07. The corrective actions addressed by OP-6 included revisions to various Nuclear Engineering Procedures (NEPs) to enhance the design control process.

Changes to the NEPs included, but were not limited to, the incorporation of Procedure CP-213 requirements, additional guidance regarding design inputs, and guidance regarding prompt revision to design basis documents following implementation of a plant modification.

During this current inspection, the inspector verified that the licensee had initiated documentation to update the FSAR, EDBD, and ITS Bases to reflect operation of EFP-2 for certain accident scenarios. Licensee interoffice correspondence (IOC) 97-0165, dated February 25, 1997, transmitted the FSAR change package from Nuclear Operations and Engineering (NOE) to Nuclear Licensing. Temporary changes 552, 553, and 554 had been incorporated into the EDBD. The ITS and ITS Bases changes were addressed in TS Change No. 210, dated June 14, 1997. The inspector also reviewed the actions taken by the licensee to strengthen the expectations regarding the use of the FSAR, EDBD and the ITS Bases. These actions included implementation of procedure NOD-55, Control of Design Basis Information, and development of the training course ST-1222, Plant Design Basis and Configuration Management. This training course was being provided to NOE, operations, Nuclear Regulatory Assurance, and other selected personnel involved in preparing modifications or performing safety assessments or USQDs.

c. Conclusions

The inspector concluded that the licensee's corrective actions for this violation were satisfactory. This item is closed.

The inspector assessed the licensee's performance, relative to corrective actions for this violation, the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of the Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - N/A

E8.12 <u>(Closed) EA 96-365, EA 96-465, EA 96-527, VIO B (Example 2) (02013): Failure to Include Applicable Design Information in the Design Input Requirements for a Modification</u>

a. <u>Inspection Scope (37550, 92903)</u>

This violation involved a failure to include design basis information in the design input requirements for the MAR which disabled the automatic opening of valve ASV-204. The inspector followed up on the licensee's corrective actions by reviewing procedure changes, training records, internal licensee correspondence, and interviewing engineering personnel.

b. <u>Observations and Findings</u>

The inspector noted that resolution of this VIO example was being tracked under licensee restart item D-40. Related corrective actions implemented to address this issue were being tracked under the MCAP II and other licensee restart items which included D-39 and OP-6. The inspector reviewed some of the actions addressed under restart item OP-6 and documented the results in NRC IR 50-302/97-07. The corrective actions addressed by OP-6 included revisions to various NEPs to enhance the design control process.

During this current inspection, the inspector verified that the licensee had incorporated design basis information into MAR 96-11-01-01. This MAR was prepared to reinstall the automatic open signal to valve ASV-204. The field work for this MAR was completed on July 10, 1997. All of the testing associated with this MAR had not been completed at the conclusion of this current inspection. The inspector also noted that the licensee had initiated documentation to update the FSAR, EDBD, and ITS Bases to reflect this MAR. IOC 97-0165, dated February 25, 1997, transmitted the FSAR change package from NOE to Nuclear Licensing. Temporary changes 552, 553, and 554 had been incorporated into the EDBD. The ITS and ITS Bases changes were addressed in TS Change No. 210, dated June 14, 1997. The inspector also reviewed the actions taken by the licensee to strengthen the expectations regarding the use of the FSAR, EDBD, and the ITS Bases. These actions included implementation of procedure NOD-55, Control of Design Basis Information; training course ST-1222, Plant Design Basis and Configuration Management. This training course was being provided to NOE, operations, Nuclear Regulatory Assurance, and other selected personnel involved in preparing MARs or performing safety assessments and/or USQDs.

c. Conclusions

The inspector concluded that the licensee's corrective actions for this violation were satisfactory. This item is closed.

The inspector assessed the licensee's performance, relative to the corrective actions for this violation, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of the Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - N/A

E8.13 Followup on Restart Issue Resolution - BWST NPSH Concern (FPC Restart Issue D-18)

a. Inspection Scope (92903)

The inspector followed up on the licensee's actions to resolve a concern regarding net positive suction head (NPSH) for the emergency core cooling system (ECCS) pumps when the spent fuel pumps (SFP) were running in recirculation to the borated water storage tank (BWST).

b. Observations and Findings

The licensee had documented this concern in problem report (PR) 96-0360 and PC 97-0085. The licensee determined that this problem would be resolved prior to restart from the current shutdown. The resolution of this problem was being tracked as licensee Restart Issue D-18. The inspector noted that the licensee had completed the corrective actions to address this concern. These corrective actions included, but were not limited to, using SFP-2 instead of SFP-1B as the preferred method for BWST recirculation; revisions to numerous calculations for the ECCS to demonstrate that the flow rate for the SFP-2 would have a negligible impact on the operability of the associated ECCS pumps; determination of the flow rate to be used to revise the calculations; and revisions to various procedures and design basis documents, etc. The inspector noted that the following calculations and procedures had been revised:

M94-0013	Building Spray System Hydraulic Spray, Rev. 4
M93-0047	CR3 Makeup System Hydraulic Analysis, Rev. 3
M94-0047	CR3 Decay Heat Removal System Hydraulic Studies, Rev. 2
M95-0004	Makeup Pump NPSH Evaluation During Post LOCA Cooling and S/O to the RB Sump, Rev. 2

M95-0016 BWST Swapover and Minimum Allowable Level Evaluation, Rev. 2

M96-0010 Head Loss in BWST to Makeup Pump, Rev. 1

M95-0005 Minimum BWST Level Necessary to Prevent Vortexing During Drawdown, Rev. 4

M94-0053 Allowable MUT-1 Indicated Overpressure vs. Indicated Level, Rev. 5

M97-0043 Head at Tie-In of Makeup Tank Surge Line to Makeup Pump Suction Line, Rev. 0 (Note: This calculation stated that it was performed to determine past operability and should be used for historical information only)

The inspector discussed this issue with licensed operators in the main control room and determined that they were aware of this issue and the changes to the applicable procedures. In addition to the above calculations, the inspector verified that the following procedures had been revised to address operation of SFP-2 for recirculation of the BWST

OP-406 Spent Fuel Cooling System, Rev. 57

SP-320 Availability of Boron Injection Sources and Pumps, Rev. 67

c. Conclusions

The inspector concluded that the licensee had taken satisfactory corrective actions to resolve this issue prior to restart. Consequently, this item is closed on the NRC Restart List.

The inspectors assessed the licensee's performance, relative to the corrective actions to resolve this issue, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of the Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - Good

E8.14 (Closed) LER 97-017-00: Personnel Error Caused Inadequate Electrical Separation Of High Pressure Flow Indicators (FPC Restart Issue D53A).

a. Inspection Scope (92903)

This item addressed multiple examples of the incorrect electrical insulation material used during installation of modifications in the main control boards.

The inspector reviewed the licensee's root cause and corrective actions. The inspection included reviews of licensee procedures relating to the use of electrical insulation material, and interviews with licensee personnel. The inspector independently verified a sampling of the corrective actions by performing a main control board walk down, and inspection of the rework associated with this LER.

b. Observations and Findings

During installation of a modification in the main control boards, the licensee noticed that the electrical insulation material on the cables of a previously installed modification differed from the material which they were presently using. The licensee subsequently identified that the wrong sleeving insulation material had been used on three modifications. The incorrect insulation material which had been used was Nextel. The qualified material for the application was Siltemp. The inspector noted that the appearance of the Nextel and the Siltemp looked similar in color and were both braided materials. However, the Nextel, which was 1/8-inch in diameter, was smaller in diameter than the Siltemp. The Nextel was to be used as a tie cord material and not for sleeving electrical cables/conductors. Licensee corrective actions completed included the following:

- performed an extent of condition review to identify other areas where Nextel may have been used;

- reworked and replaced the Nextel with Siltemp insulation on areas identified which had incorrect use of Nextel;

- clarified the procedure describing the use of Nextel and Siltemp, and

- developed a procedure for Siltemp sleeving to include an in-process verification of the installed Siltemp.

The inspector found that an extent of condition review and a walk down had been completed by the licensee for the purpose of identifying any other misuse of the Nextel. The inspector noted that the walk down did not include all pieces of equipment where the electrical separation was required. However, the walk down did include the areas where most of the design and modification activity had taken place and where it was most probable that the Siltemp material would have been used. The licensee identified that Nextel had incorrectly been used on three modifications: 1) MAR 96-02-09-01, High pressure flow indicators; 2) MAR 96-03-12-01, Emergency diesel generators Kilowatt indicators; and 3) MAR 91-08-26-04, Relay wiring separating non safety related signals from safety related signals.

The inspector reviewed the rework packages for the three modifications and noted that there was in-process verification to identify that Siltemp material had been used. In addition, Procedure MP-405A, used to describe application of Nextel and Siltemp, had been clarified and gave explicit instructions describing the correct use of Nextel. The inspector reviewed the reworked cabling in the main control boards and noted that the Nextel had been removed and replaced, as required in the rework packages. In addition, the inspector walked down the rest of the main control boards and did not identify any other misuse of Nextel. The licensee had communicated to the technicians the correct use of Nextel through department meetings and via a memo. The inspector interviewed a technician, who had done modification work at the main control boards, and found that the technician was well versed with the Nextel issue and knew the correct insulation material to use. Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 97-21, the licensee meets the criteria for enforcement discretion per Section VII.B.2 of the NRC Enforcement Policy as described in NUREG-1600. Consequently this item is closed and is identified as another example of Non-cited Violation NCV 50-302/97-21-01, Examples of Noncompliances in Design Control, 10 CFR 50.59 Evaluations, Procedure Adequacy/Adherence, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

c. Conclusions

The inspector determined that the licensee had appropriately addressed the issues with the incorrect use of Nextel and had completed the required corrective actions. Based on the inspection finding and observations this LER is closed.

The inspector assessed the licensee's corrective action performance, with respect to this restart-related issue, in the five NRC continuing areas of concern:

- Management Oversight - Good
- Engineering Effectiveness - Adequate
- Knowledge of the Design Basis - N/A
- Compliance with Regulations - Adequate
- Operator Performance - N/A

E8.15 (Closed) VIO 50-302/EA 95-126 NOV II.B: Failure to Take Adequate Corrective Action for Required Tank Volumes, Level, and Suction Points (FPC Restart Issue OP-12)

a. Inspection Scope (92903)

This item addressed multiple examples identified by the licensee and the NRC in which safety related and technical specification related tank levels, suction position, volume requirements, and their respective relationships, were incorrectly being used by the licensee.

The licensee's response to the violation was reviewed. The root cause and corrective actions were reviewed and verified by the following: 1) procedures, FSAR, and Technical Specifications reviews; 2) walk-in and scheduled interviews with engineering management, supervisors, staff, and other licensee personnel; 3) review of selected tank level, suction point and volume calculations; and, 4) field inspections of tank construction, geometries and specifications, tank gauge locations and measurement methods, and tank suction points.

b. Observations and Findings

By letter (3F0996-01) dated September 9, 1996, the licensee replied and agreed with the violation. The licensee stated that the reason for the violation was inadequate prioritization and management of existing and changing work loads associated with a rapidly changing environment.

Examples included: a change in internal philosophy to perform more work in-house, reduction of permanent staff, and the changes associated with relocating the engineering staff to the plant site.

Licensee corrective actions completed included the following:

- an organizational structure with the required resources consistent with the work load
- work loads and priorities consistent with safety significance and available resources
- increased design margins, thereby decreasing the amount of direct engineering effort, and
- comprehensive review of tank calculations associated with level, instrument errors, suction points, and usable volume

The inspector reviewed and verified specific examples of each corrective action. Relocation of the engineering staff to the site had been completed, and the design engineering staff had been increased. In addition, the use of specialty contractors had increased commensurate with emergent work loads and required specializations. To address short term issues, there had been a rapid response team (RRT) created. The team's charter included: 1) rapid engineering support to other plant organizations, 2) support to reduce the backlog of request for engineering assistance (REA), and 3) design support on minor modifications, commercial grade work, and plant equivalency replacement evaluations. The inspector reviewed the experience of the team's members. There was a total of eight engineers which made up the team. With the exception of two design engineers, the rest were degreed engineers, and included two registered professional engineers. There was good variation amongst the disciplines, i.e., mechanical, electrical, and nuclear. The years of experience ranged from 10 to 20 years, and one individual had 36 years of experience. The experience was very diversified, including: operations, design, testing, systems, maintenance, licensing, and plant construction. However, the inspector found that the RRT was not presently fully activated. Team members were on loan to special start-up projects. Engineering management indicated that the team would return to its normal charter after startup.

Efforts to increase design margins were evident. Examples included: modification of the reactor building sump screen, and the emergency diesel generator upgrade. However, the inspector found that these items were being tracked separately by the NRC and the licensee.

The communications amongst the engineering groups appeared to be good. The information provided by the various engineering groups was consistent. Engineering management had a daily morning meeting and a weekly meeting. The agenda of the daily morning meetings included: prioritization of activities, review of modifications matrix, status review of set point calculations for EOPs, and review of emergent work.

The inspector reviewed the REA backlogs. The inspector found that the REA backlogs have been continuously decreasing. The backlog was reviewed on a weekly basis at the weekly engineering managers meeting. Engineering management appeared to have a good handle on REA activity. Backlog indicators were printed weekly, and tracked on a daily basis. There was a strong focus on planning, accountability, and the continuous review of the required resources, including specialist contractors to achieve the REA workload. Engineering was strongly committed to achieving a goal of less than a 200 REA backlog prior to restarting. Additionally, engineering indicated the organization goals would be to maintain a maximum REA backlog of 200. The inspector questioned the use of the engineering contractors after the restart workload had been completed. Engineering management indicated that after the startup they planned on decreasing the contractor resources but would increase the permanent staff.

The inspector reviewed the licensee's activities related to the tank calculations. The licensee had embarked on a comprehensive program to review the calculations relating to tank requirements and had compiled a list of tanks in order of safety significance. The calculations included analysis for usable tank volume and the corresponding required tank levels, taking into account inaccuracies associated with instrument errors. Analysis on net positive suction pressure and vortexing was included where appropriate. Data from vendor tank prints and technical information, information notices, and industry reports was used, as well as licensee experience with the tanks. The inspector found that portions of the analysis were performed by contractors. However, the project was owned and driven by the licensee's engineering group. This provided good oversight on technical specifics (design specifications) relating to the plant's applications, i.e., tank tilt, materials, and modification history. Appropriate independent verification was noted on the tank calculations. In addition, management reviews and approvals were required prior to accepting results.

The inspector reviewed the design basis, FSAR, and Technical Specifications for selected tanks, and verified the licensee had included the various requirements in the analyses. In addition, the inspector reviewed assumptions made on calculations and analysis methods and found them to be consistent and appropriate. The inspector had various discussions with engineering relating to specifics on tank calculations, assumptions, equations, and analysis methods and found that the licensee was very well versed with the tank requirements and overall analysis. Field inspections and control room board walk down were made on the borated water storage tank, condensate storage tank, emergency feed water tank, and the emergency diesel generator day tanks to verify data used in the calculations and that surveillance procedures appropriately addressed Technical Specification requirements. Gauge pressure tap locations, suction heights, piping size, and flow path losses used in the calculations were verified. Accessability of gauges used in taking measurements for surveillance was reviewed and found to be appropriate. Surveillance procedures had been appropriately updated and reflected the required heights on the tanks and took into consideration the volume analysis results and instrument errors. However, the inspector found that the height requirements on the surveillance procedures for the emergency diesel generator day tank were too conservative. For example, Technical Specification 3.8.1.4 required level to be >/= to 245 gallons of fuel oil. Based on the licensee's tank calculation result (taking into consideration the instrument error) this would equate to a height >/= to 22.2 inches. The surveillance procedure, SP-345A, however, had an administrative limit of >/= 24 inches, and that height equated to 348.7 gallons. The inspector later found that these differences were associated with the modification to the emergency diesel generator and that this was being tracked under a separate issue number.

c. Conclusions

The licensee's actions to correct numerous tank parameters had appropriately addressed the difficulties associated with the engineering structure, work prioritization, and available resources to perform the work. However, continued monitoring after startup is warranted. The tank calculations were thorough and the results had been appropriately incorporated into the required procedures. Based on these observations and findings, this violation is closed.

The inspector assessed the licensee's corrective action performance, with respect to this restart-related issue, in the five NRC continuing areas of concern:

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of the Design Basis - Adequate
- Compliance with Regulations - Adequate
- Operator Performance - N/A

E8.16 (Open) URI 50-302/95-02-02: Control Room Habitability Envelope Leakage

a. Inspection Scope (92903)

On November 10, 1997, the licensee submitted a letter to the NRC titled: "Control Room Habitability, NUREG-0737, Item III.D.3.4." The purpose of the letter was to provide a description of the recent Control Complex Habitability Envelope (CCHE) in-leakage testing and the revisions to the calculational methodology used by the licensee for determining control room operator dose. The licensee stated that they planned to use the results of the testing and calculations to demonstrate operability of the CCHE and Control Room Emergency Ventilation System (CREVS) prior to plant restart from the current outage. The inspectors reviewed the technical content of the letter, reviewed the test conditions and data, inspected CCHE conditions in the plant, performed independent analyses, and discussed CCHE in-leakage with licensee engineering and licensing personnel.

b. Observations and Findings

In the November 10 letter, the licensee stated that since the CCHE is maintained at a neutral pressure by the CREVS, the only mechanism for developing an in-leakage under maximum hypothetical accident conditions is the effect of the outside wind on the building. The licensee also stated that they had examined every wall of the CCHE, assessed the condition of each penetration, and sealed penetrations to eliminate potential in-leakage sites that may have existed previously. In addition, the licensee stated that testing has demonstrated that in-leakage is extremely low for a structure the size of the CR-3 control complex.

The licensee's CCHE test concluded that there was in-leakage of 462 cubic feet per minute (CFM) with the following test conditions: CREVS in emergency recirculation, a combination of auxiliary building exhaust and supply fans running to generate a negative pressure of 0.17 inches w.g in the auxiliary building, and turbine building and intermediate building pressures at zero inches w.g. (equal to atmospheric pressure). During the test, pressure in the control room was measured to be at negative 0.045 inches water gauge (w.g.) By using the test results and a calculation method different than that previously approved by the NRC, the licensee concluded that the CCHE leakage during a maximum hypothetical accident (with loss of offsite power and no auxiliary building fans running) would not cause operator dose limits to be exceeded. In that operator dose analysis, the licensee assumed that the only cause of CCHE in-leakage would be outside wind.

The inspectors used the licensee's test data to calculate what the CCHE leakage would be if the NRC-approved Standard Review Plan (SRP) method were used. The SRP requires determining CCHE leakage with a 0.125 inches w.g. pressure in the CCHE and dividing that number by two, adding 10 CFM for opening and closing of doors and also adding leakage through CREVS closed boundary dampers due to pressures caused by the CREVS fans running. The test data showed that, with 0.125 inches w.g. in the CCHE and zero pressure in the auxiliary building, the out-leakage from the CCHE to the auxiliary building would be 462 CFM. (The differential pressure from the CCHE to the auxiliary building during the test was 0.17 minus 0.045 inches w.g., which equals 0.125 inches w.g. Also, since the auxiliary building was the only area with a lower test pressure than the CCHE, essentially all of the 462 CFM of CCHE outleakage must have gone into the auxiliary building.) During the test, pressure in the CCHE was stable. Therefore, in-leakage from the turbine building, intermediate building, and outside atmosphere into the CCHE had to equal the out-leakage to the auxiliary building of 462 CFM. Since the air leakage rate varies as the square root of the differential pressure, SRP in leakage across all CCHE walls other than the auxiliary building wall, with 0.125 inches w.g. pressure in the CCHE would be about: 462 times the square root of (0.125 divided by 0.045), which equals 770 CFM. Adding the 462 CFM for the auxiliary building wall to the 770 CFM for all other walls results in a total CCHE leakage of 1232 CFM. Then, dividing the 1232 CFM by 2 equals 616 CFM of SRP leakage. Adding 10 CFM for doors and zero for CREVS boundary dampers (because CREVS fans were running during the test and because the new boundary dampers are to have zero leakage) results in an approximate SRP CCHE leakage value of 626 CFM. The inspectors noted that 626 CFM was substantially higher than the 355 CFM maximum SRP CCHE leakage that had been stated by the licensee in their letters in the 1980s to the NRC in response to NUREG-0737, Item III.D.3.4. Since the licensee's analysis based on 355 CFM of SRP CCHE leakage resulted in a calculated operator dose that was near the regulatory limit of 30 REM to the thyroid, the inspectors assessed that a similar analysis based on approximately 626 CFM of SRP CCHE leakage would result in a calculated operator dose that would exceed the regulatory limit.

The inspectors used the test data to estimate the cumulative size of leakage paths in the CCHE walls. Since each square inch of leakage path will leak about six CFM at 0.125 inches w.g. differential pressure, the inspectors calculated that the cumulative size of leakage paths in the CCHE to auxiliary building wall was approximately 462 CFM divided by 6 CFM per square inch, or approximately 77 square inches. Similarly, the leakage paths of the other walls of the CCHE were calculated to be approximately 770 CFM divided by 6 CFM per square inch, or approximately 128 square inches. The total cumulative CCHE leakage paths were therefore approximately 77 plus 128, or 205 square inches.

The inspectors reviewed this evaluation with licensee engineers and inquired about the locations of the approximately 205 square inches of CCHE leakage paths and the difficulty in sealing them. The engineers found no fault with the inspector's analysis and stated that the leakage paths were primarily in cable trays that penetrated the CCHE walls. The spaces between the cables stacked in the trays were not fully sealed, resulting in detectable air leakage during the test. Most of these cable tray penetrations were from electrical rooms in the lower levels of the CCHE to the auxiliary building and to the turbine building. The inspectors looked at some of these cable tray penetrations in the plant and observed that physical access to them was difficult. Licensee engineers stated the licensee was testing a new fire retardant sealant and applicator, and planned to apply the sealant to the cable tray penetrations in early 1998.

The inspectors noted that the licensee's recent CCHE leak test data included measured differential pressures, both before and during the tests, between various rooms in the CCHE and the auxiliary building or the turbine building. Using the October 11, 1997 test data, the inspectors plotted the pressures in the various rooms with respect to atmospheric pressure. The plot revealed that there were substantial differences in pressures among the various rooms. The pressures in some of the CCHE rooms, with auxiliary building pressure at minus .04 and minus .17 inches w.g., are listed below.

	Pressure (in. w.g.)	
Auxiliary Building	-0.04	-0.17
CCHE Room		
Control Room	+0.06	-0.04
"B" 4160 Volt SWGR	-0.01	-0.10
Cable Spreading Room	-0.07	-0.16
EFIC/480 Volt SWGR	-0.10	-0.20

In each case, the pressure in the control room was higher than in any other CCHE room. The electrical rooms below the control room were each at a substantially lower pressure than the control room. At least one, the EFIC/480 Volt SWGR room, was at a lower pressure than the auxiliary building. The inspectors confirmed the above CCHE room relative pressures, with the CREVS in recirculation and the normal auxiliary building ventilation operating, by entering each CCHE room and noting the velocity of air flow past each room's doors when cracked open. The inspectors concluded that the CREVS fans were causing substantial differences in pressures among the CCHE rooms. The inspectors also noted that when the auxiliary building pressure was reduced by 0.13 inches w.g. (from -.04 to -.17), the pressure in each room in the CCHE went down by about 0.10 inches w.g., and the differences in pressures among rooms remained approximately constant. This provided further indication that the differences in pressures among CCHE rooms were caused by CCHE fans and not by forces originating outside the CCHE. As a result, there were also significant pressure differences between certain CCHE rooms and the auxiliary building or the turbine building. These pressure differences existed in rooms that had known cable tray leakage paths to the auxiliary building and turbine building. It was likely that a substantial portion of the CCHE leakage that was measured during the recent test was caused by the CREVS fans. Also, contrary to the licensee's analysis in their November 10, 1997 letter, it was likely that a substantial amount of CCHE leakage would be caused by the CREVS fans during a maximum hypothetical accident. Licensee personnel stated that they would address the CCHE leakage due to CREVS fans in a planned revised submittal to the NRC.

c. Conclusions

The inspectors concluded that the licensee's CCHE leakage analysis, in their November 10, 1997 letter to the NRC, failed to recognize the potential for CREVS fans to cause a substantial amount of CCHE leakage during accident conditions. Licensee personnel stated that they would address the CCHE leakage due to CREVS fans in a revised submittal to the NRC. URI 50-302/95-02-02 remains open for further licensee analysis and NRC review.

E8.17 (Closed) VIO 50-302/97-01-09: Inadequate Corrective Actions for Cable Ampacity

(Closed) LER 97-31-00: Inadequate Cable Sizing Due to Nonconservative De-rating Factors Could Reduce the Cable Remaining Qualified Life

a. Inspection Scope (92903)

The violation involved a case where a potential non-conformance with regard to the ampacity of electric cables had been identified by the licensee, but the potential non-conformances were not corrected nor satisfactorily resolved through analysis. The licensee's response dated April 23, 1997, stated that the technical issues which were the subject of the violation would be resolved through implementation of Restart Issue D-22. The inspector reviewed the licensee's analysis performed under Issue D-22. The violation indicated weaknesses in the licensee's Corrective Action Program, however that program has been substantially upgraded and has been the subject of two recent NRC inspections. Therefore, this inspection focused on the technical issue only.

b. <u>Observations and Findings</u>

As part of the resolution of Issue D-22, the licensee performed Case Study CSEH-97-0012A, Electrical Calculation E-91-0020 R0 Assessment, which reviewed all the known potential problems with the ampacity of power cables and reviewed all relevant engineering documents issued since initial plant startup, to determine whether those documents contained information affecting the ampacity calculation for power cables. The ampacity calculations consisted of a separate analysis sheet for each cable. Within the case study, after the reviews stated above were completed, ampacity calculation sheets were marked up with correct information as necessary. The case study was subjected to independent review and was approved by the responsible supervisor. Therefore, the case study helped ensure that potential problems were satisfactorily analyzed and that the ampacity calculation accurately reflected the as-built configuration. There were separate case studies for control cables and vital ac cables as well.

The ampacity calculations, (i.e. E-91-0020) were performed using ICEA Publication P-46-426, Power Cable Ampacities, as a guide. This standard determines ampacity of cable in cable tray by applying a derate factor based on the total number of conductors in the tray to the ampacity of cable in free air. In cases where the cable tray is wrapped with Thermo-Lag, Mecatiss or a combination of the two, an additional fairly severe derate factor must be applied. The fire wrap derate factor was recently revised, and this sub-issue will be discussed later in this section. A number of cables in a number of tray sections had insufficient ampacity when the ICEA and fire wrap derate factors were applied. This did not mean that the cable size was inadequate, because the ICEA derate factor is correct when all the conductors in the tray continuously carry their maximum allowable current. Since many of the cables in the tray sections in question carry currents much less than maximum allowable, the derate factor was conservative. The licensee's approach to resolving this problem was to use a thermodynamic computer program to estimate actual temperatures of conductors.

The thermodynamic analysis consisted of modeling each tray wrapped in Thermo-Lag and certain non-wrapped tray (e.g. trays 105, 505, 507 and 508 to analyze cable DCL-1). Actual load currents were entered as input data and the most heavily loaded cables were grouped at the center of the tray. This conservative modeling was necessary to deal with cases where the actual location of each cable in the tray under study was not known.

Temperature detectors were installed in certain trays and wired to recorders which also recorded the status (on - off) of the associated loads. In this manner, the actual temperature of the outer surface of a particular cable was determined. Then the actual conductor temperature could be calculated.

The results of this analysis with the conservative modeling was that certain cables in seven trays had calculated conductor temperatures in excess of the rated 90°C. All of these trays contained circuits operating at 480 V or 120 V. There were no ampacity issues with circuits operating at voltages higher than 480 V. For four of the trays, the calculated temperature was only slightly above 90°C. Remaining life calculations showed that the cables in these four trays had a remaining life of from 14 to 27 years. No further analysis was performed on these. For three of the trays (one containing all 480 V power circuits and two containing primarily 120 V control circuits with a few low-level power 480 V), the calculated conductor temperature of certain cables was considerably above 90°C. Remaining life calculations yielded inadequate remaining life. These cables were physically examined and subjected to a non-destructive Indenter Polymer Aging Monitor test. The indentor test was developed by the Electric Power Research Institute (EPRI)(refer to publication EPRI TR-104075). The indentor is a computer controlled test device that presses an anvil against the side of a cable at a constant velocity. The anvil is instrumented to provide position and force measurement. The slope of the force vs position curve is the "Indentor Modulus" which indicates the hardness of the polymer and directly relates to aging due to heat. The examination and the indentor test indicated that the cables had not been overheated in past operation. This indicated that the most heavily loaded cables were evenly distributed rather than tightly grouped as modeled in the computer program. It was also determined that past operation was not less severe than accident scenarios in terms of the currents carried by the cables. LER 97-31-00, Inadequate Cable Sizing Due to Nonconservative De-rating Factors Could Reduce the Cable Remaining Qualified Life, dated September 26, 1997, reported the situation with the later three cable trays.

LER 97-31 also discussed potential problems with the ampacity of the feeder cables for the reactor building cooling air handling fans AHF-1A, 1B and 1C. The problem was that the calculation indicated that the full load current of 170 A was carried by two conductors per phase, but actually this current was carried by one conductor per phase. Contributing to this error was the fact that the fans were two speed type and two cables were in fact run to the motor. Upon further analysis the cable was still acceptable. The correct configuration was probably understood at the time of original plant design. It was the new calculation performed in 1992 that made the error. This error in the calculations did not represent a violation of NRC requirements because it was discovered as part of the corrective action for Violation 97-01-09. The inspector checked the marked-up ampacity calculation sheets for these cables, and verified the as-built configuration of the associated trays and conduits in a walkdown inspection. LER 97-31 was closed.

The thermodynamic analysis and the remaining life calculations were contained in Electrical Cable Operability Evaluation (Cable Ampacity Concerns), Revision 1, dated November 15, 1997.

Ongoing plant modifications were putting Mecatiss on certain trays and conduits. The ampacity considerations associated with adding Mecatiss were addressed in Calculation E-96-0003, Revision 0, dated May 9, 1996. The derate factors applied in reviewing the ampacity of cables in the Mecatiss wrapped raceways were determined in a test conducted by Underwriters Laboratory (UL). A report on this test was submitted to NRC Headquarters for review pursuant to Generic Letter 92-08. This report was reviewed by NRC and its consultant, and a request for additional information was sent to the licensee in a letter dated May 22, 1997. The licensee responded to the request in a letter dated July 3, 1997. The inspector reviewed these two letters and the consultant's report, and concluded that the NRC had no further questions regarding the acceptability of the derate factors. The test also included determination of derate factors for use with Thermo-Lag and Thermo-Lag-Mecatiss combination wrap. As explained above, when the new derate factors for Thermo-Lag were applied to existing installations, certain cables became potential problems with regard to ampacity, and this was addressed in an operability evaluation. The analysis to support recent modifications to install Mecatiss applied correct derate factors, and cables were rerouted as necessary to achieve an installation consistent with the ICEA standards. An example of a cable being re-routed due to the addition of Mecatiss was cable AHC-957, which was removed from tray 643 by MAR 96-01-05-01. The inspector verified that this cable had in fact been re-routed by on-site inspection of the new installation.

The Electrical Cable Operability Evaluation concluded that cables were either acceptable for the life of the plant (i.e. another 20 years) or certain cables (order of magnitude 100 cables) were acceptable for at least one operating cycle. For these cables further analysis would be performed in an attempt to demonstrate longer qualified life. The present operability of these cables was based primarily on examination of the cable, an indentor test and remaining life calculations, which showed 14 years remaining life.

After review of the Case Study (which was really an extent of condition type review) and the operability evaluation performed under Restart Issue D-22, the inspector found that the corrective actions stated in the response to Violation 97-01-09 were satisfactorily performed. Therefore the violation and related LER 97-31 were closed. The licensee was in the process of preparing a justification for continued operation for one fuel cycle in light of the cable ampacity situation described above. Further analysis would be performed to determine a final resolution of the ampacity issue. In order to ensure NRC review of the final or long-term resolution, Inspector Follow up Item IFI 50-302/97-17-03, Review of Cable Ampacity Issue, was established.

c. Conclusions

The licensee performed sufficient review and analysis to define the extent of any potential problems with cable ampacity. A number of problems were permanently resolved. Problems with one set of cables were not permanently resolved, but an operability evaluation supported operability for at least one fuel cycle (about two years). The inspector agreed that the electrical cable operability evaluation had a sound basis. An Inspector Follow up item (IFI 50-302/97-17-03) was established to ensure NRC review of the final or long term resolution of the cable ampacity issue.

The inspector assessed the licensee's performance relative to cable ampacity issue in the five areas of continuing NRC concern.

- Management Oversight - Good
- Engineering Effectiveness Good
- Knowledge of the Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - N/A

E8.18 (Closed) IFI 50-302/97-02-05: Outstanding Issues Associated with the Emergency Diesel Generator Power Upgrade Modification

a. Inspection Scope (92903)

The inspectors reviewed the actions taken to address the issues identified in IFI 50-302/97-02-05.

b. Observations and Findings

Three issues were identified in the referenced IFI regarding the emergency diesel generator power upgrade modification.

Precursor Card 97-0996 was issued on February 28, 1997, to document that a discrepancy was identified with the emergency diesel (EDG) fuel oil volume requirements. The TS requirements for the EDG day tank (each) and underground storage tanks (combined) is 245 gallons and 37,177 gallons, respectively. The licensee had identified that during the revision to Calculation M89-0012, as part of the EDG upgrade modification, a non-conservative assumption was identified in the assumed American Petroleum Institute (API) specific gravity for the fuel oil. During the calculation review cycle, it was not identified that the calculation had impacted the existing TS limits. During the subsequent System Readiness Review, the impact was identified. The licensee's immediate corrective action was to revise SP-354A and SP-354B, the EDG monthly surveillance procedures, to require that a more conservative minimum value was maintained in the fuel oil tanks. The licensee performed a review of tank volumes between the period of January 1986 to March 1997 and identified four instances where an underground tank did not contain the minimum volume, but considering the volume in the day tank and the other tank, the total minimum requirements were met.

To control future volumes, the licensee revised the purchase specification to include a maximum limit for API gravity. SP-746I, Diesel Fuel Oil Testing Surveillance Program: New Diesel Fuel Receipt, has been revised to ensure that new diesel fuel receipts meet the new purchase specification. Technical Specification change request 210 includes proposed changes on the emergency diesel generator, including the new, more restrictive values for EDG fuel oil requirements.

Precursor Card 97-1501 was issued on March 4, 1997, to identify the potential for the EDG exciter to exceed potentially the design limitation of 54 amperes at maximum loading, based on testing performed on February 1, 1997. On March 14 and 15, 1997, further testing was conducted to evaluate the capability of the EDG exciter. The licensee concluded that the original data were inconclusive as the field amperage data were not obtained at the same time as the watt and power factor data. The second set of data was gathered with coordinated communications to allow simultaneous recording. The licensee concluded that the exciter was designed for 54 amperes operation at 3250 KW with a 0.8 power factor and had a nameplate of 3000 KW with a 0.8 power factor. The test data demonstrated that the exciter operated acceptably within the nameplate rating. The calculations based on data gathered resulted in acceptable values, even for the power upgrade. During the modification functional test for EDGD-1B, the design values for exciter amperage was not exceeded. The licensee has closed the PC based on the new calculations. The conclusions reached were supported by the testing following the modifications on the diesels.

Precursor Card 97-0999 was issued on March 4, 1997, to discuss the assessment of data obtained during the February 1, 1997, test which demonstrated that the EDG engine room temperature could potentially exceed design maximum temperature of 120°F, at the maximum loading. Precursor Card 97-3300 was issued on June 23, 1997, to identify concerns with the recirculation of hot radiator air exhaust which directly affects supply air temperature to the EDG room. The basic impact identified was an increase in the effective inlet temperature of up to 15°F. On June 26, 1997, LER 50-302/97-013 was issued to address these same issues.

c. Conclusions

The licensee has addressed the concerns with the fuel oil API Gravity through administrative controls, pending final issuance of a Technical Specification amendment on minimum fuel oil requirements. Testing has dispositioned concerns with the EDG exciter amperage. These items are closed. The third concern, with the EDG elevated temperatures will be addressed in the follow-up to LER 50-302/97-013. IFI 50-302/97-02-05 is closed.

The inspectors assessed the licensee's performance, relative to corrective actions for this issue, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Adequate
- Knowledge of Design Basis - Adequate
- Compliance with Regulations - Adequate
- Operator Performance - N/A

E8.19 (Closed) VIO 50-302/97-11-06: Failure to Follow Licensee Procedure NEP-254

a. Inspection Scope (92903)

This violation involved a determination by the inspectors that the licensee had inappropriately performed modifications to the building spray pump impellers under the Plant Equipment Equivalency Replacement Evaluation (PEERE) process rather than the MAR process, as required by licensee procedure NEP-254, Plant Equipment Equivalency Replacement Evaluation. The licensee responded to the violation in a letter dated October 8, 1997. The inspectors reviewed the licensee's corrective actions as stated in the response.

b. Observations and Findings

The licensee performed a review of completed PEEREs and determined that both PEERE 1497 and PEERE 685 should have been processed as MARs. Both PEEREs were voided, with the work previously accomplished under PEERE 1497 being reassessed and documented under MAR 97-09-08-01. PEERE 685 had not been implemented, so no further work had been performed on it. The inspector reviewed the results of the licensee review and discussed the findings with the licensee. Based on the licensee's review and the results on a review of PEEREs conducted by the licensee, no additional concern was identified with existing PEEREs.

The licensee issued an interoffice correspondence to all engineering and modification personnel on September 11, 1997, describing the issue and providing interim guidance on the correct use of the PEERE process, while clarifications were made to NEP-254. On November 7, 1997, a revision to NEP-254 was issued, which included a clarification for the use of the PEERE process and a checklist to guide the personnel through the decision making.

Training classes were conduced for engineering and modifications personnel on September 16, 1997. The violation was discussed with the personnel and corrective actions were outlined with the personnel. The inspectors reviewed the attendance sheets and noted that the majority of the required personnel were in attendance for the training.

The licensee has scheduled a self assessment conducted by the engineering department to determine the effectiveness of the corrective actions for this issue prior to July 15, 1998.

c. Conclusions

The inspectors have verified that all required corrective actions have been completed. In addition, the licensee has scheduled a self-assessment to assess the effectiveness of the corrective actions after a period of implementing the new process. This violation is closed.

The inspectors assessed the licensee's performance, relative to corrective actions for this violation, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Good
- Knowledge of Design Basis - Adequate
- Compliance with Regulations - Adequate
- Operator Performance - N/A

E8.20 (Closed) IFI 50-302/95-15-05: Relief Valves Removed From Heat Exchangers (92903)

a. Inspection Scope

This follow-up item was identified during the service water system self-assessment inspection. A design change package (MAR 80-04-13-01) had been issued to allow the removal of relief valves from various sets of heat exchangers in the Nuclear Services Closed Cycle Cooling (SW) and Decay Heat Closed Cycle Cooling (DC) systems. The modification was made to address a chronic problem with leakage and because these valves were thought to be redundant. The heat exchangers affected included the reactor building coolers and fan motor coolers; the reactor coolant pump motors, bearings, and seal coolers; and the makeup and purification pump motor coolers. Code USAS B31.1-1967, Section 122.6.1, requires that no intervening stop valves shall separate a protected component from its overpressure protection device. Contrary to this requirement, MAR 80-04-13-01 allowed intervening stop valves between the relief devices and the components being protected.

b. Observation and Findings

To address the issue raised during the service water inspection, MAR 96-10-04-02 was developed to re-install the relief valves that were removed by MAR 80-04-13-01. Also per MAR 96-10-04-02, all NuPro thermal relief valves were to be replaced with Anderson-Greenwood valves because the NuPro valves' relief capacity was derated by the vendor. As part of the process to close IFI 95-15-05, the inspector requested a walkdown with the system engineer to verify that the valves had been replaced/re-installed. During the walkdown on November 19, 1997, a DC system thermal relief valve (DCV-109) was identified as not having been replaced. DCV-109 was a NuPro valve. PC 97-7920 was written to document this discrepancy and assigned a grade level "C."

Following discussions with the project engineer associated with MAR 96-10-4-02, it was determined that a previous failure to update the Configuration Management Information System (CMIS) and a personnel error both contributed to DCV-109 not being replaced. Specifically, CMIS was not updated when the original Texsteam valve was replaced by the NuPro valve in the early 1980s. The project engineer overlooked valve DCV-109 when preparing the work request to replace the NuPro valve with an Anderson-Greenwood valve, and therefore, DCV-109 was not included in MAR 96-10-04-02. This oversight was further perpetuated when the design change package was closed-out and submitted in a restart issue folder for NRC inspection. A Field Change Notice was issued to install an Anderson/Greenwood relief valve and will be completed prior to plant restart.

The inspector determined that the original failure to recognize the requirements and establish the design basis for multiple relief valves to protect components from thermal overpressurization and the omission of a valve identified as required to be replaced and the failure to recognize this omission during closeout of the design change package, constitutes a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. This will be tracked as VIO 50-302/97-17-04, Inadequate Design Control.

c. Conclusion

The inspector assessed the licensee's performance, relative to this violation, in the five areas of continuing NRC concern:

- Management Oversight - Inadequate
- Engineering Effectiveness - Inadequate
- Knowledge of Design Basis - Inadequate
- Compliance with Regulation - Inadequate
- Operator Performance - N/A

E8.21 <u>(Closed) VIO 50-302/96-09-06, Erroneous Calculation Inputs and Inservice Inspection Boundary</u>

<u>(Open) LER 50-302/97-038: Engineering Oversight Resulted in Operation Outside Design Basis of Waste Disposal System</u>

a. <u>Inspection Scope (92903)</u>

This violation involved a failure of the licensee to assure that applicable regulatory requirements and design basis were correctly translated into specifications, drawings, procedures and instruction. The violation involved two examples of calculation errors and an example of an erroneously located Inservice inspection boundary. The inspector reviewed documentation, and interviewed licensee personnel to assess the adequacy of the licensee's corrective actions.

b. <u>Observations and Findings</u>

The inspector noted that resolution of this violation was being tracked under licensee Restart Issue OP-31A. The two parts of this violation were as follows:

- The design input currently used in calculations for safety related battery charger (MAR 93-05-07-01) and 4160/480 volt transformer (MAR 95-08-22-01) replacements was incorrect. The input currently used in the calculations was 56 amperes, whereas the correct value was 62 amperes

- The Inservice inspection class 2/3 makeup system boundary shown on FSAR drawing FD-302-661, Sheet 4, was not moved from Valve MUV-64 to Valve MUV-65 in 1984, when the Engineered Safeguard signal was removed from MUV-64. With that change, MUV-64 could no longer be considered a boundary, as it was open and would not automatically close to provide a boundary.

The inspector reviewed the corrective actions identified in the licensee's letter dated May 20, 1997. In this letter, the licensee committed (at the request of the NRC) to address the seismic classification break problems that had been identified subsequent to the issuance of VIO 96-09-06. The inspector reviewed the affected documentation and interviewed licensee personnel to determine if the corrective actions were implemented and to assess the adequacy of the corrective actions for item 1 and item 2 of the violation. The inspector reviewed sections E8.3 of NRC IR 50-302/97-07 and section E1.1 of NRC IR 50-302/96-03. Following the review of the inspection reports the inspector concluded that the licensee had implemented adequate corrective actions to address the concerns identified in Item 1. Therefore, item #1 of VIO 50-302/96-09-06 is closed.

The inspector also reviewed documentation which addressed concerns associated with Item #2. The inspector reviewed selected drawings, procedures and documentation submitted to the NRC, NRC inspection reports and interviewed licensee personnel to determine if adequate corrective actions had been implemented for those concerns identified in item #2. After reviewing NRC IR 50-302/97-16, the inspector found that additional concerns associated with Item #2 were identified. Specifically, it was concluded in IR 97-16 that, "...there was a weakness in the corrective actions for VIO 96-06-06 in that the extent of condition review did not include all of the WDS tanks and associated piping."

In a supplemental response to VIO 96-06-06, dated October 30, 1997, the licensee provided the results of its extent of condition review of the design basis requirements for seismic and inservice inspection (ISI) classification of piping systems, with focus on the interface requirements which govern the transition between class boundaries. In the licensee's October 1997 response, the discrepancies identified by the NRC in IR 97-16 were addressed. The letter also addressed discrepancies identified from the licensee's own extent of condition effort. In the letter dated October 30, 1997, the licensee stated that the result of their review effort and any corrective actions will be reported to the NRC under LER 97-038-000. The inspector reviewed the subject LER to determine what additional corrective actions had been identified, and if the corrective actions had been implemented. Following the review of the LER the inspector determined that the licensee was continuing its effort to address all discrepancies and implement the necessary corrective actions. In LER 97-038-000, the licensee made the following commitments to resolve the issues.

- The liquid waste outlet piping for the Waste Gas Decay Tanks, Miscellaneous Waste Storage Tank, Spent Resin Storage Tank, and Neutralizer Tank will be upgraded to Seismic Class 1 - Refueling Outage (11R).

- Reactor Coolant Drain Tank Liquid waste outlet piping is being evaluated by FPC, in accordance with 10 CFR 50.59, as a change from seismic to non-seismic. The FSAR will be amended as appropriate - Refueling Outage (11R).

- A Justification for Continued Operation for the WD system is being Developed by FPC, consistent with Generic Letter 91-18, Revision 1. Prior to MODE 4.

Based on the corrective actions presently completed by the licensee and the commitment made in the LER, the inspector concluded that the licensee's actions addressing #2 of VIO 50-302/96-09-06 are acceptable for restart. The NRC will track the implementation of the remaining corrective actions under LER 97-038-00. Although this item is a noncompliance with regulatory requirements, for the reasons discussed in

Inspection Report 97-21, the licensee met the criteria for enforcement discretion per Section VII.B.2 of the NRC Enforcement Policy as described in NUREG-1600. Consequently this item is closed and is identified as another example of Non-cited Violation NCV 50-302/97-21-01, Examples of Noncompliances in Design Control, 10 CFR 50.59 Evaluations, Procedure Adequacy/Adherence, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

c. Conclusions

The inspector concluded that the licensee's corrective actions for this violation were adequate. However, it should be noted that the licensee implemented several of the corrective actions only after additional discrepancies were identified by the NRC. The inspector concluded that, even though several of the corrective actions' implementation are not presently completed, the time table established by the licensee for completion of the actions is acceptable. Therefore, this violation is closed.

The inspector assessed the licensee's performance, relative to the corrective actions for this violation, in the five areas of continuing NRC concern:

- Management Oversight - Adequate
- Engineering Effectiveness - Adequate
- Knowledge of the Design Basis - Adequate
- Compliance with Regulations - Adequate
- Operator Performance - N/A

E8.22 (Closed) EA 95-16: Use of Nonconservative Trip Setpoints for Safety-Related Equipment

(Closed) LER 50-302/94-006-00 through LER 50-302/94-006-06: Deficiency in Understanding of Technical Requirements Leads to Nonconservative Safety Systems Setpoint and Violations of Improved Technical Specifications

a. Inspection Scope (92903, 37550)

As part of the continuing review of corrective actions for EA 95-16, the inspectors reviewed several new instrument loop uncertainty setpoint calculations and LER 50-302/94-006, Deficiency in Understanding of Technical Requirements Leads to Nonconservative Safety Systems Setpoint and Violations of Improved Technical Specifications, Revisions 0 through 6. In IR 50-302/95-06 the inspectors found that some safety-related trip setpoint calculations did not follow the methodology specified in Instrument Society of America (ISA) 67.04, Part II, as referenced by instrumentation and controls Design Criteria Instrument String Error and Setpoint Determination Methodology. To assess the progress the licensee had made in this area, the inspector reviewed a sample of the most recent instrument string error/setpoints. Various discrepancies with the instrument loop uncertainty calculations were reported in LER 50-302/94-006, Revision 0 through 6. The inspectors reviewed the licensee's actions associated with this LER and it's subsequent revisions.

b. Observations and Findings

The inspector reviewed a representative sample of instrument loop uncertainty setpoint calculations. These calculations were well documented, with well founded assumptions, and followed the methodology specified in ISA 67.04, Part II, a referenced by instrument and controls Design Criteria Instrument String Error/Setpoint Determination Methodology. Through a field walkdown, the inspector verified the instrumentation installed in the field was appropriately included in the setpoint calculations.

However, the licensee had not completed the field installation for some of the setpoint loop uncertainty calculations. In addition the licensee had not completed the Alarm Response Procedures, Surveillance Procedures, or Operating Procedures for several setpoint loop uncertainty calculations. These included:

- Setpoint Calculation I88-0022, RC (T Hot) Temperature Loop Accuracy, RC-4A-TE1, RC-4B-TE4, Rev. 6

    - Procedures requiring revision for this calculation:

      SP-161A, Reactor Coolant T_{hot} and T_{cold} Calibration, Rev. 19

- Setpoint Calculation I90-0019, Reactor Bldg. Pressure Loop Accuracy (BS-16/17), Rev. 1

    - Procedures requiring revision for this calculation:

      SP-162, Post Accident Monitoring Instrumentation Channel Calibration, Rev. 33 (See I95-0017 for Procedures Changed to use Narrow Range RB Pressure)

- Setpoint Calculation I91-0012, BWST Level Accuracy, Rev. 3

    - Procedures requiring revision for this calculation:

      AR-303, Esc Annunciator Response, Rev. 26

      SP-162, Post Accident Monitoring Instrumentation Channel Calibration, Rev. 33

      SP-300, Operating Daily Surveillance Log, Rev. 139

SP-301, Shutdown Daily Surveillance Log, Rev. 104

OP-103, Tank Volumes, Rev. 9

- Setpoint Calculation I91-0021, RC Flow Loop (NNI) Accuracy, Rev. 1

  - Procedures requiring revision for this calculation:

    AR-502, ICS J Annunciator Response, Rev. 10

    SP-112, Calibration of the Reactor Protection System, Rev. 57

    SP-162, Post Accident Monitoring Instrumentation Channel Calibration, Rev. 33

    SP-300, Operating Daily Surveillance Log, Rev. 139

- Setpoint Calculation I95-0017, Reactor Building Narrow Range Pressure, Rev. 0

  - Procedures requiring revision for this calculation:

    SP-135A, Engineered Safeguards Actuation Channel 1 System Response Time Test, Rev. 12

    SP-135B, Engineered Safeguards Actuation Channel 2 System Response Time Test, Rev. 15

    SP-135C, Engineered Safeguards Actuation Channel 3 System Response Time Test, Rev. 13

    SP-162, Post Accident Monitoring Instrumentation Channel Calibration, Rev. 33

    SP-300, Operating Daily Surveillance Log, Rev. 139

    SP-347, ECCS and Boration Flow Paths, Rev. 45

    SP-356, ES Manual Actuation Channel Functional Test for RB Isolation and Cooling, Rev. 20

    SP-357, ES Manual Actuation Channel Functional Test for High Pressure Injection and Low Pressure Injection, Rev. 19

    SP-456, Refueling Interval Equipment Response to an ESAS Test Signal, Rev. 18

- Setpoint Calculation I85-0004, Dedicated EFW Tank Alarm Settings, Rev. 5

  - Procedures requiring revision for this calculation:

    AR-403, PSA H Annunciator Response, Rev. 29

    OP-103F, Tank Volumes, Rev. 9

    OP-450, Emergency Feedwater System, Rev. 20

    SP-162, Post Accident Monitoring Instrumentation Channel Calibration, Rev. 33

    SP-300, Operating Daily Surveillance Log, Rev. 139

    SP-338, Remote Shutdown and Post Accident Monitoring Channel Check, Rev. 27

- Setpoint Calculation I87-0003, EFW Flow Control & Interlock, Rev. 5

  - Procedures requiring revision for this calculation:

    SP-193B, EFW Flow Transmitter Channel Calibration, Rev. 2

c. Conclusions

The inspectors concluded that the licensee continued to make progress in resolving the Improved Technical Specifications (ITS) setpoint program deficiencies. The calculations reviewed were well documented, with well-founded assumptions, and followed the methodology specified in ISA 67.04, part II, as referenced by instrumentation and controls Design Criteria Instrument String Error/Setpoint Determination Methodology.

The licensee's actions were adequate to close the open items. The items that are not complete are scheduled to be completed prior to entry into Mode 4. The completion of these items will be tracked as IFI 50-302/97-17-05, Resolution of Improved Technical Specification Setpoint Program Deficiencies Prior to Entry Into Mode 4.

The inspectors assessed the licensee's performance relative to lack of design control for assumptions in instrument setpoint calculations, in the five areas of continuing NRC concern:

- Management Oversight - Good
- Engineering Effectiveness - Superior
- Knowledge of the Design Basis - Superior
- Compliance with Regulations - Superior
- Operator Performance - N/A

IV. Plant Support

P8 Miscellaneous EP Issues

P8.1 (Closed) IFI 50-302/97-08-03: Variations in the classification and interpretation of the Emergency Action Levels (EALs) by the Site Emergency Coordinators (SECs) (92904)

a. Inspection Scope

The inspector evaluated applicable areas of Florida Power Corporation's September 4, 1997 response to IFI 50-302/97-08-03. In the response, the licensee committed to:

- develop an EAL Interpretation Guide,
- conduct classroom training, focusing on a review of the EALs, with all Emergency Coordinators,
- conduct semi-annual EAL/Protective Action Recommendation exercises for each SEC to ensure consistent application, and
- target April 1998 as the date for submitting it's NUMARC EALs to the NRC

b. Observations and Findings

As part of the EAL and initial EAL Interpretation Guide training, the licensee developed a questionnaire consisting of 19 scenarios. The questionnaire was given to the SECs during their classroom training to evaluate the SEC's ability to use the EALs and the guide to classify scenarios. The inspectors reviewed the questionnaire and noted that the scenarios were similar to the scenarios used by the inspectors during IR 50-302/97-08, but the licensee's scenarios were more diverse and challenging. The licensee stated that during the EAL and initial EAL Interpretation Guide training, the questionnaire facilitated active discussions between the SECs and the class instructor. These discussions produced a greatly improved EAL Interpretation Guide over the licensee's initial Interpretation Guide. Also in the revision, the licensee had annotated the applicable EAL with a number that corresponded to the numbered interpretation in the EAL Interpretation Guide.

The inspectors verified that all 25 SECs had received classroom training on the EAL Interpretation Guide and a focused review of the EALs, by verifying a SEC roster against the training attendance sheets.

To assess independently the effectiveness of the EAL Interpretation Guide and SEC training, the inspector interviewed SECs from the training roster. The inspector asked the SECs to classify sixteen inspector prepared scenarios. The interviewees used the EALs and EAL Interpretation Guide to accurately classify the scenario.

Because of the SECs training in October 1997, the next semi-annual EAL/Protective Action Recommendation exercises for each SEC would not be required until April 1998.

The inspectors noted that additional clarification could improve some EALs. These EALs were discussed with the licensee during the inspection and during the exit meeting. The licensee's management acknowledged and took note of the inspector's comments. The issues discussed were as follows:

- The lack of supporting documentation to provide a basis for "Containment Gross Gamma monitor reading exceeding limits" of greater than 1000 R/hr was to be classified as a Site Area Emergency (SAE) and 10,000 R/hr for a General Emergency (GE).

- In the Site Area Emergency EAL for "All Alarms Lost", the word "Transient" was defined. The definition of "Transient" in the latest revision of the "EAL Interpretation Guide" did not include "greater than 10 percent thermal power oscillation" as a "Transient".

- The titles "Safety Related Equipment" and "Safe Shutdown Equipment" resulted in an apparent inconsistency in two EALs. The title "Safety Related Equipment" was used in the EAL for "Fire within the Protected Area", and the title, "Safe Shutdown Equipment" was used in the EAL for "Missile Impact". If a fire damaged a Containment Spray Pump, it would be classified as a SAE. If a missile (nozzle broken off of a nitrogen bottle) damaged a Containment Spray Pump, it would not be classified as an emergency.

- The EAL for "Loss of Main and Emergency Feedwater" did not include "Auxiliary Feedwater". The EAL had not been modified since the licensee had determined that Auxiliary Feedwater (turbine driven) was not a "Safety Related System", and would be considered a separate non-safety related independent feedwater system.

- The word "Imminent," as defined in the "EAL Interpretation Guide," did not clarify the intent of the word.

- The SAE and GE classification criteria for a "Steam Generator tube leak" and a "Steam Generator tube rupture with a loss of offsite power" were the same.

c. Conclusion

The inspectors concluded that the revised guide was considerably more comprehensive and more precise than the initial guide and adequately addressed the inspectors' concerns in IFI 50-302/97-08-03.

- Management Oversight - Good
- Engineering Effectiveness - N/A
- Knowledge of Design Basis - Good
- Compliance with Regulations - Good
- Operator Performance - N/A

V. Management Meetings

X1 Exit Meeting Summary

The inspection scope and findings were summarized on December 1, 1997. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

X3 Management Meeting Summary

X3.1 A meeting was held on October 30, 1997 at the FPC Training Center to discuss Engineering issues. A separate meeting summary was issued on November 21, 1997.

X3.2 A Public Meeting was held on October 31, 1997 at the FPC Training Center to discuss the licensee's progress on readiness for restart. A separate meeting summary was issued on November 21, 1997.

PARTIAL LIST OF PERSONS CONTACTED

Licensees

R. Anderson, Senior Vice President, Nuclear Operations
J. Baumstark, Director, Quality Programs
J. Cowan, Vice President, Nuclear Production
R. Davis, Assistant Plant Director, Operations and Chemistry
R. Grazio, Director, Nuclear Regulatory Affairs
G. Halnon, Assistant Plant Director, Nuclear Safety
B. Hickle, Director, Restart
J. Holden, Site Director
D. Kunsemiller, Manager, Nuclear Licensing
M. Marano, Director, Nuclear Site & Business Support
C. Pardee, Director, Nuclear Plant Operations
W. Pike, Manager, Nuclear Regulatory Compliance
M. Rencheck, Director, Nuclear Engineering
M. Schiavoni, Assistant Plant Director, Maintenance
T. Taylor, Director, Nuclear Operations Training

NRC

H. Christensen, Engineering Branch Chief, Region II (October 30 through 31, 1997)
S. Collins, Director, NRR (October 31, 1997)
P. Fillion, Reactor Inspector, Region II (October 27 through November 5, November 17 through 21, 1997)
F. Hebdon, Directorate II-3, NRR (October 30 through 31, 1997)
J. Jaudon, Director, Division of Reactor Safety, Region II (October 30 through 31, 1997)
J. Johnson, Director, Division of Reactor Projects, Region II (October 30 through 31, 1997)
T. Johnson, Senior Resident Inspector, Turkey Point (November 3 through 7, 1997)
C. Julian, Technical Assistant, Region II (November 18 through 21, 1997)
K. Landis, Branch Chief, Region II (October 30 through 31, 1997)
E. Lea, Project Engineer, Region II (November 17 through 21, 1997)
L. Mellen, Reactor Engineer, Region II (November 23 through 25, 1997)
M. Miller, Reactor Inspector, Region II (October 27 through 31, November 17 through 21, 1997)
W. Miller, Reactor Inspector, Region II (November 17 through 21, 1997)
S. Ninh, Project Engineer, Region II (October 30 through 31, 1997)
L. Raghavan, Project Manager, NRR (October 30 through 31, 1997)
L. Reyes, Regional Administrator, Region II (October 30 through 31, 1997)
R. Reyes, Resident Inspector, Region II (November 10 through 14, 1997)
G. Salyers, Emergency Preparedness Specialist, Region II (November 10, 1997)
R. Schin, Reactor Inspector, Region II (November 3 through 7, November 17 through 21, 1997)
M. Thomas, Reactor Inspector, Region II (November 3 through 7, 1997)
G. Wiseman, Reactor Inspector, Region II (November 17 through 21, 1997)

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 Observations
IP 62707:	Conduct of Maintenance
IP 71707:	Plant Operations
IP 92901:	Followup - Operations
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
VIO	50-302/97-17-01	Open	Failure to Conduct an Adequate Unreviewed Safety Question Evaluation for a Modification Functional Test. (Section E2.1)
IFI	50-302/97-17-03	Open	Review of Cable Ampacity Issue. (Section E8.17)
VIO	50-302/97-17-04	Open	Inadequate Design Control. (Section E8.20)
IFI	50-302/97-17-05	Open	Resolution of Improved Technical Specification Setpoint Program Deficiencies Prior to Entry Into Mode 4. (Section E8.22)

Closed

Type	Item Number	Status	Description and Reference
VIO	50-302/97-02-01	Closed	Failure to Follow Equipment Control Procedure Requirements. (Section 08.1)
LER	50-302/96-21-00	Closed	Delayed Entry Into Technical Specification Required Action Caused by Inadequate Documentation of Out-of Service Equipment Requirements for a Modification. (Section 08.2)
VIO	EA 97-094 01013, 01023, 01033, 01043	Closed	Repeat Failure to Make Timely Reports to the NRC. (Section 08.3)
VIO	50-302/97-08-01	Closed	Inadequate Corrective Action and Procedure for External Reporting Requirements. (Section 08.3)
LER	50-302/97-002-01	Closed	Out of Calibration Fuel Pool Water Level Transmitters. (Section M8.1)

VIO	50-302/97-01-04	Closed	Failure to Perform Technical Specification Surveillance for Spent Fuel Level. (Section M8.1)
NCV	50-302/97-17-02	Closed	Maintenance Performed on Safety Related Components Without Approved Procedures or Work Instructions. (Section E2.1)
VIO	50-302/96-08-01	Closed	Failure to Take Timely Corrective Action to Address Issues and Actions For Makeup System Audit Findings and Excessive Vibration on a Spent Fuel Pool Pump Fan Motor. (Section E8.1)
LER	50-302/96-011-00	Closed	Personnel Error Causes Testing Deficiency Resulting in Condition Prohibited by Improved TS. (Section E8.2)
LER	50-302/96-025-00	Closed	Personnel Error Causes Testing Deficiency Resulting in Condition Prohibited by TS. (Section E8.2)
LER	50-302/97-003-00- 50-302/97-003-05	Closed	Personnel Errors Caused Testing Deficiencies (GL 96-01). (Section E8.2)
VIO	50-302/97-05-03	Closed	Incorrect Information in Annunciator Response Procedure for Inverters. (Section E8.3)
VIO	50-302/97-07-01	Closed	Failure to Follow Procedure CP-111 for the Processing of Precursor Cards. (Section E8.4)
URI	50-302/96-201-07	Closed	EDG not Protected Against Water Spray from the Fire Protection System Sprinkler. (Section E8.5)
VIO	EA 95-126, I.C.2 (04013)	Closed	Corrective Actions for an Inadequate Curve 8 (Two STI's and a Revised Curve 8A and 8B) were Also Incorrect. (Section E8.6)

VIO	EA 96-365, C (03013)	Closed	Inadequate Corrective Actions for 10 CFR 50.59 Evaluation Errors and for Inadequate Containment Penetration Surveillances. (Section E8.7)
VIO	EA 97-162 (01013)	Closed	Inadequate Safety Evaluations for Added Operator Actions for Design Basis SBLOCA Mitigation. (Section E8.8)
LER	50-302/96-24-01	Closed	Plant Modification Causes Unanalyzed Condition Regarding Emergency Feedwater. (Section E8.10)
EA Ex. 1	96-365, 96-465 96-527, VIO B (02013)	Closed	Failure to Update Applicable Design Documents to Incorporate Design Information (Section E8.11)
EA	96-365, 96-465 96-527, VIO B Ex. 2 (02013)	Closed	Failure to Include Applicable Design Information in the Design Input Requirements for a Modification (Section E8.12)
LER	50-302/97-017-00	Closed	Personnel Error Caused Inadequate Electrical Separation Of High Pressure Flow Indicators. (Section E8.14)
VIO	EA 95-126 NOV II.B	Closed	Failure to Take Adequate Corrective Action for Required Tank Volumes, Level, and Suction Points. (Section E8.15)
VIO	50-302/97-01-09	Closed	Inadequate Corrective Action for Cable Ampacity. (Section E8.17)
LER	50-302/97-31-00	Closed	Inadequate Cable Sizing Due to Nonconservative De-rating Factors Could Reduce the Cable Remaining Qualified Life. (Section E8.17)
IFI	50-302/97-02-05	Closed	Outstanding Issues Associated with the Emergency Diesel Generator Power Upgrade Modification. (Section E8.18)

VIO	50-302/97-11-06	Closed	Failure to Follow Licensee Procedure NEP-254. (Section E8.19)
IFI	50-302/95-15-05	Closed	Relief Valves Removed From Heat Exchangers. (Section E8.20)
VIO	50-302/96-09-06	Closed	Erroneous Calculation Inputs and Inservice Inspection Boundary. (Section E8.21)
VIO	EA 95-16	Closed	Use of Nonconservative Trip Setpoints for Safety-Related Equipment. (Section E8.22)
LER	50-302/94-006-00-50-302/94-006-06	Closed	Deficiency in Understanding of Technical Requirements Leads to Nonconservative Safety Systems Setpoint and Violations of Improved Technical Specifications. (Section E8.22)
IFI	50-302/97-08-03	Closed	Variations in the classification and interpretation of the EALs by the Emergency Coordinators. (Section P8.1)

Discussed

Type	Item Number	Status	Description and Reference
VIO	EA 97-330 (01013)	Open	Unreviewed Safety Question Involving Added EDG Protective Trips. (Section E8.9)
IFI	50-302/97-11-04	Open	Corrective Actions for Approximately 4000 Precursor Cards Not Tracked to Completion. (Section E8.7)
URI	50-302/95-02-02	Open	Control Room Habitability Envelope Leakage. (Section E8.16)

LER	50-302/97-038-00	Open	Engineering Oversight Resulted in Operation Outside Design Basis of Waste Disposal System. (Section E8.21)

LIST OF ACRONYMS USED

AI - Administrative Instruction
AP - Abnormal Procedures
API - American Petroleum Institute
AR - Annunciator Response
BAST - Boric Acid Storage Tank
BL - (NRC) Bulletin
BS - Building Spray
BWST - Borated Water Storage Tank
CCHE - Control Complex Habitability Envelope
CFM - Cubic Feet per Minute
CFR - Code of Federal Regulations
CFT - Core Flood Tank
CMIS - Configuration Management Information System
CP - Compliance Procedure
CREVS - Control Room Emergency Ventilation System
CR3 - Crystal River Unit 3
CT - Current Transformers
DBD - Design Basis Document
DH - Decay Heat
DHP - Decay Heat Pump
DHV - Decay Heat Valve
DNPO - Director, Nuclear Plant Operations
ECCS - Emergency Core Cooling System
EDBD - Enhanced Design Basis Document
EDG - Emergency Diesel Generator
EFIC - Emergency Feedwater Initiation and Control
EFW - Emergency Feedwater
EOP - Emergency Operating Procedure
EPRI - Electric Power Research Institute
ES - Engineered Safeguards
ESQPM - Environmental and Seismic Qualification Program Manual
FCN - Field Change Notice
FLA - Full Load Amperes
FME - Foreign Material Exclusion
FMEA - Failure Modes and Effects Analysis
FPC - Florida Power Corporation
FSAR - Final Safety Analysis Report
FSP - Fire Service Pump
FT - Functional Test
GL - Generic Letter
HPI - High Pressure Injection
HVAC - Heating Ventilation and Air Conditioning
I&C - Instrumentation and Control

IFI - Inspection Followup Item
IOC - Interoffice Correspondence
IP - Inspection Procedure
IPAP - Integrated Performance Assessment Process
IR - Inspection Report
ISA - Instrument Society of America
ISI - Inservice Inspection
ITS - Improved Technical Specifications
JCO - Justification for Continued Operation
kw - Kilowatt
LCO - Limiting Condition for Operation
LER - Licensee Event Report
LOCA - Loss of Coolant Accident
LOOP - Loss of Offsite Power
LPI - Low Pressure Injection
LPMS - Loose Parts Monitoring System
MAR - Modification Approval Record
MCAP - Management Corrective Action Plan
MCC - Motor Control Center
MP - Maintenance Procedure
MSLB - Main Steamline Break
MUT - Makeup Tank
MUV - Makeup Valve
NCV - Non-cited Violation
NEP - Nuclear Engineering Procedure
NOD - Nuclear Operations Directive
NOE - Nuclear Operations and Engineering
NOV - Notice of Violation
NPSH - Net Positive Suction Head
NP&SM - Nuclear Procurement and Storage Manual
NQA - Nuclear Quality Assessments
NRC - Nuclear Regulatory Commission
NRR - Office of Nuclear Reactor Regulation
NSM - Nuclear Shift Manager
OCR - Operability Concerns Resolution
OI - Operating Instruction
OJT - On The Job Training
OP - Operating Procedure
OSB - Operations Study Book
OSTI - Operational Safety Team Inspection
OTSG - Once Through Steam Generator
PC - Precursor Card
PEERE - Plant Equipment Equivalency Replacement Evaluation
PM - Preventive Maintenance
PMRG - Plant Modification Review Group
PMT - Post Maintenance Test
PORV - Power Operated Relief Valve
PR - Problem Report
PRC - Plant Review Committee
PSIG - Pounds Per Square Inch Gauge

PT - Pressure Transmitters
QPS - Quality Programs Surveillance
RCA - Radiologically Controlled Area
RCBT - Reactor Coolant Bleed Tanks
RCP - Reactor Coolant Pump
RCS - Reactor Coolant System
REA - Request for Engineering Assistance
RG - Regulatory Guide
RO - Reactor Operator
RPS - Reactor Protection System
RRT - Rapid Response Team
RTD - Resistor Temperature Detector
RW - Raw Water
SA - Safety Assessment
SAR - Safety Analysis Report
SASS - Smart Analog Select System
SBLOCA - Small Break Loss of Coolant Accident
SBU - Strategic Business Unit
SD - Site Drain System
SDBI - Suspected Design Basis Issue
SEC - Site Emergency Coordinators
SEL - Security Event Log
SER - Safety Evaluation Report
SFP - Spent Fuel Pump
SIR - Security Information Reports
SLUR - Second Level Undervoltage Relays
SM - Shift Manager
SP - Surveillance Procedure
SPDS - Safety Parameter Display System
SR - Surveillance Requirement
SRO - Senior Reactor Operator
SRP - Standard Review Plan
SSOD - Shift Supervisor on Duty
STI - Short Term Instruction
TC - Temporary Change
TDBD - Topical Design Basis Document
TMAR - Temporary Modification Approval Record
TS - Technical Specifications
UFSAR - Updated Final Safety Analysis Report
UL - Underwriters Laboratory
URI - Unresolved Item
USQD - Unreviewed Safety Question Determination
VIO - Violation
WCC - Work Control Center
WG - Water Gauge
WI - Work Instructions
WR - Work Request

Summary of EGDG-1B Test Runs

Type of Run	Start Date/Time	Stop Date/Time	Parameter(s) of Interest
Unloaded maintenance run per MP-499	11/9/97 2:18 am	11/9/97 2:19 am	Stopped due to fan drive clutch slippage
Unloaded maintenance run per MP-499	11/9/97 3:00 am	11/9/97 3:07 am	Stopped due to fan drive clutch slippage
Unloaded maintenance run per MP-499	11/9/97 3:22 am	11/9/97 3:36 am	Stopped due to fan drive clutch slippage
Unloaded maintenance run per MP-499	11/9/97 3:45 am	11/9/97 3:55 am	Stopped due to jacket coolant leak
Unloaded maintenance run per MP-499	11/10/97 5:35 am	11/10/97 6:06 am	Overspeed trip testing
Unloaded maintenance run per MP-499	11/10/97 5:04 pm	11/10/97 5:43 pm	Secured due to high vibrations
Unloaded run per MP-531 to troubleshoot vibration problems	11/12/97 4:15 am	11/12/97 4:33 am	Run with fan clutch disengaged. Secured when high temperature alarms received.
Slow start. Loaded 2625 - 2825 kw for approximately 2 - 3 hours	11/12/97 6:04 pm	11/12/97 10:24 pm	Breaker 3210 closed from 6:31 pm until 10:20 pm.
Slow start. Loaded 2625 - 2825 kw for approximately 2 - 3 hours	11/13/97 6:39 am	11/13/97 11:09 am	Breaker 3210 closed from 7:08 am until 11:04 am. Second slow start with 2 - 3 hour loaded run, following adjustment of fan blade pitch.
Slow start Loaded 2625 - 2825 kw for 24 hours	11/15/97 1:42 pm	11/15/97 2:28 pm	Breaker 3210 closed from 2:10 pm until 2:28 pm. Breaker opened and diesel secured due to fan drive clutch slippage. Loaded only to 1680 kw.
Slow start Loaded 2625 - 2825 kw for 24 hours	11/15/97 5:33 pm	11/16/97 10:16 pm	Breaker 3210 closed from 5:56 pm on 11/15/97 until 10:12 pm on 11/16/97.
Fast start Loaded 2625 - 2825 kw for 4 hours	11/17/97 4:35 am	11/17/97 11:00 am	Breaker 3210 closed from 5:05 am until 10:55 am.

Attachment

Type of Run	Start Date/Time	Stop Date/Time	Parameter(s) of Interest
Slow start Loaded 2625 - 2825 kw for 24 hours	11/18/97 3:17 am	11/19/97 8:22 pm	Breaker 3210 closed from 3:37 am on 11/18/97 until 8:19 pm on 11/19/97. Multiple attempts to raise load above 2825 kw resulted in high vibration conditions in the pedestal bearing.
Slow start Loaded 3100 - 3138 kw for 14 hours	11/20/97 1:41 am	11/21/97 5:23 am	Breaker 3210 closed from 2:01 am on 11/20/97 until 5:21 am on 11/21/97.
Slow start Loaded 3300 - 3375 kw 20 hours at 3300 - 3375 kw 2 hours at 3325 - 3375 kw	11/22/97 10:38 pm	11/23/97 10:37 pm	Breaker 3210 closed from 11:01 pm on 11/22/97 until 10:34 pm on 11/23/97.
Hot condition fast start Loaded 2625 - 2825 kw for 1 hour	11/23/97 10:42 pm	11/24/97 12:55 am	Breaker 3210 closed from 10:51 pm on 11/23/97 until 12:48 am on 11/24/97.
Fast start Loaded 2625 - 2825 kw for 1 hour	11/24/97 4:51 am	11/24/97 6:40 am	Breaker 3210 closed from 5:01 am until 6:36 am.
Maintenance run unloaded	11/27/97 11:46 am	11/27/97 12:08 pm	Run to support performance of PM-123, Periodic Electrical Checks of Emergency Diesel Generators
SP-354B Operability Run >2625 kw	11/29/97 10:25 am	11/29/97 3:49 pm	Breaker 3210 closed from 10:52 am until 3:43 pm. EGDG-1B declared Operable 5:21 am on 11/30/97.

Attachment