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:	November 30, 1997 through January 3, 1998
Inspectors:	S. Cahill, Senior Resident Inspector T. Cooper, Resident Inspector S. Sanchez, Resident Inspector P. Fillion Reactor Inspector, section E8.23 C. Julian, Technical Assistant, section E8.29 E. Lea, Project Engineer, sections E8.24 - E8.26 M. Miller, Reactor Inspector, sections E8.6 - E8.13 R. Schin, Reactor Inspector, sections O7.1, O8.1, E1.2 - E1.4, E8.14 - E8.18 L. Stratton, Physical Security Specialist, sections S1.1, S2.1, S8.1 M. Thomas, Reactor Inspector, sections E8.19 - E8.22 J. York, Reactor Inspector, sections E8.1 - E8.5
Approved by:	K. Landis, Chief, Projects Branch 3 Division of Reactor Projects

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 results of announced inspections of open restart and security items by regional inspectors.

Operations

The licensee was effectively tracking requirements prior to Plant Mode ascension at their Mode hold point meetings. Personnel attending the meeting were well prepared, appropriate topics were discussed, and the licensee was effectively reviewing program areas prior to mode ascension (Section 01.1).

The inspectors observed an excellent example of initiative and questioning attitude by a licensed operator who voluntarily reviewed a test procedure and identified several valid concerns (Section 01.1).

The inspector verified that the licensee had appropriately incorporated administrative controls for the Low Temperature Overpressure Protection Technical Specifications License Amendment 161 (Section 03.1).

As a result of a significant realignment of the shift organization, the senior manager on shift is now a licensed senior reactor operator. This was considered an improvement in the Operations organization and another example of licensee management's effort to define areas of responsibility and ownership clearly (Section 06.1).

The inspectors concluded that the licensee's progress to date on the Management Corrective Action Plan (MCAP II) was satisfactory for plant restart (Section 07.1).

A Non-Cited Violation (NCV 50-302/97-19-01) was identified for inappropriate use of immediate work copy changes to procedures. Immediate corrective actions were taken by the licensee to address the identified weaknesses. The licensee was taking long term corrective actions by proceeding with a procedure upgrade program (Section 07.2).

The inspector concluded that the licensee's quality assurance auditors continued to have a positive impact on the safety and operation of the plant and their observations and conclusions were consistent with inspector observations and conclusions (Section 07.3).

One precursor card reportability determination was found to be poor in that operability requirements of the affected component were not initially recognized by Operations (Section 07.4).

Licensee management continued to exercise oversight of precursor card screening results and developed appropriate barriers to correct inappropriate rating decisions. They have also taken appropriate preliminary actions to address problems with supervisory reviews of precursor cards (Section 07.4).

The inspectors concluded that the approximately 4000 Precursor Cards (PCs) from late 1996, and early 1997, which had been closed without tracking completion of corrective actions, had been adequately reviewed for potential restart items (Section 08.1).

The inspector observed procedure revision inefficiencies in that enhancement comments posted against procedures were often left unresolved when a procedure was revised. The licensee has determined that operational procedure weaknesses exist in all departments and has initiated plans for a procedure upgrade program (Section 08.3).

Maintenance

Although the licensee was appropriately upgrading their foreign material exclusion (FME) program in response to earlier problems, the resolution of the specific FME problem that resulted in blockage of a raw water pump was extremely poor. Specific appropriate corrective actions were not developed, apparent cause evaluations were cursory efforts, and an extent of condition review for other potential blockages was not performed (Section M1.2).

Engineering

The licensee adequately diagnosed and corrected an emergency feed pump air entrainment problem. The inspector determined that system engineering performed well in supporting troubleshooting of this problem (Section E1.1).

The inspectors concluded that the installation of a modification (Auto-Close Makeup Valve-27) was adequate to close out the design issue related to High Pressure Injection system modifications to improve Small Break Loss of Coolant Accident Margins (Section E1.2).

The inspectors concluded that the installation, testing, and 50.59 safety evaluations for the three modifications related to Emergency Feedwater System Upgrades and Diesel Generator Load Impact were adequate to close out the associated restart design issue (Section E1.3).

The inspectors concluded that the licensee's actions to develop leakage tests for the normal make-up control valve and high pressure injection throttle and isolation valves were appropriate (Section E1.4).

The inspector concluded the licensee had properly addressed and repaired degraded containment coating areas and updated Quality Control inspection techniques to ASME Section XI requirements. The licensee also took corrective actions to eliminate a program weakness (Section E8.1).

Several Licensee Event Report issues were found to be adequately resolved and corrected by the licensee and were identified as further examples of Non-cited Violation NCV 50-302/97-21-01, Examples of Noncompliances in Design Control, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion (Sections E8.2, E8.3, E8.9, E8.10, E8.11, E8.12, E8.17, E8.18, E8.25, E8.32, E8.33, E8.34, E8.36, E8.37 and E8.38).

The inspector concluded that all of the licensee's corrective actions for an open violation on an error in service water heat load calculations had not been completed. This item remains open, but the licensee's completed actions were acceptable for restart (Section E8.4).

A violation for use of unverified calculations in modifications was closed based on the licensee's efforts to review calculations and upgrade procedural controls (Section E8.6)

The inspectors concluded there were no safety concerns with the licensee's justification for continued operations with Decay Heat Valves (DHV)-34 and DHV-35 maintained closed during normal operations (Section E8.13).

The inspectors reviewed the licensee's completed modification package and completed post-modification test for the B Emergency Diesel Generator trip circuit, which appropriately tested the modified circuits. The licensee had a restart item to ensure that the modification was installed and tested on the A Emergency Diesel Generator prior to plant restart (Section E8.14).

As a result of calculation inadequacies, procedure inadequacies, and lack of past Control Complex Habitability Envelope (CCHE) leakage testing, the CCHE and Control Room Emergency Ventilation System (CREVS) had not been operable in the past. Nine violations of requirements were identified as more examples of NCV 50-302/97-21-01, Examples of Noncompliances in Design Control, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion (Section E8.18).

The inspector concluded that all the corrective actions specified for Violation 50-302/97-02-03, Adequacy of Procedures to Take the Plant from Hot Standby to Cold Shutdown from Outside the Control Room, had not been completed at the inspection conclusion. Specifically, the revision to Abnormal Procedure AP-990, Shutdown from Outside Control Room, had not been completed and was not available for review by the inspector. This item remains open (Section E8.21).

A Non-Cited Violation (NCV 50-302/97-19-02) was identified for failure to validate operating curves for technical accuracy when they were received from the vendor during initial construction. Once the violation was identified, the licensee took aggressive and appropriate corrective actions (Section E8.24).

The inspectors concluded that thermocouples, installed in accordance with the licensee's plan, would enable the licensee to meet the license condition for detecting flow in the decay heat system drop line and the auxiliary spray line (Section E8.29).

The post modification testing for the power uprate on Emergency Diesel Generator 1A was successfully completed. Deficiencies in the control of the bearings for the radiator drive shaft were identified but were corrected prior to returning the diesel to operation and did not impact previous operability (Section E8.30).

The inspectors concluded the licensee had adequately modified and tested the building spray pump impellers to assure net positive suction head was adequate to prevent cavitation during accident conditions. This significant design and restart issue is therefore considered closed (Section E8.35).

The licensee adequately modified and tested the plant to resolve licensee restart Design Issue 1, regarding high pressure injection pump recirculation. It was identified as another example of NCV 50-302/97-21-01, Examples of Noncompliances in Design Control, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion. However, the licensee's test procedure was not completed and issued in a timely manner to allow for thorough review. The test safety assessment did not adequately reflect the test in that it erroneously assumed plant conditions not required by the test, was issued 12 days prior to the test issuance, and contained errors in Low Temperature Overpressure Protection parameters (Section E8.37).

Licensee restart Design Issue 8, on Generic Letter (GL) 96-06 concerns, was adequately resolved. The licensee implemented a unique solution to one concern by the expansion chamber design and fully embraced the intent of GL 96-06. However, the licensee's Restart closure packages were occasionally poorly organized compilations of raw data that did not effectively justify the resolution of a safety concern. The inspectors considered this another example of a previously observed problem with corrective action not being driven by a precursor card in the corrective action system (Section E8.38).

Plant Support

A Non-Cited Violation (NCV 50-302/97-19-03) was identified for failure to rescreen Fitness for Duty personnel under the provisions of 10 CFR 26 (Section S1.1).

A Non-Cited Violation (NCV 50-302/97-19-04) was identified for failure to have a protected barrier in place (Section S2.1).

A Non-Cited Violation (NCV 50-302/97-19-05) was identified for failure to compensate for a security zone in access mode (Section S2.1).

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																					
	O7.1	O7.4	O8.2	O8.3	E1.2	E1.3	E1.4	E8.1	E8.2	E8.3	E8.4	E8.5	E8.6	E8.7	E8.8	E8.9	E8.10	E8.11	E8.12	E8.13	E8.14	E8.15
Management Oversight	A	G	A	A	A	A	A	G	G	G	G	G	A	A	A	A	A	A	A	A	G	A
Engineering Effectiveness	A		A		A	A	A	G	G	G	G	G	A	A	A	A	A	A	A	A	G	A
Knowledge of Design Basis	A		A	A	A	A	A	G	G	G	G	A	A	A	A	A	A	A	A	G	G	A
Compliance With Regulations	A	A	A	A	A	A	A	G	G	G	G	A	A	A	A	A	A	A	A	A	G	A
Operator Performance	A	I		A																		

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

NRC AREA OF CONCERN	ASSESSMENT SECTION																				
	E8.16	E8.17	E8.18	E8.19	E8.20	E8.21	E8.22	E8.23	E8.24	E8.25	E8.26	E8.27	E8.28	E8.31	E8.32	E8.33	E8.34	E8.35	E8.36	E8.37	E8.38
Management Oversight	A	A	I	A	G	A	G	G	G	A	G	A	A	A	A	A	A	A	I	A	A
Engineering Effectiveness	A	A	I	A	G	A	A	G	G	A	G	A	A	A	G	G	A	A	I	A	G
Knowledge of Design Basis	A	A	I	A			A	G	[illegible]		G	A	A	A	A	A	A	A	A	A	G
Compliance With Regulations	A	A	I	A	G	A	A	G	G	A	G	A	A	A	A	A	A	A	A	A	G
Operator Performance	A		I			A			G											S	

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

Section 07.1:	Management Corrective Action Plan (MCAP II)
Section 07.4:	Licensee Corrective Action Program Performance
Section 08.2:	(Open) EA 95-126, VIO I.D.1, (05013): Design Controls Failed to Ensure Adequate Safety Margin for HPI Pumps for Certain LOCA Scenarios
Section 08.3:	(Open) IFI 50-302/97-14-01: Review of Operational Procedures Prior to Restart
Section E1.2:	(Closed) CR3 D.I. 2: HPI System Modifications to Improve SBLOCA Margins
Section E1.3:	(Closed) CR3 D.I. 5: Emergency Feedwater System Upgrades and Diesel Generator Load Impact
Section E1.4:	(Closed) MUV-27 Section XI Leakage Testing
Section E8.1:	(Closed) URI 50-302/97-07-03: Reactor Building Liner Plate Degradation
Section E8.2:	(Closed) LER 50-302/97-16-00: Reactor Building Coatings Not Included in Sump Calculations
Section E8.3:	(Closed) LER 50-302/97-18-00: High Energy Line Break Could Result in Loss of Chilled Water to Control Complex Ventilation System
Section E8.4:	(Open) EA 96-365, EA 96-465, EA 96-527, Violation B (example 3) (02013): Error in Design Calculations for Service Water System Heat Loads
Section E8.5:	(Closed) IFI 50-302/97-14-03: Follow-up on Verification of ASME Section XI Valve Testing
Section E8.6:	(Closed) EA 96-365, VIO B (example 4) (02013): Use of Unverified Calculations to Support Modifications
Section E8.7:	(Closed) VIO 50-302/95-21-03: Failure to Isolate the Class 1E From the Non-Class 1E Electrical Circuitry for the RB Purge Valves
	(Closed) LER 50-302/95-25-00 and 02: Personnel Errors By Architect Engineer Result in Operation Outside Design Basis Due to Inadequate Safety/non-Safety Related Circuit Isolation
Section E8.8:	(Closed) IFI 50-302/96-201-17: Coordination of SLUR and Fuse Protection
Section E8.9:	(Closed) LER 50-302/96-19-00, 01, and 02: Classification of

Transfer Switch Causes Potential for Loss of Power to ES Status Lights

Section E8.10: (Closed) LER 50-302/97-23-00: Design Engineering Process Allows Installation of Vital Bus Inverters Containing an Unanalyzed Trip Circuit

Section E8.11: (Closed) LER 50-302/97-32-00: Inadequate Electrical Isolation on a Safety Related Power Supply Due to a Design Error

Section E8.12: (Closed) LER 50-302/97-21-00 and 01: Loss of A Battery Leads to the Inability to Bypass ES Actuation Signals as a Result of Inadequate System Knowledge

Section E8.13: (Open) VIO 50-302/97-14-13: Failure to Take Corrective Actions to identify and Correct the Design Weaknesses Associated with Adequacy of the Past 10 CFR 50.59 Review for Positioning of DHV-34 and DHV-35 During Normal Operation

Section E8.14: (Closed) EA 97-330 (01013): Unreviewed Safety Question Involving Added EDG Protective Trips

Section E8.15: (Closed) LER 50-302/97-05-00: Unanalyzed Condition Regarding Small Break LOCA and Emergency Feedwater

Section E8.16: (Closed) LER 50-302/95-09-00: Minimal Release During Sulfur Dioxide Delivery Causes Actuation of Toxic Gas Monitor Resulting in Control Room Emergency Ventilation Actuation

Section E8.17: (Closed) LER 50-302/96-04-00: Control Complex Habitability Envelope (CCHE) Control Dampers Found Damaged and Leaking

Section E8.18: (Open) URI 50-302/95-02-02: Control Room Habitability Envelope Leakage

(Closed) LER 50-302/97-22-00 and 01: Calculation Errors Associated With Control Complex Habitability Envelope Unfiltered Air Inleakage Could Allow Operator Dose Limits to be Exceeded

Section E8.19: (Closed) VIO 50-302/96-09-05: Failure to Incorporate Design Information into Operations Procedures

Section E8.20: (Closed) VIO 50-302/96-09-07: Inadequate Corrective Action for Implementation of EFIC Task Force Recommendations

Section E8.21: (Open) VIO 50-302/97-02-03: Adequacy of Procedures to Take the Plant from Hot Standby to Cold Shutdown from Outside the Control Room

Section E8.22: (Open) VIO 50-302/97-16-03: Failure to Design and Install

Radioactive Waste Disposal System Piping as Described in the FSAR

(Open) LER 50-302/97-38-00: An Engineering Oversight Resulted in Operation Outside of the Design Basis for the Waste Disposal System

Section E8.23: (Closed) IFI 50-302/96-201-15: Verification of Motor Starting Data; [identified as IFI 50-302/96-201-05 in Inspection Report 50-302/96-201]

Section E8.24: (Closed) URI 50-302/96-201-03: Operating Curves 16, 17, and 18 in OP-103B are not Validated by Licensee

Section E8.25: (Closed) LER 50-302/97-30-00: Installation Error Resulted in a Containment Isolation Check Valve Disc Sticking in The Open Position

Section E8.26: (Closed) VIO 50-302/97-14-04: Failure to Adequately Test HPI Valves MUV-23, 24, 25, and 26 Power Selector Switches

Section E8.27: (Open) LER 50-302/97-13-00: Emergency Diesel Generator Room Could Exceed Maximum Design Temperature During Operation Due to Inadequate Room Cooling

(Open) LER 50-302/97-19-00, 01: Elevated EDG Supply Air Temperatures Due to EDG Radiator Discharge Air Recirculation Effect

(Open) LER 50-302/97-27-00: Failure to Add Antifreeze to the Diesel Generator Coolant Radiators May Render Emergency Diesel Generator Inoperable During Sub-Freezing Temperatures

Section E8.28: (Closed) EA 96-365, VIO A (01062): Inadequate 50.59 Evaluation for Post-LOCA Boron Precipitation Control

Section E8.31: (Closed) IFI 50-302/95-15-04: Code Requirement for Thermal Relief Valves on Decay Heat Removal Heat Exchangers

Section E8.32: (Closed) LER 50-302/97-15-00 and 01: Non-Conservative Differential Pressure Rating for Letdown Line Inboard Containment Isolation Valves Prohibiting Safety Function Performance

Section E8.33: (Closed) LER 50-302/97-28-00: Flow Element Accuracy Uncertainty May Be Greater Than Assumed Due to a Design Error

Section E8.34: (Closed) LER 50-302/97-39-00: Unqualified Material Left in Reactor Building During Construction Could Affect Post-LOCA Cooldown Capability

Section E8.35: (Closed) URI 50-302/96-201-02: Net Positive Suction Head for Building Spray Pump

Section E8.36: (Closed) LER 50-302/97-24-00: Feedwater Valves Do Not Meet the Requirements for Main Steam Line or Feedwater Line Break Event Mitigation Due to Inadequate Design and Calculation Errors

Section E8.37: (Closed) Restart Design Issue D.I.1: High Pressure Injection Pump Recirculation to Make-up Tank

(Closed) LER 50-302/97-08-00: Potential of HPI Pump Recirculation Capability Resulting in Possible MUT Overflow or Possible Pump Failure

(Closed) IFI 50-302/96-17-02: Potential for HPI Recirculation Resulting in MUT Overflow

Section E8.38: (Closed) Restart Design Issue D.I.8: Generic Letter (GL) 96-06 (Thermal Overpressure Protection for Containment Piping, Penetrations and Coolers)

(Closed) LER 50-302/97-04-01: Thermal Relief Valves Inside Containment Do Not Meet Requirements For A Design Basis Accident

(Closed) LER 50-302/97-12-00: Industrial Cooling (CI) System Penetrations Not Designed for Containment Isolation

(Closed) IFI 50-302/96-08-02: Reactor Building Cavity Cooling Piping Thermal Relief Protection

Report Details

Summary of Plant Status

The unit began the inspection period in Mode 5 with the reactor coolant system (RCS) pressurized to approximately 50 pounds per square inch gauge (psig) by a pressurizer bubble, continuing in the outage that began on September 2, 1996. Several train swaps between operable forced decay heat removal system trains were performed throughout the report period to support testing and modification 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. A vacuum in the main condenser was maintained throughout the report period using steam from adjacent coal power plants. Long cycle cleanup of the secondary condensate system was established on December 25, 1997.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707 the inspectors conducted routine reviews of ongoing plant operations which included shift turnovers, response to problems, coordination and planning meetings, restoration of systems to service following modifications, and review of clearance tagging processes. Significant observations are discussed in the following paragraphs.

The inspector attended a Mode 4 Hold Point Coordination meeting on December 3, 1997. The licensee was controlling the mode ascension process via Administrative Instruction (AI) Procedure AI-256, Station Readiness for Restart Sequence, Reactor Restart and Power Ascension Plan, Revision 2. The inspector observed that the meeting effectively reviewed the open items and program goals contained in the AI-256 checklists assigned to each department and that each item had a cognizant and accountable owner. The inspector observed that personnel attending the meeting were well prepared, appropriate topics were discussed, and the licensee was effectively reviewing program areas prior to mode ascension.

The licensee continued to experience clearance tagging problems. NRC Inspection Report (IR) 50-302/97-17 discussed two examples of inadequate clearance implementation. Inspection Reports 50-302/97-01 and 50-302/97-05 each contained violations for inadequate clearance processes and failure to follow the process. During the month of December 1997, the licensee experienced three more significant clearance problems which included: 1) manipulation of a red tagged breaker handle with the breaker removed from the cubicle; 2) the authorization by operations shift management and subsequent manipulation of a motor-operated valve red tagged breaker handle to open the breaker cabinet; and 3) the closure of red-tagged open electrical links by contract instrument technicians. The inspectors were significantly concerned with these items because they indicated a disregard for the requirements to not manipulate red tagged components and a disregard for the protection of personnel and equipment provided by the tagging system. However, the licensee was also very concerned with these events and was addressing them seriously via their corrective action system under Precursor Card (PC) 97-8720, which was graded at the most significant A-level and therefore required a thorough investigation and root cause determination. The licensee had also taken prompt and appropriate short-term corrective actions. The inspector will make a final assessment of these problems at the conclusion of the licensee's investigation and corrective action plan determination.

As discussed in Section E8.37, the inspectors observed an excellent example of initiative and questioning attitude by a licensed operator who reviewed a test procedure and identified several valid concerns.

O3 Operations Procedures and Documentation

O3.1 Low Temperature Overpressure Technical Specification Implementation

a. Inspection Scope (71707)

The licensee submitted Technical Specification Change Request Notice (TSCRN) 213, Revision 0, on July 18, 1997, and Supplement 1 on September 12, 1997. The submittal proposed new Low Temperature Overpressure Protection (LTOP) System requirements and bases. The licensee administratively implemented the requirements on December 8, 1997. The Technical Specification revision was approved on December 22, 1997 (License Amendment 161). The inspector verified that the proposed changes were properly incorporated in administrative controls.

b. Observations and Findings

On December 5, 1997, the licensee issued a Night Order per Operations Instruction (OI)-06, Shift Orders, Revision 3, for the plant operations personnel which stated that on December 8, 1997, new controls for LTOP would be implemented in a number of plant procedures. The license amendment approved on December 22, 1997, provided the same requirements that were implemented on December 8, 1997. The inspector reviewed the procedures and verified that the requirements were properly implemented.

c. Conclusions

The inspector verified that the licensee had appropriately incorporated administrative controls for the LTOP TS amendment.

O6 Operations Organization and Administration

O6.1 Shift Management Organization Changes (71707)

On December 8, 1997, Operations implemented a significant change in shift operations' management titles and positions. The former organization consisted of a Nuclear Shift Manager (NSM), who was considered the senior shift manager and was not required to be licensed; a Nuclear Shift Supervisor (NSS) who was the senior licensed operator; and an Assistant Nuclear Shift Supervisor (ANSS) who was also a senior licensed operator. The individuals filling the NSM position were transferred to a new position of Shift Technical Advisor (STA)/Work Control Center Supervisor (WCS), and the previous NSSs were renamed as the NSMs. The ANSS position was renamed as NSS. The significance of this realignment is that the senior manager on shift (the NSM) is now a licensed senior reactor operator (SRO) who is a more appropriate individual to direct shift operations. It also eliminated overlaps and unclear responsibilities between the old NSMs and NSSs and allowed the licensee to implement staffing of their new Work Control Center with the STA/WCS position. The inspector considered this an improvement in the Operations organization, and it was another example of licensee management's effort to define areas of responsibility and ownership clearly.

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 November 1997, an NRC inspection of MCAP II concluded that the licensee had one MCAP II item on which additional progress was needed prior to restart. The results of that inspection are documented in IR 50-302/97-17. During this inspection, the inspectors followed up on the status of that item.

b. Observations and Findings

The one remaining MCAP II item on which licensee action was needed prior to restart was providing licensing basis information and training to the plant staff to support operability evaluations, TS interpretations, and 10 CFR 50.59 evaluations. The inspectors verified that the licensee had provided licensing basis information to the plant staff by placing it on the plant computer system and issuing an internal correspondence describing the information and how to access and use it. The inspectors reviewed the licensee's internal correspondence, used it to access the licensing basis information, and concluded that the instructions were sufficiently clear and simple for plant personnel to be able to readily access the licensing basis information. The inspectors also had a member of the plant staff, who had not seen the instructions before, demonstrate that he could effectively use the instructions to access the licensing basis information.

c. Conclusions

The inspectors concluded that the licensee's progress to date on MCAP II was satisfactory for plant restart.

The inspectors assessed the licensee's performance, relative to MCAP II, 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

O7.2 Procedure Adherence and Temporary Changes

a. Inspection Scope (71707)

The inspectors reviewed the licensee's Quality Assurance (QA) audit, 97-11, issued on December 22, 1997, which addressed procedure adherence and use of the temporary change process

b. Observations and Findings

On December 2, 1997, the QA organization issued PC 97-8169, which documented an audit finding on the use of the Immediate Work Copy Change (IWCC), as described in licensee procedure AI-400C, New Procedures and Procedure Change Processes. The finding stated that there were at least 100 examples where the process outlined in the procedure had not been followed. The audit identified that only approximately 20 percent of the IWCCs identified in PCs were necessary due to plant conditions. QA concluded that the remainder identified procedural deficiencies that impacted performance.

The inspector reviewed the finding for the IWCC program, discussed the concerns with the QA auditor, and reviewed a number of the individual procedure changes. The QA audit identified that many of the deficient conditions were administrative in nature, such as failure to identify the working copy number of the procedure on the associated PC. One issue identified by the audit was the change of Procedure AI-602, Modification Approval Record (MAR) Work Package Preparation, Implementation, and Closure, to allow a safety related modification to be installed prior to the development of the total MAR package. This procedure normally only allowed this to be done on non-safety related modifications. The QA audit considered this to be an inappropriate use of the IWCC process. The inspector reviewed the corrective action program and located three PCs that documented this same change: PC 97-6792, dated July 14, 1997, PC 97-7060, dated October 15, 1997, and PC 97-7786, dated November 13, 1997. PC 97-7060 identified that even though the IWCC had been performed, it was an inappropriate use of the process, since it changed the intent of the procedure. Licensee procedure AI-400C, New Procedures and Procedure Change Processes, Step 3.3.2, Change of Intent, and Enclosure 11, of the then current revision, stated that IWCCs could not be made to change the intent of the procedure.

As a result of the PC and the QA audit, AI-602 has been permanently revised to allow the implementation of WRs generated describing work to be done prior to modification preparation, design verification, and approval. Work may proceed, provided the Director, Nuclear Engineering and Projects or his designee and the Director, Nuclear Plant Operations, or his designee, approved the WR prior to the beginning of field installation work. The procedure required the WR to contain the reason for the exception as well as any limitations. The procedure stated that the work performed under this WR may not be considered complete or the component returned to service prior to completion, design verification, and approval of the MAR. The prior revision restricted to non-safety related modifications, but the revision has extended this to safety related modifications.

Procedure AI-400C was revised to clarify the use of IWCCs. Conflicts between the body of the procedure and the checklist controlling IWCCs have been removed by deleting the checklist and installing sign-off steps in the body of the procedure. The PC for this issue is being included in the root cause and corrective actions for a major procedure upgrade program, which is in process of being developed.

Failure to follow AI-400C requirements for the use of the IWCC process is a violation. However, the licensee identified and corrected problem is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. This issue is identified as NCV 50-302/97-19-01, Failure to Follow Procedure for the Use of Immediate Work Copy Changes to Procedures.

c. Conclusions

A Non-Cited Violation (NCV 50-302/97-19-01) was identified by the licensee quality assurance program for a significant problem with the use of immediate work copy changes to procedures. Immediate corrective actions were taken by the licensee to address the identified weaknesses. The licensee was taking long term corrective actions for Precursor Card 97-8169 by proceeding with a procedure upgrade program.

O7.3 Licensee Self-Assessment Activities (71707, 40500)

The inspectors reviewed various licensee self-assessment activities including routine reviews of Nuclear Quality Assessments (NQA) activities and surveillance report findings. The inspectors have not observed any problems with NQA activities and noted that NQA continued to allocate their efforts appropriately. The inspector noted NQA had walked down numerous plant systems and verified the adequacy of the licensee's restart item resolutions and restart restraint processes. The inspector has observed that NQA has achieved some productive results from these activities. NQA findings have included inadequate tracking and labeling of spool pieces, inconsistent or poor implementation of commitments, and inadequate transfer of requirements deleted from the Technical Specifications (TS) during the transition to the Improved TS in 1995. Each of these items was appropriately dispositioned in the licensee's corrective action system. The inspector concluded that the licensee's quality assurance auditors continued to have a positive impact on the safety and operation of the plant and their observations and conclusions were consistent with inspector observations and conclusions.

07.4 Licensee Corrective Action Program Performance

a. Inspection Scope (71707, 40500)

The inspectors routinely review items entered into the licensee's Corrective Action Program and the subsequent rating and disposition of the items. During these reviews, the inspectors review the reportability determinations done on new PCs. The inspectors also routinely review items on the licensee's Mode Restraint Tracking List.

b. Observations and Findings

On December 8, 1997, PC 97-8333 was initiated to document radiation monitor (RM-A) wires which were discovered to be not terminated. The wires transmitted an automatic shift function to mid and high range indication on increasing counts for RM-A1, the reactor building purge duct monitor, and RM-A2, the auxiliary building exhaust monitor. These functions were required in Modes 1 through 4 by the Offsite Dose Calculation Manual (ODCM) and were therefore considered reportable, but PC 97-8333 was initially determined to be potentially reportable by the NSM. The NSM had stated he could not find a requirement for the function to be operable. It was then sent to a reportability reviewer and determine 10 days later to be reportable as a Special Report to the NRC per the [illegible] it had also been inappropriately screened as a "C" level PC by [illegible] Screening Committee (PCSC), which only would have required an apparent cause investigation and no extent of condition review. This error was corrected by licensee management at their routine review of PCSC ratings and upgraded to a Level "B", requiring an extent of condition evaluation and formal root cause. The licensee and inspectors have continued to observe PC level screening problems but licensee management has corrected the problems during their reviews. They have taken further steps to improve the oversight and performance of the PCSC.

The inspectors have also observed other problems with PCs, such as several duplicates issued on the same problem and others where no PC was issued. The inspector considered these indicative of poor supervisory review of PCs. The licensee has also observed problems with the wording and identification of the problem in PCs that they have attributed to the same cause. They have reenforced their expectations for supervisory performance in writing and have promulgated guidance to the PCSC to hold supervisors accountable, contacting them for unclear PCs.

The inspectors review of the Restart Mode Restraint List identified several examples of items not tracked for completion prior to entering Mode 4. These included: 1) security restoring an unmonitored area to vital area status; 2) the resolution of open setpoint problems opened in

IFI 50-302/97-17-05, Resolution of Improved Technical Specification Setpoint Program Deficiencies Prior to Entry Into Mode 4; and 3) engineering resolution of some emergency diesel issues. The tracking of these items was corrected, and the inspector noted that the issues were scheduled to be resolved prior to the Mode change. The licensee's NQA auditors have also had similar observations. The licensee and inspector considered these omissions to be indicative of the vulnerability of the licensee's manual mode restraint tracking system as discussed in Inspection Report 50-302/97-16.

c. Conclusions

Although it took 10 days to make the reportability determination for the RM-A PC, the special report was issued when required. The inspector concluded the delay was unnecessary due to the poor initial reportability determination. Licensee management continued to exercise oversight of precursor card screening results and developed appropriate barriers to avoid incorrect rating decisions. The licensee has also taken appropriate preliminary actions to address problems with supervisory reviews of precursor cards. Omissions from the Mode Restraint Tracking system were indicative of the challenge to maintain a manual tracking system.

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 - Inadequate

O8 Miscellaneous Operations Issues

O8.1 (Open) IFI 50-302/97-11-04: Corrective Actions for Approximately 4000 Precursor Cards Not Tracked to Completion

a. Inspection Scope (40500)

This Inspector Follow-up Item (IFI) was opened to follow up on the licensee's corrective actions for approximately 4000 PCs that had not been tracked to completion. From about November 1996 through June 1997, the licensee's corrective action program did not require tracking the completion of corrective actions for Grade C and D PCs. The corrective action program also had no record of completion of corrective actions for these PCs. 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires that, for conditions adverse to quality, corrective action be taken and records of that corrective action be maintained. However, individual departments may have accomplished the corrective actions and maintained records, and this IFI was opened for NRC followup to determine if that had occurred.

b. Observations and Findings

The licensee had opened restart item OP-2B on this issue. A licensee audit of the issue, documented in Quality Programs Surveillance (QPS) Report QPS-97-0129, dated September 22, 1997, assessed the potential concern. The audit reviewed 3674 Grade C and D PCs from January through May 1997 and determined that 2046 of them were conditions adverse to quality. The remaining PCs were considered improvements to processes, positive comments, or concerns. Further Nuclear Safety Assessment Team (NSAT) review of Grade C PCs determined that 144 were not addressed and 426 identified actions for resolution for which there was no record of implementation in the PC file. Subsequent NSAT efforts in contacting responsible departments and individuals reduced the number of unresolved PCs to 156. NSAT personnel also reviewed all Grade C and D PCs from November 1996 through June 1997 and determined that six should be considered restart items. Of those six, five were being addressed by existing licensee restart items and one was reopened to track completion of corrective actions. In discussions with NSAT personnel, the inspectors found that Grade C and D PCs from November - December 1996 had not been reviewed by the licensee to assure that, for all identified conditions adverse to quality, corrective actions had been accomplished. The licensee stated that they would include the 1996 PCs in their review and expected to complete the review in January 1998. The inspector reviewed sample C and D PCs from November 1996 - June 1997 for additional items that would need to be addressed before plant restart and did not identify any.

c. Conclusions

The inspectors concluded that the approximately 4000 PCs, which had been closed without tracking completion of corrective actions, had been adequately reviewed for potential restart items. Also, restart items identified by this review were being adequately addressed by the licensee. This item remains open pending NRC review of the licensee's followup on these approximately 4000 PCs to assure that corrective actions for all conditions adverse to quality were accomplished and recorded, but it is acceptable for restart of the plant.

O8.2 (Open) EA 95-126, VIO I.D.1, (05013): Design Controls Failed to Ensure Adequate Safety Margin for HPI Pumps for Certain LOCA Scenarios

a. Inspection Scope (92901, 92903)

The inspector reviewed EA 95-126, VIO I.D.1, dated July 10, 1996, which addressed the failure of the licensee to incorporate the design basis of the emergency core cooling system (ECCS) into the plant procedures. Specifically, the makeup tank operating curve was not a true reflection of the design basis. The licensee's response to the violation, in a letter dated September 9, 1996, outlined the corrective action plan for this violation. The inspector reviewed the corrective actions and performed an inspection to assure the successful completion of these actions.

b. Observations and Findings

The corrective actions addressed in the licensee letter included the same general improvements in the engineering process as for violations EA 95-126, VIO I.C.1 (03013), Failure to Take Adequate Corrective Actions for Operator Concerns Regarding OP-103B, Curve 8, for MUT Pressure/Level Limits, and VIO I.C.2 (04013), Corrective Actions for an Inadequate Curve 8 Were Also Incorrect. These corrective actions were reviewed in IRs 50-302/97-07 and 50-302/97-17 and were found to be acceptable and were closed.

Licensee corrective actions included reviewing the licensee operating curve procedure series, OP-103A through OP-103F, and revising the procedures to reflect accurately the ECCS design basis. The licensee's system readiness review (SRR) process was used to verify the design basis for the ECCS systems. The inspectors verified that the licensee has updated the procedures, as calculations and modifications have been implemented in the plant, as a result of the SRR and system upgrades. The inspector verified that the operating curve procedure series, with the exception of OP-103F, Tank Volumes, had been updated, as part of the corrective actions for this violation. The inspectors verified that the calculations for the tank volumes were completed, as a corrective action for EA 95-126, VIO II.B (08014), Failure to Take Adequate Corrective Actions for Tank Volumes/Level/Suction Point, as discussed and closed in IR 50-302/97-17.

c. Conclusions

This violation remains open, pending the final revision of OP-103F. This procedure is scheduled to be revised prior to entering Mode 4 operation.

The inspector assessed the licensee's performance, with respect to these issues, in the five areas of continuing NRC concern.

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

O8.3 (Open) IFI 50-302/97-14-01: Review of Operational Procedures Prior to Restart

a. Inspection Scope (71707, 92901)

Inspection Report 50-302/97-14 concluded that the adequacy of the system operating procedures for the Makeup and Purification system and the decay heat removal system could not be determined due to the large number and types of changes pending. The inspector has continued to inspect these procedures to assess the adequacy of the procedures, following issuance of the revisions.

b. Observations and Findings

Licensee Procedure OP-402, Makeup and Purification System, Revision 95, was issued on December 8, 1997. The inspector reviewed the NUPOST computer data base and verified that following the issuance of the revision, the number of outstanding comments on this procedure decreased from 22 to four. The remaining items were all graded as low priority by the operations procedure group. The inspector reviewed the revised procedure and determined that at least one of the remaining comments was discussed as an outstanding comment in IR 50-302/97-14; step 4.2.12 for testing the MUP backup gear oil pump would not verify that this pump would auto start if necessary, since the main gear oil pump was started in a previous step. In addition, the step did not identify to the operator that the shaft driven gear oil pump might be faulty and that the MUP may need to be secured if the main gear oil pump did not shutdown in two minutes after the MUP was started. Licensee Procedure OP-404, Decay Heat Removal System, Revision 111, was still in revision at the time of the inspection.

The inspector reviewed outstanding NUPOST entries for various operations procedures and determined that even though the licensee issues a revision to a procedure, at times comments are left open, intended to be incorporated in future revisions to the procedures. This process lacked efficiency, in that recognized desirable enhancements to procedures were not always incorporated into the latest revision but remained open.

The licensee has determined that operational procedure weaknesses exist in all departments and has planned to implement a procedure upgrade program in the near future. The inspector will continue to review this effort as it is accomplished.

c. Conclusions

This item remains open. The inspector will continue to review operations procedures as they are revised, prior to restart, to determine adequacy for use during operations.

The inspector assessed the licensee's performance, with respect to these issues, in the five areas of continuing NRC concern.

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

II. Maintenance

M1 Conduct of Maintenance

Mi.1 General Comments

a. Inspection Scope (62707, 61726)

Using Inspection Procedures 62707 and 61726, the inspectors observed all or portions of the following work requests (WR) and surveillances and reviewed associated documentation. The following activities were included:

- PT-315, Remote Shutdown Relay Operability
- SP-335C, Radiation Monitoring Instrumentation Functional Test of RM-A1, A2, A6, A11 and A12

b. Observations and Findings

The inspectors observed the activities identified above and concluded that all work observed was performed with the work packages present and in active use. Pre-job planning was thorough and in sufficient detail to prepare the technicians for the assigned tasks. Technicians were experienced and knowledgeable of their assigned tasks. The inspectors frequently observed supervisors and system engineers monitoring job progress, and quality control personnel were present whenever required by procedure.

c. Conclusions

The inspectors concluded that Maintenance activities were performed in accordance with procedures and desired results were obtained.

M1.2 RWP-3A Suction Blockage Root Cause (62707)

a. Inspection Scope (62707)

In IR 50-302/97-17, the inspectors discussed the October 26, 1997, suction blockage and inoperability of the Nuclear Services and Decay Heat Sea Water (RW) pump RWP-3A. The inspector reviewed the results of the licensee's investigation and resolution of that problem.

b. Observations and Findings

A wooden shim and attached lanyard were found blocking the pump suction and were retrieved by divers. Since the licensee determined that the shim and lanyard had been known to be lost during maintenance on the seawater intake structure which eventually feeds the RWP suction, the licensee initiated PCs 97-7417 and 7385. The licensee and inspector were significantly concerned regarding the adequacy of foreign material exclusion (FME) practices used for that work and why the wood had not been retrieved before it became a problem. The licensee planned to do a rigorous root cause investigation. PC 97-7417 was initiated by the maintenance department to address the problems that resulted in the loss of the shim. The inspector reviewed the completed PC and noted that it was graded Level C so it did not receive a root cause determination. The inspector determined the apparent cause, which attributed the problem to misjudgement and lack of information regarding the need for FME control, to be a poor and shallow evaluation. The inspector also observed that the majority of the maintenance corrective actions, which included briefing "all craft people" on their complete knowledge of the FME procedure, were done the same day the PC was initiated. The inspector considered these actions misleading and poorly worded because only the craft people associated with the problem were included. PC 97-7417 did not perform a formal extent of condition evaluation to determine if other work with inadequate FME control could result in another RW pump problem. The PC also made no reference to any other PC that would be evaluating the FME program.

Precursor Card 97-7385 was initiated by Operations and assigned to engineering Technical Support. The inspector observed this PC was also graded as a Level C. It did not address the cause of the pump blockage but documented the operational action plan discussed in IR 50-302/97-17 to get divers in to the RWP pit to remove the blockage. Although it thoroughly documented this action plan, an adequate apparent cause evaluation was not done. As a result no long term corrective actions were developed. The only action listed was to get the diver in to remove the blockage. An extent of condition evaluation was also not done and another PC was not referenced as addressing the FME problem.

Both PCs 97-7417 and 7385 were closed. The licensee had conceptually assigned the root cause effort for the RWP blockage to an already ongoing Level B PC review for general FME problems. This was partially initiated to respond to Violation 50-302/97-14-07, Failure to Follow FME Procedure Requirements, under PC 97-6903. However, there was no reference to this PC in either of the two RWP PCs, and specific actions to address the RWP concerns were not incorporated in PC 97-6903. The inspector observed that the licensee was aggressively addressing the FME programmatic deficiencies identified in PC 97-6903 and that it was appropriate that the programmatic FME issues from the RWP blockage could be assigned to this effort. However, PCs 97-7385 and 97-7417 were unsatisfactory resolution of the specific problem associated with the RWP blockage. The inspector considered the potential to block RWP suctions to be a very significant issue for the potential to render all RWPs inoperable. The RW supplies essential cooling to systems such as decay heat removal, spent fuel pool cooling, and high pressure injection pumps. The licensee's effort to resolve this issue was very poor because a specific RW extent of condition (EOC) evaluation was not performed, apparent cause evaluations were cursory, and corrective actions appropriate to the specific problem were not developed. The inspector determined that the root cause of this issue was already encompassed by existing VIO 50-302/97-14-07 for programmatic FME concerns. An example highlighting the inspector's concern for lack of an EOC review was seen on December 13, 1997, when PC 97-8563 was initiated to document assorted debris found in the A RWP pit and a stop log shim gasket found on a RW heat exchanger tubesheet partially blocking flow. If an adequate EOC had been performed for the RWP blockage, these items could have been identified and dispositioned earlier.

c. Conclusions

Although the licensee was appropriately upgrading their foreign material exclusion program in response to earlier problems, the resolution of the specific problem that resulted in blockage of a raw water pump was extremely poor. Specific appropriate corrective actions were not developed, apparent cause evaluations were cursory efforts, and an extent of condition review for other potential blockage was not performed.

III. Engineering

E1 Conduct of Engineering

E1.1 General Comments (37551)

Using Inspection Procedure 37551 the inspectors conducted routine reviews of ongoing engineering activities which included shift support, response to problems, restoration of systems to service following modifications, scaffolding installation reviews and technical restart item resolutions. Significant observations are discussed in the following paragraphs.

The inspectors reviewed engineering's support and troubleshooting of a problem with emergency feed pump (EFP)-1 air entrainment. During testing of block valve throttling on December 14, 1997, EFP-1 flow became erratic, loud audible noise was apparent, and EFP-1 discharge pressure and motor amps dropped significantly. The pump was immediately secured and a troubleshooting plan assembled. The inspector reviewed the licensee's plans, performance, and results. The inspector determined their effort to determine the cause, done under Grade B PC 97-8542, was a logical and methodical action plan and was a good effort. They found an inadequate system operations procedure for venting of the suction header which allowed an air pocket to remain in the header after modification work. The air passing through the pump resulted in the observed indications. The licensee adequately resolved and eliminated the potential for any other air entrainment and the inspector determined that system engineering performed well in supporting troubleshooting of this problem.

The inspector performed routine reviews of scaffolding installation and safety analyses to verify the impact on safety-related equipment was minimized and within procedural requirements. The inspectors observed some scaffolding that was installed in both the operable and inoperable emergency diesel (EDG) rooms. However, the inspector determined that the scaffolding in the operable, "protected train" EDG room was adequately restrained and presented no concern on EDG operability. Scaffolding being installed in the inoperable EDG room met licensee procedural requirements for maintenance around an inoperable piece of equipment. No concerns were identified.

E1.2 (Closed) CR3 D.I.2: HPI System Modifications to Improve SBLOCA Margins

a. Inspection Scope (37550)

The NRC Confirmatory Action Letter to Crystal River of March 4, 1997, required that, prior to restarting the plant, FPC adequately address the eight design issues identified by the licensee in a letter to the NRC dated October 8, 1996. HPI system modifications to improve small break loss of coolant accident (SBLOCA) margins was one of the eight design issues. The licensee's October 8, 1996 letter stated that the HPI system currently met the licensing and design basis. The letter stated that the plan for the HPI system was to install no modifications prior to plant restart and to install three modifications after plant restart, during the next refueling outage: 1) one modification would automatically isolate normal makeup flow upon an engineered safeguards (ES) signal; 2) another modification would install cavitating venturis to limit flow in an injection leg due to a postulated downstream break; and 3) a third modification would install cross-tie piping between injection legs downstream of the injection control valves.

Subsequent NRC inspections found that the HPI system did not meet its design and licensing bases. That finding was described in IR 50-302/97-09 and in enforcement action (EA) 97-330, Violation 01013: Inadequate Safety Evaluations for Added Operator Actions for Design Basis SBLOCA Mitigation. In response to that finding, the licensee installed the modification, to isolate automatically the normal makeup flow upon an ES signal, during the current outage and also submitted TSCRN 210, dated June 14, 1997. TSCRN 210 in part requested that the NRC review and approve the operator actions needed for SBLOCA mitigation. The licensee recognized that NRC approval of TSCRN 210 was needed prior to plant restart.

During this inspection, the inspectors reviewed the licensee's modification to isolate automatically the normal makeup flow upon an ES signal.

b. Observations and Findings

The inspectors reviewed MAR 97-02-17-01, Auto-Close MUV-27. This modification added an automatic closure signal to MUV-27, the normal makeup isolation valve, on a 1500# RCS low pressure signal or a manual initiation of ES. That was in addition to the existing automatic closure on a high reactor building pressure signal. The inspectors verified that the field installation and functional testing were complete and identified no deficiencies. Additional post-modification testing was scheduled to be completed as part of required surveillance procedures prior to plant mode changes.

c. Conclusions

The inspectors concluded that the installation of MAR 97-02-17-01, Auto-Close MUV-27, was adequate and that no further NRC inspection of it was needed. CR3 D.I.2 is closed.

The inspectors assessed the licensee's performance, relative to this issue, 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 - Not Applicable

E1.3 <u>(Closed) CR3 D.I.5: Emergency Feedwater System Upgrades and Diesel Generator Load Impact</u>

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

The NRC Confirmatory Action Letter to Crystal River of March 4, 1997, required that, prior to restarting the plant, FPC adequately address the eight design issues identified by the licensee in a letter to the NRC dated October 8, 1996. Emergency feedwater (EFW) system upgrades and diesel generator load impact was one of the eight design issues. The licensee's stated plan was to install three modifications prior to plant restart. The three modifications included: 1) installation of EFW cavitating venturis to eliminate the EFW Net Positive Suction Head (NPSH) concern; 2) restoration of the A train emergency feedwater initiation and control (EFIC) system actuation of auxiliary steam valve ASV-204 to ensure that EFP-2 auto-starts on a failure of ASV-5 or the B side EFW initiate logic; and 3) installation of motor operators on EFW pump discharge cross-tie valves. Prior NRC inspections, documented in IR 50-302/97-08 and IR 50-302/97-11, had determined that the licensee's evaluation and installation of these three modifications was adequate, with one exception. That exception was that the licensee needed to resolve EFW flow control problems, with the EFW cavitating venturis installed, and needed to write a 50.59 safety evaluation for plant operation with the EFW cavitating venturis installed. During this inspection, the inspectors followed up on the EFW cavitating venturis modification.

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

The inspectors reviewed MAR 96-10-02-01, Field Change Notice (FCN) 5, Flow Limit Bypass, dated September 22, 1997. This FCN modified the EFIC system to bypass permanently the feedwater flow input to the EFW flow control valves so that in the future the valve positions would be determined solely by OTSG level. FCN 5 also included Revision 2 to the 50.59 safety evaluation, which concluded that plant operation with the EFW cavitating venturis installed did not introduce an unreviewed safety question.

The inspectors verified that the MAR installation and testing were complete. Satisfactory testing of the EFW flow control, with the cavitating venturis installed and the feedwater flow input to the EFW flow control valves temporarily bypassed, had been accomplished and was reviewed by the NRC as described in IR 50-302/97-08.

The three modifications related to this design issue, and related operator emergency actions, were included in TSCRN 210, which was being reviewed by the NRC. The licensee recognized that NRC approval of TSCRN 210 was needed prior to plant restart.

c. Conclusions

The inspectors concluded that the installation, testing, and 50.59 safety evaluations for the three modifications related to CR3 D.I.5 were adequate. CR3 D.I.5 is closed.

The inspectors assessed the licensee's performance, relative to this issue, 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 - Not Applicable

E1.4 (Closed) MUV-27 Section XI Leakage Testing

a. Inspection Scope (37550)

The licensee had opened restart item R-21 to evaluate what leakage testing needed to be done on MUV-27, the makeup isolation valve and MUV-18, the Reactor Coolant Pump (RCP) seal injection isolation valve, to support the closing of these valves for high pressure injection. The licensee had determined in 1996 that both MUV-27 and MUV-18 needed to be closed when HPI was initiated in order to ensure sufficient HPI flow to the reactor coolant system. Restart item R-21 also evaluated the need for leak testing of HPI valves MUV-23, 24, 25, and 26 to prevent the potential for thermal stratification in the injection lines. The inspectors reviewed the licensee's response to this issue.

b. Observations and Findings

The licensee's evaluation of this issue concluded that all of these valves needed to be leak tested. To accomplish the testing, the licensee developed Performance Testing Procedure PT-601, MU System HPI, Seal Injection, and Makeup Isolation Valve Testing, Rev. 0, dated November 21, 1997. The procedure included acceptance criteria of not more than 0.005 gpm leakage for each of MUV-23, 24, 25, and 26; and not more than 3.783 liters/minute combined leakage for MUV-18, 24, and 27. The inspectors reviewed the licensee's evaluation and PT-601, and identified no deficiencies. The inspectors also verified that the performance of PT-601 was scheduled to be completed prior to Mode 4 and was on the licensee's Mode 4 Restart Mode Restraint List.

c. Conclusions

The inspectors concluded that the licensee's actions in response to the need for leakage testing of MUV-27 were appropriate.

The inspectors assessed the licensee's performance, relative to this issue, 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 - Not Applicable

E8 Miscellaneous Engineering Issues

E8.1 (Closed) URI 50-302/97-07-03: Reactor Building Liner Plate Degradation

a. Inspection Scope (92903)

The unresolved item involved a concern about the condition of an area on the Reactor Building (RB) liner noted during an NRC walkdown. The inspectors followed up on the licensee's corrective actions by reviewing procedures, training records, interviews with cognizant coating personnel, and inspecting conditions in the plant.

b. Observations and Findings

The inspector noted degradation in several spots and visual evidence of localized rusting of the liner plate during a walkdown inspection at the RB 95 ft. elevation. Also, the inspector noted that an area of the plate referenced in PC-2042 had been prepared and recoated without the required ASME Section XI inspections. The licensee's coatings personnel and planners were not aware of the recent NRC Information Notice (IN) 97-10 discussing corrosion of liner plates and the Section XI requirements for RB liner plates. Visual inspections showed surface irregularities to a height of about four inches above the floor slab.

As a part of the resolution of this unresolved item (Restart Issue M-12), the licensee performed walkdowns of the containment liner/penetrations to identify all areas of coating failure and resultant corrosion. Before performing the inspections, the licensee wrote two procedures, N-18 and N-19, for inspecting liner and penetration coatings to ASME Section XI requirements and trained the coatings inspectors on the procedures. In addition, the nuclear planning group was briefed on the new Section XI requirements for coatings and told to route all WRs pertaining to containment boundary through the Engineering Programs group to determine Parts IWE/IWL (Sec. XI) applicability. This corrected an identified program weakness.

The inspector performed two walkdowns of the containment, discussed liner plate thickness measurements, and the evaluation of an uncoated area (approximately 36" wide) found between a concrete wall and the liner. The uncoated area was inspected by Quality Control (QC) and Ultrasonic Test (UT) plate thickness measurements were made of the plate. Using the data, engineering will evaluate the possibility of coating this area at a later date. The results of these efforts by the licensee were acceptable and the unresolved item is closed.

c. Conclusions

The inspector concluded the licensee had properly addressed and repaired the containment coating areas identified and updated QC inspection techniques to ASME Section XI requirements. The licensee also took corrective actions to eliminate a program weakness.

The inspector assessed the licensee's performance relative to resolution of reactor building liner plate degradation 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

E8.2 (Closed) LER 50-302/97-16-00: Reactor Building Coatings Not Included in Sump Calculations

a. Inspection Scope (92903)

The Licensee Event Report (LER) involved the failure to consider unqualified coatings on the core flood tanks and the possibilities that penetrations on the south wall of the D-ring could pass these coatings through to block the containment sump. The inspector followed up on the licensee's corrective actions by reviewing LER information, a MAR package, and observing conditions in and around the D-Ring and containment sump area.

b. Observations and Findings

As a part of the resolution of FPC Restart Issue D-71, the licensee issued PC 97-3292 and PC 97-3277. The first PC identified unsealed and unscreened penetrations in the RB Secondary Shield Wall (D-Ring) that could provide a flow path of water to the RB sump. This flow path was not considered in Calculation S89-0049, which was performed to assure that no more than approximately 50 percent flow blockage of the sump would occur. In addition, PC 97-3277 identified that the CFTs had unqualified coatings, allowing the possibility for more debris that could block the containment sump. Additional walkdowns revealed that the steam generator supports within the D-Ring also had unqualified coatings, and that certain floor drains and floor openings which could contribute to the direct flow path had no screens or covers.

The inspector reviewed MAR 97-06-17-01 for installing screens over the penetrations in the D-ring wall and for adding screens to certain floor drains that provide a direct path from one floor elevation to another. Additionally, the inspector reviewed some of the WRs used to replace the unqualified coatings with qualified coatings on the CFTs and support skirts for steam generators. WRs to cover some of the floor openings with plates were also reviewed. The inspector then walked down the D-ring area and verified the screens over penetrations and floor drains and newly installed plates covering some of the floor openings were installed. The licensee's corrective actions resulted in effective improvements and LER 50-302/97-16-00 is closed.

Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

c. Conclusions

The inspector concluded that the corrective actions had been completed by the licensee and this LER is closed.

The inspector assessed the licensee's performance relative to resolution of reactor building coatings/sump calculations 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

E8.3 (Closed) LER 50-302/97-18-00: High Energy Line Break Could Result in Loss of Chilled Water to Control Complex Ventilation System

a. Inspection Scope (92903, 37550)

This LER (FPC Restart Issue D-79) involved the possibility of losing both control complex chillers following a high energy line break (HELB). The inspector followed up on the licensee's corrective actions by reviewing LER information, a MAR package, having discussions with the design group and observing some of the modification in progress.

b. Observations and Findings

HELB in Zone 19 of the Intermediate Building would cause steam to pass across certain penetration cooling heat exchanger coils, thereby causing an excess cooling load. This heat load would cause the chilled water temperature to rise and eventually cause the chiller rupture disc to fail and relieve vessel pressure. This same condition would also cause failure of the standby chiller (if in service), resulting in a complete loss of chilled water. These failures could affect the ability to maintain the control complex envelope and potentially the ability to ensure continued operation of critical electrical equipment.

MAR 97-11-02-01 was being implemented to stop the chilled water flow through the penetration heat exchanger cooling coils in operation during a HELB in the Intermediate building (95' elevation). Stopping the chilled water flow ensures the temperature of the safety related chilled water system remains below a certain temperature. Keeping the temperature of the chilled water below this temperature would prevent tripping or damaging the operating chiller. This modification would install four solenoid valves to isolate RB penetration cooling system chilled water coils (AHHE-13A, 13B). These solenoid valves are energized to open and spring closed. Thermostats would be mounted in the area of the coils and in case of a HELB and resultant increase in temperature to 135°F and beyond, the solenoid valves would lose power and close, thereby protecting the chillers.

The inspector observed some of the prefabrication in progress, i.e. welding piping and valves. The inspector also toured the area of the penetration cooling coils with the design engineer. This package was partially complete but the inspector's review of the package and in progress work provided enough confidence to close the LER.

Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

c. Conclusions

The inspectors concluded that most of the licensee's corrective actions had been implemented. Completion of the modification prior to Mode 4 was in a licensee tracking system. The LER is closed.

The inspector assessed the licensee's performance relative to resolution of a HELB on the chillers for the control complex ventilation system 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

E8.4 (Open) EA 96-365, EA 96-465, EA 96-527, Violation B (example 3) (02013): Error in Design Calculations for Service Water System Heat Loads

a. Inspection Scope (92903)

This violation was identified for an error in the calculation of total heat loads to the Nuclear Services Closed Cycle Cooling (SW) system and consequently, an error in the affected operating procedure curve. The inspector reviewed the Justification for Continued Operation (JCO), safety evaluation, and modification that was in progress.

b. Observations and Findings

During the evaluation of FPC Restart Item D-28, engineering discovered that the analyses performed in 1987 to support the increase in design basis seawater temperature from 85 to 95°F were deficient.

These analyses assumed only one service water (SW) pump and one raw water (RW) pump running and the RB fans operating at maximum fouling factor. Running two RB fan coolers (assuming the coolers are clean, fouling factor 0.000 and maximum heat would be removed) and two SW pumps supplying 2000 gpm to each RB fan would provide a considerably greater heat transfer rate into the SW. This condition could cause SW heat exchanger outlet temperatures to exceed the 110°F limit. Exceeding SW design basis could cause failure of SW cooled loads and lead to unacceptable accident mitigation capability.

The licensee revised Calculation M94-0056, Allowable SW Heat Exchanger Tube Blockage vs Ultimate Heat Sink (UHS) Temperature. Additionally, Operations Procedure OP-103B, Plant Operating Curves, and the associated curve (Curve 15) were revised to prevent the possibility of exceeding the design SW temperature. This procedure assumed two RB fan coolers coming on and maximum heat removal conditions. The inspector reviewed the JCO and the safety analysis and noted that based on data for the UHS the licensee should be able to operate without a potential UHS temperature problem until mid-March 1998 (therefore no restart constraint).

The inspector reviewed MAR 97-09-05-01, which was being installed to allow only one fan to start on an ES actuation. This would allow the UHS to rise to 95°F (with 0 percent tube blockage). This modification will be completed before Mode 4 and will then be removed from service until a Licensing Amendment Request (LAR) is approved by the NRC. After the LAR is approved the logic to start only one pump will be put back in service and the allowable UHS temperature will be raised.

c. Conclusions

The inspector concluded that all of the licensee's corrective actions for this violation had not been completed at the time of this inspection. This item remains open and will be reviewed further during subsequent NRC inspections. The licensee's completed actions are acceptable for restart.

The inspector assessed the licensee's performance relative to modification for resolution of the SW/UHS heat removal 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

E8.5 <u>(Closed) IFI 50-302/97-14-03; Followup on Verification of ASME Section XI Valve Testing</u>

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

During a licensee self-assessment valves were identified that were not being adequately tested to Section XI requirements.

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

During a Safety System Functional Inspection (SSFI), the team reviewed a list of valves not presently in the Section XI test program and were required to be tested by the present code. The team informed the licensee that these valves should be tested before startup.

The inspector reviewed the list of valves identified by the team and discussed the testing of these valves with the licensee. Of the 22 valves listed, all but one of these valves had been or was scheduled to be tested before startup. Valve RCV-10, a power operated relief valve, is only tested when sent out to a specialized vendor and will not be tested before startup.

c. <u>Conclusions</u>

The licensee's actions to address the Section XI valve test were adequate. IFI 50-302/97-14-03 is closed.

The inspector assessed the licensee's performance relative to resolution of the valve testing issue in the five areas of continuing NRC concern.

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

E8.6 <u>(Closed) EA 96-365 VIO B (example 4) 02013); Use of Unverified Calculations to Support Modifications</u>

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

Part 4 of Violation B was identified as a deficiency where regulatory requirements were not translated into procedures, and the licensee failed to provide measures to verify the adequacy of design by an individual other than those who performed the original design. Specifically, Nuclear Engineering Procedure (NEP)-210, Modification Approval Records, Revision 15, dated January 16, 1996, was inadequate in that it allowed unverified calculations to be relied upon to support modification installation and return to service. As a result, Request for Engineering Assistance (REA) 96-047, EDG Loading Case Study, was not verified and was used to support modification package MAR 96-04-12-01 in April 1996, which contributed to the introduction of three USQs related to EDG loading. The inspectors reviewed the licensee's corrective action implemented for this item to determine if it was in accordance with the licensee's commitments, requirements, procedures, and NRC regulations.

b. Observations and Findings

The licensee assessed their NEP guidance and design control program to identify the necessary corrective actions needed. The inspectors verified that the engineering procedures were revised to reflect additional requirements for: 1) verification of technical information, 2) guidance and criteria for review and verification of calculations and plant modification "return to service", 3) controls for interdisciplinary modifications, 4) completion of a verification report for any new or revised calculation, 5) development of a verification checklist, and 6) use of a Design Review Board for design packages.

The following procedures were revised to reflect the requirements discussed:

- NEP-201, Preparation and Processing of Special Projects and Engineering Studies
- NEP-202, Preparation and Control of Conceptual Designs and Design Walkdowns
- NEP-210, Modification Approval Records
- NEP-211, Commercial Grade Design Control
- NEP-212, Processing of Modification Projects by Nuclear Projects
- NEP-213, Design Analyses/Calculations
- NEP-222, Qualification for Equipment in the Scope of 10CFR50.49
- NEP-234, Station Blackout
- NEP-253, Preparation and Control of a Document Change Notice
- NEP-254, Plant Equipment Equivalency Replacement Evaluation
- NEP-261, Design Verification
- I-410, Plant Preparation and Nuclear Engineering Processing of Request for Engineering Assistance
- AI-400F, New Procedures and Procedure Change Processes for EOPs, APs and Supporting Documentation

The Mechanical Engineering Group completed a detailed review and re-verification of all the safety-related ES System hydraulic calculations. Each calculation was reviewed and verified to be correct, except for the

Decay Heat System, which had one discrepancy. This discrepancy was corrected.

The Electrical Engineering Group completed a detailed review and assessment of all their calculations. This assessment required a review of all design change documents, such as modifications, REAs, deficiency reports, etc. These calculations and design documents were sent to three separate engineering firms (vendors) for review and evaluation to determine if any calculations were affected. The Electrical Engineering Group had received and completed reviewing the initial vendor's work and was in the process of reviewing the final work packages from the vendors. An Evaluation Report was developed for each calculation. Each electrical calculation would not be revised at this time; however, it would have the completed Evaluation Report attached to it until it was revised. If an electrical calculation would be needed in the future, the design engineer would be required to review the attached Evaluation Report prior to its use. Three electrical calculations were completely reviewed and re-verified at this time by the licensee:

- E-91-0026, EGDG-1A Scenario Based Loading, Voltage Dip, Frequency Dip and Transient Motor Starting Analysis dated October 17, 1997

- E 93-0001, Vital Bus Inverter Maximum Steady State Operating Loads dated October 16, 1997

- E 96-0004, 500 KV Backfeed Voltage Drop, Load Flow, Motor Starting, Short Circuit and Parallel Operation Analysis dated August 21, 1997

The licensee was tracking this item as FPC Restart Issue OP-6.

c. Conclusions

This violation was closed based on it being completed prior to restart. It was being tracked as restraint items RMR 97-0354 (case studies) for Mode 2 and RMR 97-0418 (EGDG loading) for Mode 4 in the licensee's Restart Mode Restraint Report.

The inspectors concluded the licensee was in the process of implementing and had completed a sufficient amount of corrective action to closeout this violation. Violation EA-365, VIO B, Example 4 (02013) was 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 Design Basis - Adequate
- Compliance with Regulations - Adequate
- Operator Performance - N/A

E8.7 (Closed) VIO 50-302/95-21-03: Failure to Isolate the Class 1E From the Non-Class 1E Electrical Circuitry for the RB Purge Valves

(Closed) LER 50-302/95-025-00 and 02: Personnel Errors By Architect Engineer Result in Operation Outside Design Basis Due to Inadequate Safety/non-Safety Related Circuit Isolation

a. Inspection Scope (92903)

This violation involved the failure of the licensee to isolate Class 1E from non-Class 1E electrical circuitry for the reactor building purge and mini-purge valves. The inspectors followed up on the licensee's corrective action for this violation in conjunction with LERs 50-302/95-025-00, 01, an 02, which initially identified that the valves AHV-1A, 1B, 1C, and 1D were outside the design basis. Mini-purge valves LRV-70, 71, 72, and 73 were also identified as being outside the design basis concerning electrical isolation. The inspectors reviewed the licensee's corrective action implemented for this item to determine if it was in accordance with the licensee's commitments, requirements, procedures, and NRC regulations.

b. Observations and Findings

The licensee had completed implementing corrective action for two modification packages. Modification package MAR 91-08-26-04, Field Changes FC-1 and FC-2, MCB Separation Resolution, added isolation fuses in terminal box AH-325-TB at motor control center Vent MCC 3A and two isolation relays in the main control room. These fuses and relay contacts isolated the power from the non-safety radiation monitors for valves LR-70, 71, 72, 73, AHV-1B, and AHV-1C.

Modification package MAR 97-06-10-01, AHV-1A/1D Circuit Isolation, was implemented to provided similar corrective action for the other two purge valves. Isolation fuses were added to new terminal boxes AH-23-TB and AH-24-TB for valves AHV-1A and 1D, for the isolation from the non-safety related radiation monitors and from non-safety related DP switch contacts in AH-552-DPS and AH-266-DPS. The inspectors conducted a walkdown inspection and reviewed the completed "return to service" forms to verify the installation was completed for all the valves.

The licensee was tracking this item as FPC Restart Issue D-30.

c. Conclusions

This item was closed based on it being completed prior to restart. It was being tracked as restraint item RMR 97-0187 for Mode 4 in the licensee's Restart Mode Restraint Report.

The inspectors concluded the licensee had implemented corrective action to isolate the Class 1E electrical circuitry from the non-safety related circuitry for valves AHV-1A, 1B, 1C, 1D, LRV-70, 71, 72, and 73. Violation 50-302/95-21-03 and LERs 50-302/95-25-00 and -02 are closed.

(LER 50-302/95-25-01 was closed in IR 50-302/97-05.)

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 Design Basis - Adequate
- Compliance with Regulations - Adequate
- Operator Performance - N/A

E8.8 (Closed) IFI 50-302/96-201-17: Coordination of SLUR and Fuse Protection

a. Inspection Scope (92903)

An NRC inspection team identified a concern regarding fuse coordination, with the second level undervoltage relay (SLUR) trip setpoint. The team's concern was that a lack of fuse coordination to withstand delayed pickup of the SLUR could result in fuses blowing and multiple system failures could occur. In reviewing the coordination between the control circuit fuses and SLUR, the team found that there was neither margin for errors nor were important errors considered. Tolerances missing in the calculations were for fuses, which could result in errors up to 10%, and up to 10% for the SLUR in the 5-second time frame (5 sec. delay for SLUR). The inspectors reviewed the licensee's corrective action implemented for this item to determine if it was in accordance with the licensee's requirements, procedures, and NRC regulations.

b. Observations and Findings

The licensee initiated PC 96-3471 that recommended electrical calculation E-91-0012 be reviewed to determine the impact of fuse coordination on other electrical calculations. A Case Study, CSE-97-0003A, An Evaluation of Fuse Ratings Calculated by E-91-0012, was performed to evaluate the adequacy of the fuse ratings, using a tolerance of 10% on the time-current characteristic curves and a 10% tolerance on the SLUR time delay (5 sec). Electrical calculation E-91-0012, MCC/ACDP Safety Related Control Circuit Voltage Drop Calculation, was performed to evaluate the discrepancies on control power transformer secondary side fuses identified in PC 96-3471. The evaluation identified twenty fuses that were recommended for replacement as an enhancement. No safety concerns were identified with the fuses and the coordination with SLUR.

Modification package MAR 97-08-05-01, Calculation Fuse Modification, was initiated to change the twenty fuses. The licensee had completed the changing of the fuses for the B train. The changing of the A train fuses was in progress.

The licensee was tracking this item as FPC Restart Issue D-31.

c. Conclusions

This item was closed based on it being completed prior to restart. It was being tracked as restraint item RMR 97-0589 for Mode 2 in the licensee's Restart Mode Restraint Report.

The inspectors concluded the licensee had performed a satisfactory evaluation of fuse coordination with the SLUR and was in the process of implementing satisfactory corrective action. This IFI is closed.

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

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

E8.9 (Closed) LER 50-302/96-19-00, 01, and 02: Classification of Transfer Switch Causes Potential for Loss of Power to ES Status Lights

a. Inspection Scope (92903)

During the time Regulatory Guide (RG) 1.97 was being implemented, transfer switch ESCP-1 was not reclassified as safety-related. Consequently, a non-safety related switch in the main control room was not upgraded to a safety related classification. The issue was a non-safety related transfer switch, ESCP-1, was providing power to the safety related ES Status Light matrix. The inspectors reviewed the licensee's corrective action implemented for this item to determine if it was in accordance with the licensee's requirements, procedures, and NRC regulations.

b. Observations and Findings

Modification package MAR 97-01-03-01, ESCP-1 Upgrade, FCN-002, ESCP-1 Removal, was implemented to remove the transfer switch from the safety related circuit. The inspector reviewed "Return to service" and conducted a walkdown inspection to verify the switch was removed. ESCP-1 was not listed in any operating procedures, and therefore no procedure changes were required.

Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

The licensee was tracking this item as FPC Restart Issue D-21.

c. Conclusions

This item was closed based on it being completed prior to restart. It was being tracked as restraint item RMR 97-0187 for Mode 4 in the licensee's Restart Mode Restraint Report.

The inspectors concluded the licensee had satisfactorily removed transfer switch ESCP-1 from the safety-related circuits. LERs 50-302/97-19-00, 01, and 02 are closed.

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

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

E8.10 (Closed) LER 50-302/97-23-00: Design Engineering Process Allows Installation of Vital Bus Inverters Containing an Unanalyzed Trip Circuit

a. Inspection Scope (92903)

On July 30, 1997, the licensee discovered a trip coil in the control circuitry for vital bus inverters VBIT-1A/1C, direct current circuit breakers that were not analyzed when they were replaced during refueling outage 10. The inverter's DC input circuit breaker's low voltage trip could cause the source of power from the battery(s) to be interrupted. The inverters are designed to provide an uninterruptable 120 VAC vital bus power. The inspectors reviewed the licensee's corrective action implemented for this item to determine if it was in accordance with the licensee's requirements, procedures, and NRC regulations.

b. Observation and Findings

PC 97-5255 was initiated to identify that the new inverters, VBIT-1A and VBIT-1C, had a trip coil for the DC input breaker, which was not present on the old style inverters. Inverters VBIT-1A and VBIT-1C were replaced, using plant modification MAR 93-05-07-03.

Plant modification MAR 93-05-07-04 was developed for the replacement of inverters VBIT-1B and VBIT-1D. This modification had two other requirements: 1) provide alarm setpoints and 2) remove the DC breaker shunt trip circuits for all four inverters (VBIT-1A, 1B, 1C, and 1D) by removing the fuses and associated wiring.

The inspectors reviewed the modifications package to ensure the documentation was in accordance with the licensee's procedures. A walkdown inspection was conducted to verify that the DC input breaker's shunt trip circuits were disconnected as design. The installation was completed for VBIT-B and VBIT-D, and both inverters were back in operation. The fuses were disconnected for VBIT-A and VBIT-C. However, the modification was not quite complete at the time of the walkdown.

Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

The licensee was tracking this item as FPC Restart Issue D-78.

c. Conclusions

This item was closed based on it being completed prior to restart. It was being tracked as restraint item RMR 97-0333 for Mode 4 in the licensee's Restart Mode Restraint Report.

The inspectors concluded the licensee had implemented adequate corrective action to disconnect the DC input breaker's shunt trip circuits from inverters VBIT-1A, 1B, 1C, and 1D.

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

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

E8.11 (Closed) LER 50-302/97-32-00: Inadequate Electrical Isolation on a Safety Related Power Supply Due to a Design Error

a. Inspection Scope (92903)

The licensee identified that electrical isolation was not provided between the safety-related indication and the non-safety-related circuit for solenoid valve RW-63-SV. Solenoid valve RW-63-SV, which is powered from vital bus distribution panel VBDP-4, breaker 8, is used to test the emergency nuclear services seawater pump RWP-2B. The inspectors reviewed the licensee's corrective action implemented for this item to determine if it was in accordance with the licensee requirements, procedures, and NRC regulations.

b. Observations and Findings

Plant modification package MAR 97-11-01-01, Circuit Isolation for RW-63-SV, was developed to provide for the installation of an isolation fuse. The fuse is to be installed in the ESF-B main control board (MCB). The inspectors reviewed the MAR and concluded it was adequate to provide the design requirements and instructions for the installation of the required isolation fuse. The required field work is relatively minor. No field work was started at the time of this inspection.

Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

The licensee was tracking this item as FPC Restart Issue D-30A.

c. Conclusions

This item was closed based on it being completed prior to restart. It was being tracked as restraint item RMR 97-0420 for Mode 4 in the licensee's Restart Mode Restraint Report.

The inspectors concluded it would be acceptable to close this LER based on the fact that the MAR was adequate and the licensee had this item scheduled on their Restart Mode Restraint list.

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

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

E8.12 (Closed) LER 50-302/97-21-00 and 01: Loss of A Battery Leads to the Inability to Bypass ES Actuation Signals as a Result of Inadequate System Knowledge

a. Inspection Scope (92903)

During the licensee's review of the Class 1E DC Failure Modes and Effects Analysis (FMEA), a deficiency with the failure of the A battery was identified. A loss of the A battery bank (250 VDC) during a Loss of Offsite Power/Loss of Coolant Accident (LOOP/LOCA) would have prevented operations from bypassing the B ES actuation matrix. This condition would have invalidated some of the accident mitigation strategies where ES equipment must be manually manipulated later during an event scenario. It would also have prevented the starting and loading of EDG-1A and all of train A power. The inspectors reviewed the licensee's corrective action implemented for this item to determine if it was in accordance with the licensee's requirements, procedures, and NRC regulations.

b. Observations and Findings

The licensee identified this issue during the Class 1E DC FMEA evaluation. The problem was loss of train A power to the ES cabinets (1 & 2) containing the power supplies for the RC pressure transmitters (HPI & LPI), bistables, and permissive bistables during a LOOP/LOCA. The ES system has three channels (1, 2, and 3), where each channel has a separate test and relay cabinets for the RC pressure transmitters. Train A vital bus 3A powers part of channel 1 and vital bus 3C powers all of channel 2. Train B vital bus 3D powers part of channel 1 and all of channel 3.

The ES channel logic was arranged to obtain a 2-out-3 actuation for each train. Therefore, loss of all train A power would result in a 2-out-3 actuation for channel A and a 1-out-3 actuation for channel B. If there would be a loss of train A power, the channel 1 bypass permissive bistable would not have power and could not be placed in "Bypass" mode. This would prevent bypassing the ES actuation matrix and prevent the manual operation of ES equipment

Plant modification MAR 97-08-12-01, Loss of A Battery ES Modification, was designed to install a relay to monitor the channel 1 train A power. The new relay contacts were placed in parallel to the existing "bypass permissive bistable" contacts. On loss of train A power the new relay would de-energize and its contacts would close. This would permit the use of the "bypass" function. A test switch was also installed to allow for the testing of the new relay.

The inspectors reviewed the modification package, the field work package, and the post modification test procedure, Functional Test - Loss of A Battery ES Modification. A walkdown inspection was performed. The licensee was in the process of implementing the work and the functional test had not been performed. However, the inspectors concluded that this corrective action would be satisfactory when the work was complete and the functional test satisfactorily performed.

Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

The licensee was tracking this item as FPC Restart Issue D-7A.

c. Conclusions

This item was closed based on it being completed prior to restart. It was being tracked as restraint item RMR 97-0262 for Mode 4 in the licensee's Restart Mode Restraint Report.

The inspectors concluded the licensee was in the process of implementing corrective action. The corrective action would be adequate when the installation and functional test were satisfactorily completed.

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

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

E8.13 (Open) VIO 50-302/97-14-13: Failure to Take Corrective Actions to Identify and Correct the Design Weaknesses Associated with Adequacy of the Past 10 CFR 50.59 Review for Positioning of DHV-34 and DHV-35 During Normal Operation

a. Inspection Scope (92903)

The inspectors examined the licensee's Operability Determination and Safety Assessment of maintaining the decay heat valves DHV-34 and DHV-35 in the "closed" position during normal operation. The "justification for continued operation" with DHV-34 and DHV-35 maintained closed was reviewed to evaluate the impact on the Decay Heat Removal System and the Reactor Building Spray System.

b. Observations and Findings

In 1976, before commercial operation, FPC submitted correspondence to the NRC which committed the plant to maintain DHV-34 and DHV-35 in the "open" position during normal operation. In 1985, FPC revised operating procedures to maintain DHV-34 and DHV-35 in the "closed" position. This revision was based on issues raised during an Appendix R fire study. During the SSFI conducted in October 1997, the NRC identified a concern regarding the DHV valves. Violation 50-302/97-14-13 was cited because decay heat valves DHV-34 and DHV-35 were being maintained in the "closed" position without an adequate safety evaluation. This violation was discussed in detail in IR 50-302/97-14.

The licensee issued PC 97-7755 to address the violation. They also performed a Safety Assessment, an Unreviewed Safety Question Determination (USQD) per 10 CFR 50.59, and an Operability Determination.

The results were as follows:

- Maintaining DHV-34 and DHV-35 closed was contrary to a commitment made to the NRC in 1976.

- Maintaining DHV-34 and DHV-35 in the closed position constituted a USQ.

- The licensee will apply for a license amendment to allow DHV-34 and DHV-35 to remain closed.

- The probability of malfunction is increased by the valves having to stroke open as opposed to being normally open. These are highly reliable valves, and the probability of not opening on demand is quite low.

- The safety function of the valves is to "open" to provide a flow path and suction source from the BWST to the ECCS during a LOCA.

The inspectors verified the following for DHV-34 and DHV-35:

- Both valves are the gate type and are safety-related.

- Both valves are in the Generic Letter (GL) 89-10 MOV program. The safety margin for thrust was 106 percent or twice as much thrust as required to open the valves under the worst case condition of a maximum pressure of 53 psig.

- Drawing 208-021, Sheets DH-11 and DH-12, required that both valves have their torque switch bypassed for the first 25 percent of travel. This provides maximum output thrust to initially crack open the valves.

- DHV-34 is in Surveillance Procedure SP-340B, DHP-1A, BSP-1A and Valve Surveillance. DHV-35 is in Surveillance Procedure SP-340E, DHP-1B, BSP-1B and Valve Surveillance. The purpose of these two procedures was to provide assurance that the decay heat pumps, reactor building spray pumps, and associated safety-related valves are operable by performing functional testing on a quarterly basis in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, subsections IWP and IWV. These procedures provide for compliance with the Inservice Testing Program.

- Plant modification MAR 97-06-014-01, Remote Shutdown Circuits Reroutes, was implemented to re-route 33 cables away from fire areas as specified in Appendix R fire study. Cable DHC-44 for valve DHV-34 and cable DHC-49 for valve DHV-35 were included in re-routes.

- Plant modification MAR 97-06-13-01, Appendix R MOV Rewire, was implemented to address the concerns in IN 92-18, Potential For Loss Of Remote Shutdown Capability During A Control Room Fire. Both valves were included in this modification.

The inspectors did not identify safety concerns with the valves being maintained closed. No negative impact with the Decay Heat Removal System or the Reactor Building Spray System was identified by the inspectors.

c. Conclusions

The inspectors concluded that Valves DHV-34 anc DHV-35 were being well maintained in the GL 89-10 program and adequately tested on a quarterly surveillance basis to assure proper operation. The cables to the valves have been re-routed for fire protection and the valves re-wired to prevent hot shorts identified in IN 92-18. The inspectors concluded there were no safety concerns with the "justification for continued operation with DHV-34 and DHV-35 maintained closed during normal operation". The licensee's corrective actions were acceptable for restart and the long term resolution will be addressed by the license amendment.

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

E8.14 (Closed) EA 97-330 (01013): Unreviewed Safety Qu[illegible]n 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 had reviewed the licensee's partially completed corrective actions for this violation, documented the results of that inspection in IR 50-302/97-17, and left the violation open for further review of the licensee's resolution of the unreviewed safety question and installation of MAR 97-08-01-01. During this inspecti[illegible] the inspectors reviewed the licensee's completed MAR package and completed post-modification test for the B EDG. The inspectors also reviewed the NRC letter dated December 1, 1997, which approved the licensee's new EDG protective relaying scheme.

b. Observations and Findings

The inspectors noted that the NRC approval letter appropriately described the licensee's new EDG protective relaying scheme. Also, the licensee's completed post-modification test appropriately tested the modified circuits on the B EDG. The licensee was tracking this item as FPC Restart Issue D-12, to ensure that the modification was installed and tested on the A EDG prior to plant restart.

c. Conclusions

The inspectors concluded that the licensee's corrective actions for this violation were good. EA 97-330 (01013) 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 - Not Applicable

E8.15 (Closed) LER 50-302/97-05-00: Unanalyzed Condition Regarding Small Break LOCA and Emergency Feedwater

a. Inspection Scope (92903)

This LER addressed a design condition which could result in inadequate cooling to the reactor core during a small break loss of coolant accident concurrent with a loss of offsite power and a failure of the motor-driven emergency feedwater pump. The design condition of concern was an automatic trip of the motor-driven emergency feedwater pump at 500 psig RCS pressure. This trip had been installed to remove sufficient load from the A EDG to allow starting the A low pressure injection pump without overloading the A EDG. However, if the turbine-driven EFW pump was unavailable when the motor-driven EFW pump tripped, there could be inadequate core cooling until the RCS pressure dropped below the discharge pressure of the low pressure injection pumps (about 180 psig). The inspectors followed up on the licensee's corrective actions.

b. Observations and Findings

The modification that installed the 500 psig trip of the EFW pump was addressed in EA 96-365, 96-465, 96-527 Violation B (02013), Failure to Update Applicable Design Documents to Incorporate EFW Design Information. The licensee's resolution of that issue was inspected and closed in IR 50-302/97-17. The inspectors verified that the licensee had kept the plant shut down to resolve this and other design issues. Also, the inspectors verified that the licensee had conducted an engineering "stand down" to emphasize the importance of improving safety culture. In addition, the inspectors verified that the licensee had increased engineering staffing levels and had hired experienced personnel from outside FPC. The power upgrade for the A EDG has been inspected and will be addressed separately under NRC restart item CR3 Design Issue 6, Emergency Diesel Generator Loading. Also, the EDG power upgrade and EDG load management strategies were included in TSCRN 210, which was being reviewed by the NRC prior to plant restart. In addition, EDG load management procedures were being inspected prior to plant restart by the NRC emergency operating procedures (EOP) inspection team.

c. Conclusions

The inspectors concluded that the licensee's corrective actions were either adequately completed, included in another open restart item, or being addressed by another NRC inspection prior to restart. LER 50-302/97-05-00 is closed.

The inspectors assessed the licensee's performance, relative to this event, 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 - Not Applicable

E8.16 (Closed) LER 50-302/95-09-00: Minimal Release During Sulfur Dioxide Delivery Causes Actuation of Toxic Gas Monitor Resulting in Control Room Emergency Ventilation Actuation

a. Inspection Scope (92903)

This voluntary LER involved the automatic actuation of the control room emergency ventilation system (CREVS) during the receipt of a delivery of sulfur dioxide by the CR-1&2 fossil units. Connection of the sulfur dioxide tank fill hose had actuated the toxic gas alarm; however, there was no continuous release of sulfur dioxide. Control room operators had followed their abnormal procedure and donned self contained breathing apparatus (SCBAs) until they verified that there was no sulfur dioxide release in progress. The inspectors followed up on the licensee's additional corrective actions.

b. Observations and Findings

The inspectors verified that the licensee had placed controls in place to avoid unnecessary CREVS actuations and donning of SCBAs. The controls included requiring communication between the CR-3 control room and CR-1&2 personnel during sulfur dioxide deliveries and placing the CREVS in recirculation during the deliveries. Also, the licensee had reduced the maximum amount of sulfur dioxide stored at CR-1&2 and analyzed that, with the reduced amount of sulfur dioxide storage, CR-3 operators did not need to don SCBAs in the event of a toxic gas alarm. In addition, the licensee is planning to remove the sulfur dioxide from CR-1&2 and replace it with a pelletized emission control system.

c. Conclusions

The inspectors concluded that the licensee's corrective actions for this event were adequate. LER 50-302/95-09-00 is closed.

The inspectors assessed the licensee's performance, relative to this event, 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

E8.17 (Closed) LER 50-302/96-04-00; Control Complex Habitability Envelope (CCHE) Control Dampers Found Damaged and Leaking

a. Inspection Scope (92903)

This LER involved the discovery, while the plant was shut down, that CCHE damper AHD-2 had a damaged blade and AHD-12 had missing seals. The additional damper leakage paths exceeded the allowable CCHE leakage for plant operation. Had a design basis event occurred, when the plant had been operating with these damaged dampers, the control room operator dose limits could have been exceeded. These dampers had not been previously inspected for leakage and were being inspected for the first time in January 1996 in response to NRC inspector inquiries. The inspectors followed up on the licensee's corrective actions for this event.

b. Observations and Findings

The inspectors verified that the licensee had repaired the damaged dampers. The inspectors also verified that the licensee had added periodic inspection of the dampers to the preventive maintenance program. In addition, the inspectors verified that the licensee subsequently replaced all of the old CCHE dampers with new bubble-tight dampers to substantially reduce the CCHE leakage (and potential operator dose during an accident). NRC inspection of the damper damage is described in Inspection Report 50-302/95-16. Licensee installation of the new dampers is described in this report in paragraph E8.18.

c. Conclusions

The inspectors concluded that the licensee's corrective actions for the leaking CCHE dampers were adequate. Although this item is a noncompliance with regulatory requirements, for the reasons discussed in

Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

The inspectors assessed the licensee's performance, relative to this event, 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 - Not Applicable

E8.18 (Open) URI 50-302/95-02-02; Control Room Habitability Envelope Leakage

(Closed) LER 50-302/97-22-00 and 01; Calculation Errors Associated With Control Complex Habitability Envelope Unfiltered Air Inleakage Could Allow Operator Dose Limits to be Exceeded

a. Inspection Scope (92903)

This unresolved item was identified in January 1995 based on inspector concerns about excessive leakage paths by CCHE doors, lack of CCHE integrity testing or other surveillances, the presence of large amounts of toxic gas at the nearby fossil units, and a licensee position that CCHE integrity did not affect CREVS operability.

Subsequently, additional problems with CCHE integrity (doors, dampers, drains, electrical penetrations, and calculations) were identified and documented in the following:

- IR 50-302/95-09
- IR 50-302/95-11
- IR 50-302/95-16
- IR 50-302/95-21
- IR 50-302/97-05
- IR 50-302/97-07
- IR 50-302/97-13
- IR 50-302/97-17
- LER 50-302/95-01-00
- LER 50-302/95-04-00
- LER 50-302/95-09-00
- LER 50-302/96-04-00
- LER 50-302/97-22-00, 01

LER 50-302/97-22-01 described four examples of calculation errors that resulted in the CCHE being outside its design basis. The inspectors reviewed the licensee's corrective actions for this LER. In addition, the inspectors reviewed the cumulative past effect of the identified problems with CCHE integrity on CREVS operability.

b. Observations and Findings

The inspectors verified that the licensee had accomplished the corrective actions stated in LER 50-302/97-22-01. The licensee submitted a TS Change Request to revise charcoal absorber laboratory test requirements for humidity and committed to perform the testing prior to plant restart, using both the existing TS testing method and the proposed new TS testing method. The licensee completed new calculations for control room operator dose. Also, the licensee replaced the CCHE boundary dampers with bubble-type dampers, installed two additional dampers to provide redundancy at boundary locations, sealed about 400 electrical cable conduits that penetrate the CCHE boundary and committed to seal electrical cable trays that penetrate the CCHE boundary during the first half of 1998. Additionally, the licensee revised four engineering procedures to prevent recurrence of the identified errors in calculations.

The inspectors assessed past leakage of CCHE penetrations based on the results of recent licensee inspections and testing. As described in IR 50-302/97-17, inspector analysis of the licensee's CCHE test data from October 1997 found that CCHE leakage, calculated by using the NRC Standard Review Plan (SRP) method, would be approximately 626 CFM. Of the 626 CFM, 10 CFM had been allowed for the opening and closing of doors, zero leakage was estimated for damper leakage, and licensee analysis based on laboratory testing allowed approximately 110 CFM for door leakage. Therefore, the balance of approximately 506 CFM was due to penetration leakage.

In addition, just prior to the October 1997 test, the licensee had sealed approximately 400 electrical conduits that penetrated the CCHE boundary. (The licensee had observed air leaking through the conduits with only a small differential pressure across them.) If the 400 conduits averaged approximately four inches in diameter and approximately 5% of the conduit area was a leakage path prior to sealing, that would represent: 400 x .05 x (3.14 x 4) = 251 square inches of leakage paths. At an SRP leakage rate of approximately 2.96 CFM per square inch: 2.96 x 251 = 743 CFM of potential SRP leakage through the conduits. While that seems very high, the inspector concluded that the conduit leakage paths were potentially very substantial.

The licensee's Control Room Habitability Evaluation Report, submitted to the NRC on June 30, 1987, stated that the operator dose analysis assumed a maximum 355 CFM of total CCHE SRP leakage, of which 245 CFM would be through penetrations and doors. The CCHE doors installed in the plant at that time leaked substantially more (by inspector visual examination and comparison) than the new replacement doors installed in 1995. Since laboratory tests showed that the new double doors leaked at least 36 CFM each at a differential pressure of .125 in. w.g., the inspector estimated that the old doors had an SRP leakage of at least 40 CFM each. The three old double doors thus would have leaked about 120 CFM, and adding approximately 2 CFM for the three single CCHE doors would result in a total old door leakage of approximately 122 CFM. The total SRP leakage for the CCHE penetrations and old doors would have been approximately 506 + 122 = 628 CFM plus potentially several hundred CFM for conduits sealed just prior to the October 1997 CCHE test. The inspectors concluded that the actual CCHE SRP leakage due to doors and penetrations had exceeded that in the operator dose analysis by over 628 - 245 = 383 CFM.

The licensee's Control Room Habitability Evaluation Report also stated that the operator dose analysis assumed that CCHE SRP leakage through dampers would be less than 100 CFM, of which 70 CFM was filtered leakage and thus would result in very little operator dose. However, as discussed in LER 50-302/97-22-01, the leakage through dampers would have actually been approximately 197 CFM, all unfiltered.

In summary, calculation errors resulted in the licensee stating to the NRC in June 1987 that total CCHE leakage, calculated by the prescribed SRP method, was less than 355 CFM when in hindsight it would have been substantially more than 628 (doors and penetrations), plus 197 (dampers), plus 10 (opening of doors) or a total of over 835 CFM. Using the 355 CFM, the licensee had calculated an operator post-accident thyroid dose of 26.5 Rem, which was just under the regulatory limit of 30 Rem. In December 1997, the licensee performed a new operator dose calculation, using new calculation methods (i.e., using ICRP 30 dose factors vs. ICRP 2; and using Murphy-Campe wind loading vs. the SRP method.) Use of these new calculation methods effectively reduced the calculated operator dose by approximately 50%. The licensee's new calculation concluded that, after completion of the CCHE modifications to reduce inleakage to the amount measured in the October 1997 CCHE tests, the calculated operator thyroid dose would be 26.5 Rem. The inspectors concluded that an operator dose calculation based on over 835 CFM of CCHE leakage, using the old methods or the new (unapproved) methods, would have exceeded the 30 Rem limit on operator thyroid dose.

In addition, the CCHE boundary dampers AHD-2, AHD-12, and AHD-99 were each single dampers and thus susceptible to single failure. The licensee had no procedures for emergency repair of these dampers (The licensee had stated to the NRC that they would have such procedures and the NRC SER approving the licensee's CCHE design had relied upon such procedures.) (See IR 50-302/97-05.)

Also, the CCHE boundary had not been maintained. Dampers had not been inspected for leak tightness prior to 1996 and then were found to have excessive leakage. (See IR 50-302/95-16 and LER 50-302/96-04-00.) Door inspections and maintenance had been inadequate for maintaining leak tightness. (See IR 50-302/95-02, IR 50-302/95-16, and LER 50-302/95-01-00.)

c. <u>Conclusions</u>

The inspectors concluded that the licensee's corrective actions, for the incorrect calculations related to the CCHE and CREVS, were adequate. LER 50-302/97-22-00 (including supplement 01) is closed.

The inspectors concluded that, as a result of calculation inadequacies, procedure inadequacies and lack of past CCHE leakage testing, the CCHE and CREVS had not been operable in the past. The inadequacies constitute a violation of NRC design control requirements, with nine examples:

1) Incorrect calculations for CCHE boundary door leakage (NRC identified, documented in IR 50-302/95-09 and above.)

2) Incorrect calculations for CCHE boundary damper leakage (NRC identified; documented in IR 50-302/97-05, IR 50-302/97-07, and LER 50-302/97-22-01.)

3) Incorrect calculations for CCHE boundary penetration leakage (NRC identified; documented in IR 50-302/97-17 and above.)

4) Incorrect calculations for available CCHE breach margin (licensee identified, documented in LER 50-302/97-22-01.)

5) No calculation for humidity of charcoal filtered air (licensee identified, documented in LER 50-302/97-22-01.)

6) Calculations not revised to reflect reduced CREVS air flow caused by cooling coil baffles installed by MAR 77-11-04 (licensee identified, documented in IR 50-302/97-05 and LER 50-302/97-22-01.)

7) No procedures for emergency repair of three CCHE boundary dampers susceptible to single failure (NRC identified, documented in IR 50-302/97-05.)

8) Inadequate procedures for periodic inspection and maintenance of CCHE boundary door seals (NRC identified, documented in IR 50-302/95-02, IR 50-302/95-16, and LER 50-302/95-01-00.)

9) No procedures for periodic inspection and maintenance of CCHE boundary damper blades and seals (NRC identified, documented in IR 50-302/95-16 and LER 50-302/96-04-00.)

Although these items are noncompliances with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, the licensee meets the criteria for enforcement discretion per Section VII.B.2 of the NRC Enforcement Policy as described in NUREG-1600. Consequently, these noncompliances are identified as more examples of Non-cited Violation NCV 50-302/97-21-01, Examples of Noncompliances in Design Control, 50.59 Evaluations, Procedure Adequacy, Reportability.

and Corrective Actions That Are Subject to Enforcement Discretion. URI 50-302/95-02-02 remains open for licensee post-modification testing of the CREVS, for licensee testing of the charcoal filter material, and for NRC review of the testing.

The inspectors assessed the licensee's performance relative to URI 50-302/95-02-02, in the five areas of continuing NRC concern:

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

E8.19 <u>(Closed) VIO 50-302/96-09-05; Failure to Incorporate Design Information into Operations Procedures</u>

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

This violation involved the licensee's failure to revise operations procedures or provide training to operators (to incorporate design information from the design input record of MAR 95-01-07-01) after modifying makeup valve MUV-64, changing the valve operator from a disabled air operated valve (locked in the open position) to manual operation with a manual gear driven chain operator. The inspectors followed up on the licensee's corrective actions by reviewing procedure changes, internal licensee correspondence, and interviewing engineering personnel.

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

The inspectors previously reviewed this item and documented the results in IR 50-302/97-11. During previous reviews of this VIO, the inspectors noted that the corrective actions implemented to address this VIO were being tracked under licensee restart item OP-27. The inspectors verified that the corrective actions, stated in the licensee's response to this VIO dated March 18, 1997, had been implemented. Corrective actions reviewed by the inspectors included licensee interoffice correspondence (IOC) NOE97-0228, dated March 14, 1997, which corrected the design input requirements (DIR) for MAR 95-01-07-01. The inspectors also reviewed revisions and enhancements to various Nuclear Engineering Procedures (NEPs).

During previous followup of this item, the inspectors noted that the licensee's March 18, 1997 response indicated that the design and licensing basis for valve MUV-64 warranted clarification and the corrective actions included steps to provide the clarification. The inspectors questioned licensee personnel as to whether any documentation (in the form of a licensing submittal) had been submitted to the NRC Office of Nuclear Reactor Regulation (NRR), which included the clarifications to the licensing and design basis for MUV-64. The inspectors also questioned whether the licensing submittal (if submitted by the licensee) had been reviewed by NRR and if NRR had issued a safety evaluation report (SER) for the changes to the licensing and design basis for MUV-64. The licensing submittal to NRR would be in addition to the licensee's response to VIO 50-302/96-09-05. Licensee personnel indicated to the inspectors that a licensing submittal had not been submitted to NRR and the questions raised by the inspectors required further review to determine if additional documentation needed to be submitted to NRR for review. This item was left open pending the inspector's verification that appropriate documentation had been submitted to NRR for review, describing the changes and clarifications made to the licensing and design basis for valve MUV-64.

During this current inspection, the inspectors noted that the licensee had submitted a letter (3F1197-20) to the NRC dated November 11, 1997. The letter stated that FPC had determined that it was not necessary to manually close valve MUV-64 within 10 minutes following an ES actuation. The letter also described proposed revisions to the Final Safety Analysis Report (FSAR) for the design and licensing basis for MUV-64. The licensee concluded in this letter that elimination of the requirement for MUV-64 to close within 10 minutes following an ES actuation did not constitute an unreviewed safety question. The licensee also requested that the NRC review the conclusions of letter 3F1197-20.

During review of the resolution of this violation, the inspectors reviewed PC 97-5035, which discussed a potential problem with the level instrumentation for the makeup tank (MUT) in relation to RG 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident. Makeup tank level was a RG 1.97 Type D, Category 2 variable. RG 1.97 requires that instrumentation covered by the guide be capable of indicating over the full range of the measured variable. PC 97-5035 questioned whether the "full range" criterion was met in light of the fact that the tank was expected to empty during certain design basis accident scenarios. The NRC had previously reviewed the tank emptying issue through review of licensee calculation M94-0053, which was developed to support the pressure-level curve for the MUT. Thus, that aspect was not reviewed during this inspection. The PC question had not yet been answered by the licensee. The inspectors reviewed the Post Accident Monitoring Instrumentation Design Basis Document, NRC SER for the licensee's RG 1.97 instrumentation which was issued June 16, 1987, and the function of MUT level indication in terms of post accident monitoring requirements. The inspectors found that, since the MUT outlet valve was intended to remain open following an accident, the only post-accident function of the level instrumentation was to provide status on operation of the makeup and letdown system should that system be needed during the recovery phase. The SER stated that MUT level was the primary variable for this function. The fact that the MUT could empty during certain accident scenarios did not affect the ability of the level instrumentation to fulfill the function stated above. Therefore, the inspectors considered that the PC 97-5035 question was resolved. The licensee had yet to document this conclusion within their corrective action program, but that could be done after restart of the plant.

The inspectors informed the licensee that, based on the licensee's submittal of letter 3F1197-20 for NRC review, VIO 50-302/96-09-05 is closed.

c. Conclusions

The inspectors concluded that the corrective actions for this violation had been completed by the licensee and the violation is closed based on the licensee's submittal of letter 3F1197-20 for NRC review.

The inspectors 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

E8.20 (Closed) VIO 50-302/96-09-07: Inadequate Corrective Action for Implementation of EFIC Task Force Recommendations

a. Inspection Scope (37550, 92903)

This violation involved failure of the licensee to take adequate and timely corrective actions to implement recommendations from the Emergency Feedwater Initiation and Control (EFIC) task force. The inspector followed up on the licensee's corrective actions for this violation.

b. Observations and Findings

The inspector had reviewed this item previously and documented the results in NRC IR 50-302/97-01. The inspector reviewed the corrective actions specified in the licensee's response to this violation. The inspector noted that the corrective actions implemented to address this violation were being tracked under licensee restart item OP-28. The inspector reviewed these corrective actions for compliance with the FSAR, TS, and applicable licensee procedures. The inspector noted that some of the corrective actions specified in the response had been implemented. Corrective actions implemented included: 1) procedure revisions requiring all requests for engineering assistance (REA), which requested a plant modification, to be reviewed and approved by the Plant Modification Review Group (PMRG); 2) a list of high priority modifications was being maintained by the PMRG; 3) high priority EFIC/EFW issues were being addressed during the present shutdown; and 4) additional resources (permanent and contract personnel) were added to the engineering organization to ensure that high priority tasks were being worked.

During this current inspection, the inspector noted that modifications to address the high priority EFIC/EFW issues had been implemented. This included the EFIC Task Force recommendation discussed in PC 97-0595 regarding the EFIC control module upgrade. The inspector further noted that some of the EFIC Task Force recommendations that were not included on the licensee's restart list had been evaluated by the restart panel as being acceptable for deferral until refueling outage (RFO) 11. The inspector verified that the EFIC Task Force recommendations which were not implemented during this current outage had been approved and funded for implementation in 1998. This item is closed.

c. Conclusions

The inspector concluded that the licensee's implementation of the corrective actions specified in the response to this violation were satisfactory. This violation is closed.

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

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

E8.21 (Open) VIO 50-302/97-02-03: Adequacy of Procedures to Take the Plant from Hot Standby to Cold Shutdown from Outside the Control Room

a. Inspection Scope (92903)

This violation involved a concern identified by the NRC where the licensee did not have procedures in effect which provided adequate instructions for taking the plant from hot standby to cold shutdown from outside the main control room during an Appendix R fire.

b. Observations and Findings

The licensee provided the corrective actions for this violation in a letter to the NRC dated May 23, 1997. The inspector noted that the corrective actions implemented to address this violation were being tracked under licensee restart item OP-19A. The inspector reviewed the corrective actions for compliance with 10 CFR 50 Appendix R, the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS), NRC SER requirements, the FSAR, and licensee procedures. The inspector reviewed AI-400F, New Procedures and Procedure Change Processes for EOPs, APs, and Supporting Documents, Revision 4, dated June 16, 1997. The inspector verified that AI-400F (Enclosure 7, Technical Review Checklist) included requirements to review procedure changes to determine if any of the proposed changes affected any of the assumptions made in the Appendix R fire study.

In addition to reviewing the corrective actions specified in the licensee's response to this violation, the inspector also reviewed the corrective actions specified in PC 97-1522. These corrective actions included but were not limited to changes to Abnormal Procedure AP-990, Shutdown from Outside Control Room; changes to Operating Procedure OP-880, Fire Service System, and changes to the CR-3 Fire Study. The inspector reviewed the status of these corrective actions and noted that the corrective actions had not been completed. Abnormal Procedure AP-990 was still in the process of being revised to include instructions for taking the plant from hot standby to cold shutdown. The inspector also noted that Procedure OP-880 was in the process of being revised and the licensee was preparing an interim change to the CR-3 Fire Study. The changes to Procedures AP-990 and OP-880 and the change to the CR-3 Fire Study had not been completed at the conclusion of this inspection. These changes were scheduled to be completed prior to restart.

During discussions of the status of Procedure AP-990, the inspector raised a question regarding the number of operations personnel needed to perform AP-990 and the number of operations personnel needed for the fire brigade. The inspector reviewed applicable sections of the CR-3 ITS, the CR-3 Fire Protection Plan, Administrative Instruction AI-500, Conduct of Operations, and Administrative Instruction AI-2205, Administration of CR-3 Fire Brigade Organization to determine the operations minimum shift staffing and the fire brigade staffing requirements. During this review, the inspector questioned whether Administrative Instruction AI-2205 was consistent with the CR-3 Fire Protection Plan with regard to minimum shift staffing and fire brigade staffing, relative to performance of procedure AP-990. The inspector will review this question regarding minimum shift staffing and fire brigade staffing during a subsequent inspection to follow up on the corrective actions for this violation.

c. <u>Conclusions</u>

The inspector concluded that all the corrective actions specified for this violation had not been completed at the conclusion of this inspection. Specifically, the revision to Procedure AP-990 had not been completed and was not available for review by the inspector. This item remains open.

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

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

E8.22 <u>(Open) VIO 50-302/97-16-03: Failure to Design and Install Radioactive Waste Disposal System Piping as Described in the FSAR</u>

(Open) LER 50-302/97-38-00: An Engineering Oversight Resulted in Operation Outside of the Design Basis for the Waste Disposal System

a. Inspection Scope (37550, 92903)

This violation and LER involved the licensee's failure to design and install portions of the radioactive waste disposal system (WDS) piping as described in the FSAR. The inspector reviewed the status of the licensee's corrective actions to address this violation.

b. Observations and Findings

The inspector noted that the licensee was in the process of preparing their response to this violation at the time of this inspection. The licensee issued LER 50-302/97-38-00, on November 22, 1997, to address aspects of this issue. The inspector noted that the corrective actions for this violation were being tracked under licensee restart item D-51A. Corrective actions specified in LER 50-302/97-38-00 included the following:

- Upgrading the liquid outlet piping to Seismic Class I prior to restart from the next scheduled refueling outage (11R) for the waste gas decay tanks (WGDT), miscellaneous waste storage tank (MWST), spent resin storage tank (SRST), and neutralizer tank.
- Development of a JCO for the WDS by FPC, consistent with NRC Generic Letter 91-18, Revision 1, prior to entering Mode 4 from the current outage.
- Reactor coolant drain tank (RCDT) process piping and liquid outlet piping were being evaluated by FPC, in accordance with 10 CFR 50.59, as a change from seismic to non-seismic.

The inspector noted that the licensee had prepared and issued a JCO and 10 CFR 50.59 safety evaluation for applicable portions of the WDS liquid outlet piping. The 50.59 safety evaluation determined that an unreviewed safety question (USQ) existed with the MWST, SRST, and neutralizer tank not fully meeting the seismic requirements of FSAR Section 5.1.1.1. The operability determination performed in accordance with GL 91-18, Revision 1, provided justification for continued operation of the WDS until the piping could be modified and upgraded to meet the FSAR seismic requirements. The inspector also noted that the licensee had prepared and issued a 10 CFR 50.59 safety evaluation to change the RCDT process piping and liquid outlet piping from seismic to non-seismic. Engineering had submitted documentation to update the FSAR via interoffice correspondence NOE97-1820 dated November 14, 1997. The inspector considered the corrective actions implemented for LER 50-302/97-38-00 were acceptable for restart.

In addition to the corrective actions specified in LER 50-302/97-38-00, the inspector noted that corrective actions in progress at the time of this inspection included implementation of modification field work to upgrade the WGDT piping from Seismic Class III to Seismic Class I. The WGDT piping upgrade was scheduled to be completed prior to CR-3 restart. In an effort to complete the WGDT piping upgrade prior to restart from the current outage, the licensee had initiated the modification field work prior to issuance of MAR 97-10-01-01. The inspector reviewed the CR-3 ITS and various administrative and design control procedures to determine if the modification field work had been implemented in accordance with these controls. The inspector determined that, as required by the CR-3 ITS, the modification field work had been approved by the Director, Nuclear Plant Operations (DNPO) prior to implementation. The inspector also reviewed recent revisions to administrative instruction AI-602, MAR Work Package Preparation, Implementation, and Closure; and Nuclear Engineering Procedure (NEP) 212, Processing of Modification Projects by Nuclear Projects. The inspector determined that AI-602 allowed modification field work to begin (with the approval of the DNPO and the Director, Nuclear Engineering and Projects) prior to issuance of the MAR. During further review of NEP-212, the inspector noted that there were some sections of the NEP which appeared to be inconsistent with AI-602 regarding implementation of MAR field work prior to issuance of the MAR. The inspector discussed this inconsistency with licensee engineering personnel who indicated that, although NEP-212 was revised (Revision 21 dated November 21, 1997) to address implementation of modification field work prior to MAR issuance, some of the procedure sections needed additional clarification to ensure that NEP-212 was in agreement with AI-602. Engineering personnel initiated PC 97-8373 to resolve this inconsistency between NEP-212 and AI-602. The inspector considered the corrective actions being implemented for the WGDT piping acceptable for restart.

c. <u>Conclusions</u>

The inspector concluded that the corrective actions which had been implemented for VIO 50-302/97-16-03 and LER 50-302/97-38-00 at the conclusion of this inspection were acceptable for restart. These items remain open and the corrective actions will be reviewed further during subsequent NRC inspections.

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

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

E8.23 <u>(Closed) IFI 50-302/96-201-15; Verification of Motor Starting Data; [identified as IFI 50-302/96-201-05 in Inspection Report 50-302/96-201]</u>

a. Inspection Scope (92903)

This inspector follow-up item expressed a concern that the models used for motors in the emergency diesel generator load sequencing (i.e. motor starting) calculations were not validated by on-site testing. In response to this concern, the licensee performed measurements of voltage, current and power for safety-related motors during the starting period. The inspector compared this data to the calculation results. The acceptance criterion applied by the inspector was that the measured values matched the calculation or showed the calculation to be conservative with regard to the peak momentary real power (kilowatts) reached during sequencing of engineered safeguards equipment onto the emergency diesel generator.

b. Observations and Findings

The licensee made measurements of voltage, current and power during the starting period of the following motors: DCP-1A, DHP-1A, BSP-1A, MUP-1A, RWP-2A, SWP-1A and RWP-1A. This group represented one of each type of large motor except for EFW-1. The test equipment was connected to capture voltage at the feeder bus and current and power flow to the motor. The measurements were made using the Model 1000 Omega Monitor which is manufactured by Reliable Power Meters Co. The Omega Monitor is a microprocessor based instrument. The inspector reviewed the specifications for this test equipment, and observed that it had the capability to produce the desired measurement, which was peak power (kilowatts) during the motor starting period.

The Omega Monitor measured and stored instantaneous values of current and voltage for each of the three phases. The software multiplied those values to obtain the power value and then summed the three phases to obtain total power. Also, the software allowed manipulation of the data to generate plots of voltage , current and power which graphically depicted the starting transient.

The inspector found that the Omega Monitor had been checked against a calibrated instrument before the measurements were taken. The calibration check was made on September 22, 1997, and results were recorded on data sheets, which the inspector reviewed.

As stated above, the focus of evaluation of the data was with peak momentary real power reached during sequencing. At the time the inspector follow-up item was established, the calculated peak momentary power during sequencing was close to the maximum output power rating of the emergency diesel generator. Therefore, it seemed prudent to "validate" the calculation model for certain motor parameters which were based on calculated parameters. Since that time, more sophisticated computer design tools were applied to the load sequencing problem, and these new calculations showed there was considerably more margin than previously determined. Nevertheless, the program for making the measurements went forward.

Load block six, the last load block in the sequence, had the highest calculated peak momentary power. This load block contains DCP-1A (480 V) and BSP-1A (4160 V) motors. The measurements were taken when the system was aligned to offsite power. This meant that the measurement data could not be directly compared with the emergency diesel generator sequencing calculation. Bus voltage at the time of the measurement was significantly higher than for the corresponding sequencing calculation. The inspector adjusted the field data to account for the voltage difference, and compared the adjusted values to the calculation results. The measurement data showed that the calculation was conservative, and that the actual peak should be less than calculated.

The inspector found that all the field data tended to validate the design inputs to the calculation both in magnitude of parameters and shape of transient response. The licensee was in the process of creating a special calculational case to match the test conditions. This will allow a more detailed comparison of actual measurements and computer program results without need for any separate adjustments for voltage differences.

c. Conclusions

The licensee made special field measurements of power flow and other electrical parameters on motors in the starting mode to address an NRC concern related to the sequencing of loads to the emergency diesel generator. Review of the methodology and data by the NRC inspector led to the conclusion that the measurements validated the design input to the calculations for load sequencing, thus resolving the original concern.

The inspector assessed the licensee's performance relative to resolving the issue contained in the Inspector Follow-up 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

E8.24 (Closed) URI 50-302/96-201-03: Operating Curves 16, 17, and 18 in OP-103B are not Validated by Licensee

a. Inspection Scope (92903)

This item involved a URI that was identified during an Integrated Performance Assessment Process (IPAP) team inspection. The URI identified concerns with several curves in operating procedure OP-103B, Plant Operating Curves. Specifically, the IPAP team concluded that curves #16 and #18 had not been validated and curve #17 lacked any basis. The inspector reviewed documentation to determine what corrective actions had been taken to address the concerns of the URI.

b. Observations and Findings

The inspector reviewed documentation and interviewed personnel to assess how the licensee addressed those concerns identified in the URI. During the inspection the inspector found that the licensee was tracking those concerns identified in the URI under restart issue package O-01. Following the review of restart issue package O-01 and other associated documentation and interviewing licensee personnel, the inspector concluded that the licensee had adequately addressed those concerns identified by the IPAP team.

A letter dated September 4, 1997, from Framatome Technologies, provided the results of the engineering evaluation requested by the licensee concerning the curves in question. The engineering evaluation provided in the September 4, 1997, letter indicated that the curves had not been validated using CR3 plant specific design data. Instead the curves were derived by using plant specific data from other power plants. Plant specific data from ANO-1, TMI-1, and Rancho Seco was used to derive the values to plot the curves in question. Based on these findings, analytical validations of the curves, using plant specific data from CR3, were performed. The inspector reviewed the calculations, and verified that the licensee reviewed the calculations performed by Framatome. The inspector verified that the licensee had implemented the changes that resulted from the engineering evaluation. The changes included modifying curves #16 and #18 to reflect plant specific values and the deletion of curve # 17 from the procedure. The inspectors also determined that other curves in OP-103B had been validated for technical accuracy.

During the inspection the inspector also determined that the licensee's failure to validate the curves for technical accuracy was not in accordance with procedural requirements. Specifically, licensee Procedure NEP-213, Design Analyses/Calculations, Revision 44, required in part that "Design Analyses/Calculations must be technically accurate.." Contrary to the requirements of NEP-213, the licensee failed to validate the curves for technical accuracy when they were received from the vendor during initial construction. The failure to validate the curves as required by procedure is a violation of NRC requirements. However, because of the aggressive and appropriate corrective actions taken once the violation was identified, this violation is being treated as a Non-cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. This violation is identified as NCV 50-302/97-19-02, Failure to Validate Operating Curves.

c. Conclusions

The inspector determined that the licensee's corrective actions for the URI were satisfactory. This item is closed.

The inspector assessed the licensee's performance, relative to the corrective actions for this URI, 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

E8.25 <u>(Closed) LER 50-302/97-30-00; Installation Error Resulted in a Containment Isolation Check Valve Disc Sticking in The Open Position</u>

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

This issue involved a check valve that the licensee found stuck in the open position. The inspector reviewed LER 50-302/97-30-00, which was issued once the licensee identified that the check valve was stuck in the open position. The inspector reviewed associated documentation, and interviewed licensee personnel to determine the adequacy of the licensee's response to the issue identified in the LER.

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

The inspector reviewed the documentation associated with the LER and interviewed the licensee to obtain information concerning corrective action and root cause. The stuck open check valve was found by the licensee while performing a scheduled ASME Section XI internal inspection of feedwater valve FWV-46. The sticking of the check valve was attributed to a missing hinge pin and retaining pins on one side of the check valve disc.

The corrective actions identified in the LER included:

- Locating and retrieving the missing hinge and retaining pins;

- Revising procedures to assure that adequate work instructions are provided to assure that welds are properly placed to prevent hinge and retaining pins from becoming dislodged;

- Inspecting other valves of similar design to determine if identical failures could occur; and

- Determining the root cause of the failure

Following the review of associated documentation and interviewing of licensee personnel, the inspector determined that the licensee had implemented the corrective actions identified and that the corrective actions were adequate. The licensee concluded, and the inspector agreed, that personnel error and inadequate work instruction were the contributing factor in the hinge pins and retaining pins becoming dislodged. Specifically, the work instruction or work description did not provide sufficient details as to the location of welds to prevent movement of the valve's hinge and retaining pins. The inspector determined that the failure to provide sufficient work instructions/details, as to the specific location of the welds, were not in accordance with plant procedures. Plant Ccompliance Procedure CP-113B, Work Request Evaluating/Planning, Revision 23, Step 4.1.4.29, states in part, "Write a work description to indicate the activity to be performed and provide sufficient guidance to perform the activity..." The failure to provide sufficient work instruction as required by procedure is a violation of NRC requirements. Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

c. Conclusions

The inspector determined that the licensee's corrective actions for the 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 - Adequate
- Engineering Effectiveness - Adequate
- Knowledge of the Design Basis - N/A
- Compliance with Regulations - Adequate
- Operator Performance - N/A

E8.26 (Closed) VIO 50-302/97-14-04: Failure to Adequately Test HPI Valves MUV-23, 24, 25, and 26 Power Selector Switches

a. Inspection Scope (92903)

This VIO involved the failure of the licensee to adequately test HPI valves MUV-23, 24, 25, and 26 power selector switches: specifically, the failure to perform testing that would determine if the valves would function as designed when powered from their alternate power supply. The inspector reviewed the associated procedure and interviewed selected licensee personnel to determine if the procedure had been revised to include testing of the valves when they are powered from their alternate power supply.

b. Observations and Findings

The inspector reviewed Surveillance Procedure SP-457, Refueling Interval ECCS Response To A Safety Injection Test Signal, Revision 15, and applicable drawings to determine if the licensee had incorporated changes to the procedure to verify that HPI valves MUV-23, 24, 25, and 26 would function as required when powered from their alternate power supply. Following the review of the procedure and associated documentation, and interviews of selected licensee personnel, the inspector concluded that appropriate procedure changes had been incorporated. However, at the conclusion of the inspection period, actual performance of the revised procedure had not been completed. The inspector was informed that actual performance of the procedure would occur prior to going to Mode 4. The NRC will review the results of the test once completed by the licensee.

c. Conclusions

The inspector determined that the licensee's corrective actions for the violation were satisfactory. Specifically, the procedure had been revised to include testing of HPI valves MUV-23, 24, 25, and 26 when they are powered from their alternate power supply. 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

E8.27 Emergency Diesel Generator Issues

(Open) LER 50-302/97-13-00: Emergency Diesel Generator Room Could Exceed Maximum Design Temperature During Operation Due to Inadequate Room Cooling

(Open) LER 50-302/97-19-00, 01: Elevated EDG Supply Air Temperatures Due to EDG Radiator Discharge Air Recirculation Effect

(Open) LER 50-302/97-27-00: Failure to Add Antifreeze to the Diesel Generator Coolant Radiators May Render Emergency Diesel Generator Inoperable During Sub-Freezing Temperatures

a. Inspection Scope (92903)

The inspector reviewed the corrective actions for LER 50-302/97-13-00, Emergency Diesel Generator Room Could Exceed Maximum Design Temperature During Operation Due to Inadequate Room Cooling, LER 50-302/97-19-00 and 01, Elevated EDG Supply Air Temperatures Due to EDG Radiator Discharge Air Recirculation Effect, and LER 50-302/97-27-00, Failure to Add Antifreeze to the Diesel Generator Coolant Radiators May Render Emergency Diesel Generator Inoperable During Sub-Freezing Temperatures. The inspector reviewed the status of modifications affecting the operation of the emergency diesel generators, which have not been covered in previous inspection reports.

b. Observations and Findings

Modification MAR 97-04-02-01 was implemented to replace the control switches for SWP-1A, RWP-2A, SWP-1B, and RWP-2B with pull-to-lock switches. When the new switches are in the pull-to-lock position, the operating contacts will remain open to prevent auto start of the pumps on an Engineered Safeguards signal. Annunciators were installed to notify the operators any time the switches are in pull-to-lock position, in an attempt to prevent an inadvertent operation of the pull-to-lock switches.

The pull-to-lock feature on SWP-1A and RWP-2A will be controlled by Emergency Operating Procedures after the first ES actuation coincident with a loss of offsite power and a failure of EFP-2, provided the B train of SW and RW pumps are running. The licensee concluded that securing the A train pumps is acceptable since the B pumps will start (or remain running) to provide the necessary heat sink. By securing the A train pumps, the licensee concluded that EGDG-1A will have sufficient capacity to allow the simultaneous loading of DHP-1 and EFP-1.

The purpose for installing the pull-to-lock switches on the B train pumps was to prevent the automatic starting of the pumps, in certain circumstances, which could result in pump damage due to low SW surge tank level or other system problems. Currently the DC power switches for the pumps are opened to prevent starting of the pumps in these circumstances.

The inspector reviewed the USQD and installation of the MAR and found no problems with the modification.

Modification MAR 96-03-12-01 was installed to increase the accuracy of the EGDG KW indication on the main control board. This would allow the operators to control EGDG loading closer to the limits. The existing KW transducers, KVAR transducers, and KW indicators were replaced as part of the original modification. The new, combined KW/KVAR transducers are smaller than the components being replaced, which will allow an increase in separation between the wiring between the two trains.

A Field Change Notice was issued to this modification to reduce the EDG potential transformer (PT)/current transformer (CT) burdens, which had the result of further increasing the accuracy of the KW indicators. This was accomplished through two changes: (1) Eliminating a watt transducer which provided EGDG wattage inputs to the non-safety load monitor timers. The wattage input signals were replaced with signals from the safety-related KW indicators installed under this modification. This modification also provided fuse isolation between the KW indicators and the load monitors. (2) Utilizing spare conductors from the EGDG CT to the new watt/var transducer, which reduces the resistance seen by the CT, reducing CT burden.

The inspector reviewed the USQD and installation of the MAR and FCN and found no problems with the modification.

In response to the issue discussed in LER 50-302/97-27-00, concerning not complying with the design requirement to operate the emergency diesel generators during the winter months with anti-freeze in the generator coolant system, the licensee added ethylene glycol anti-freeze to the coolant systems. Quantities were added to protect the engine from freezing down to 15°F. The licensee calculations showed that this amount of ethylene glycol adversely affected the heat transfer capabilities of the coolant. The licensee calculated that with the anti-freeze, the emergency diesel generator could not transfer the required heat loads during accident conditions with ambient temperature above 85°F.

LER 50-302/97-13-00, issued June 26, 1997, addressed a condition discovered during functional testing of the emergency diesel generators where a potential existed for the EGDG rooms to exceed the design basis temperature of 120°F when the outside air temperature exceeds 95°F. The 120°F temperature limit is to provide an acceptable environment for the electrical equipment in the EGDG room. The licensee concluded that inadequate room cooling resulted in the potential to exceed the design basis temperature limits. To increase room cooling, the licensee installed MAR 97-04-03-02, which changed control logic so that two fans would operate on an EGDG start, instead of one. Additional modifications were made by this MAR to the duct work, dampers, and filters in the room ventilation system.

LER 50-302/97-19-00 was issued on August 1, 1997 and revised (supplement 01) on December 18, 1997. This LER identified that a portion of the elevated EGDG room temperature and supply temperature was caused by recirculation of the EGDG exhaust air into the room. The licensee determined that the recirculated air can cause the supply air to the rooms to be increased approximately 15°F above the design basis outside air temperature of 95°F. The licensee developed a modification, MAR 97-08-04-01, to reduce the amount of recirculation air. This MAR has not been installed, but Revision 1 to the LER states that the work will be completed by February 28, 1998.

A JCO was reviewed by the Plant Review Committee (PRC) on December 29, 1997 to address continued operation with the remaining issues from the three LERs. The licensee determined that ambient air temperature of less than 86°F will not impact the operability of the EGDGs, but above that temperature, the EGDGs will be inoperable. This determination was independent of mode. The JCO assumes that during the winter months of December through February, the ambient air temperature will not exceed 86°F. The PRC conditionally approved the JCO with requirements that operations procedures be revised to provide guidance to the licensee personnel for actions to be taken if the temperature approaches or exceeds the limits. This guidance was to include increased monitoring requirements if air temperature exceeds 81°F and requirements for adding or removing anti-freeze from the coolant system, based on air temperature. As of the end of the inspection period, these procedure revisions had not been made.

c. Conclusions

These items remain open, pending the development of procedure revisions necessary to meet the conditions of the PRC approval of the JCO. The remainder of the issues with the EGDGs have been completed, with the exception of the modification to reduce the air recirculation, which the JCO states will be implemented by February 28, 1998.

The inspector assessed the licensee's performance, with respect to these issues, in the five areas of continuing NRC concern.

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

E8.28 (Closed) EA 96-365, VIO A (01062): Inadequate 50.59 Evaluation for Post-LOCA Boron Precipitation Control

a. Inspection Scope (92903)

The inspector reviewed the corrective actions taken in response to EA 96-365, VIO A, issued March 12, 1997 and responded to by the licensee in a letter dated June 16, 1997.

b. Observations and Findings

The licensee has made programmatic improvements to the 10 CFR 50.59 evaluation process, which were assessed as being adequate in IR 50-302/97-13. In a letter dated June 26, 1997, the licensee submitted a proposed license revision addressing changes to the boron precipitation methodology and license conditions. A safety evaluation and unreviewed safety question analysis were completed as part of this submittal.

c. Conclusions

Based on the license submittal and the successful completion of the 10 CFR 50.59 evaluation process upgrade, this violation is closed. Final dispositioning of the license amendment by NRR will close outstanding concerns for boron precipitation.

The inspector assessed the licensee's performance, with respect to these issues, in the five areas of continuing NRC concern.

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

E8.29 Boron Precipitation License Submittal

a. Inspection Scope (92903)

In a letter dated June 26, 1997, the licensee submitted a proposed license revision addressing changes to the boron precipitation methodology and license condition 2.C(5) requirements. The inspector reviewed the modifications installed to satisfy the proposed license amendment.

b. Observations and Findings

The existing license condition requires that the licensee be able to measure at least 40 gpm flow through the decay heat drop line. The installed flow instrumentation is calibrated for a zero to 5000 gpm flow range and is not sensitive enough to read 40 gpm. The licensee has performed a modification to replace the installed flow instruments with thermocouples on the pressurizer auxiliary spray line and the decay heat drop line. Low flow through the lines should be indicated as an elevated temperature reading in the lines. A license amendment request has been submitted to change the license condition and to propose a revised boron precipitation strategy.

The NRC-issued Operating License for the Crystal River plant contains a condition, 2.C.(5), which requires that within six months of the date of issuance of the license the licensee would install "flow indicators in the emergency core cooling system to provide indication of 40 gallons per minute flow for boron dilution." Licensee records show that these flow indicators were installed in May 1977 and they consist of two ultrasonic flow detectors installed on the 12" decay heat (DH) removal drop line from the RCS to the DH system and on the 2" auxiliary pressurizer spray line. The licensee identified that these flow indicators have not been reliable over the years, requiring recurring maintenance, and that their range is so wide that they could not accurately indicate water flows as low as 40 gpm as specified in the operating license. The licensee decided to replace the ultrasonic flow indicators with two thermocouples. In certain postulated accident scenarios, days after a Loss of Coolant Accident (LOCA), one might want to initiate flow through either of those flow paths to promote mixing of the water remaining in the reactor vessel to prevent postulated dilution of boron concentration in the vessel due to boil-off. The purpose of the flow indicators is to confirm that water flow begins in the pipes when downstream valves are opened. The thermocouples could perform that function by sensing a temperature increase of the pipe surface as hot RCS water begins to flow through the pipe after the valves are opened.

The inspectors went to the location of the flow sensors and indicators and discussed the proposed modification with a licensee engineer preparing the modification. The thermocouples were to be placed on pipes in a room outside containment, very close to the location of the original ultrasonic flow sensors. The thermocouple readout would be connected to an existing multiple point temperature recorder in the control room. The inspectors concluded that the thermocouple installation as proposed would allow operators in the control room, in the postulated accident scenario, to observe pipe temperature rise after opening valves, and thus verify that mixing flow had been initiated.

It was the conclusion of the inspectors that thermocouples, installed in accordance with the licensee's plan, will enable the licensee to meet the license condition for detecting flow in the DH drop line and the auxiliary spray line. NRC will confirm during a future inspection that the thermocouple installation has been completed satisfactorily.

The inspector reviewed the modification, MAR 97-12-01-01, and the installed equipment. Originally, the thermocouple on the DH drop line was installed at the top of a horizontal run of piping. After concerns were raised regarding thermal stratification effects, the licensee relocated the thermocouple to the bottom of the pipe. Indication was added to an existing recorder in the main control room. The inspector identified that even though the points were indicating on the recorder, the placard did not identify that the points had been added, procedures had not been revised to identify the points on the recorder, and the control room operators were not aware of the modification. The MAR was still open at the end of this inspection period. The inspectors will continue to monitor completion of this modification.

c. Conclusions

The inspectors concluded that thermocouples, installed in accordance with the licensee's plan, would enable the licensee to meet the license condition for detecting flow in the decay heat system drop line and the auxiliary spray line.

E8.30 Emergency Diesel Generator Power Uprate Testing

a. Inspection Scope (92902, 92903)

The inspectors completed a review of the emergency diesel generator upgrade modification testing. An issue that arose during the testing was followed-up by the inspectors.

b. Observations and Findings

Precursor Card 97-8184 was written on December 3, 1997, regarding a grease seal that was identified as being damaged in the bearing housing for the radiator fan for EGDG-1A. This condition was discovered while the licensee was preparing to upgrade the fan shaft assembly as part of the diesel generator upgrade modification. Licensee investigation discovered that the grease seal was damaged and that the bearings on the shaft to be replaced had not been fully packed with grease. Signs of corrosion were also noted on the bearing races. The licensee replaced the entire assembly, as had been planned, and performed a root cause evaluation on the removed components to determine past operability.

A task force of licensee personnel, contract failure analysis personnel, and the diesel generator vendor, in consultation with the manufacturer of the bearing, conducted a root cause analysis and evaluation. The root cause determined that during shaft installation, the rubber seal lip was deformed with a cold set, suggesting improper installation. No signs of overheating were noted. The cold set allowed more moisture intrusion, which collected at the bottom bearing interface, causing corrosion in the area. It was concluded that even though the bearings had not been fully packed with grease, as recommended, the amount of grease present on each bearing was sufficient for the design basis, seven day operation, of the emergency diesel generator. The licensee verified that the bearings on EGDG-1B had been hand packed prior to assembly. New bearings were installed on EGDG-1A, which were verified to have been hand packed.

Following completion of this modification, the licensee conducted additional testing on EGDG-1A for the modification functional test. The testing performed is described in the attachment at the end of the inspection report.

c. Conclusions

The post modification testing for the power uprate on EGDG-1A was successfully completed. Weaknesses in the control of the bearings for the radiator drive shaft were identified, but were corrected prior to returning the diesel to operation.

E8.31 (Closed) IFI 50-302/95-15-04; Code Requirement for Therma. Relief Valves on Decay Heat Removal Heat Exchangers (92903 and 62707)

This issue was first discussed in the Service Water System Self-Assessment IR 50-302/95-15 and again in IRs 50-302/95-21 and 50-302/97-08. This issue was also the subject of NRR Task Interface Agreement 96-014. The final resolution by the licensee was in agreement with the NRC in that both DH heat exchangers require the installation of thermal relief valves to protect the components from overpressurization. The inspectors reviewed the design change package and observed some of the field work that installed the valves. No concerns were noted. This issue is considered closed.

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

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

E8.32 (Closed) LER 50-302/97-15-00 and 01: Non-Conservative Differential Pressure Rating for Letdown Line Inboard Containment Isolation Valves Prohibiting Safety Function Performance (92903)

During the licensee's review of a calculation for the maximum differential pressure (dP) for Makeup Valves MUV-40/41/505, which are the letdown line inboard containment isolation valves, it was discovered that the evaluation of the maximum dP across these valves was non-conservative. Per the Motor Operated Valve (MOV) Diagnostic Testing Program, these valves could only be rated to close against a maximum dP of 1800 pounds per square inch differential (psid) Calculation M97-0047 indicates the valves could be subject to a dP of 2380.4 psid under design basis conditions.

Part of the licensee's corrective actions was to install a new valve located downstream of the letdown coolers and adjacent to reactor building penetration 333 that could close against a dP of 2380.4 psid. This new valve (MUV-567) replaced the engineered safeguards actuation system (ESAS) containment isolation requirement of MUV-40/41/505. The ES signal was removed from all three valves with the ES signal for MUV-505 being added to MUV-567. Makeup Valves MUV-40 and 41 will maintain power operation capability from the control room and remote shutdown panel while power for MUV-505 will be removed completely and the actuator abandoned in place.

The inspectors reviewed MAR 97-06-20-01 and the associated restart package (FPC Restart Issue D-67). The inspectors concluded that the licensee's corrective actions were appropriate and acceptable. Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

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

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

E8.33 (Closed) LER 50-302/97-28-00: Flow Element Accuracy Uncertainty May Be Greater Than Assumed Due to a Design Error (92903)

This issue was identified during the Raw Water System Readiness Review. The licensee determined that the Nuclear Services Closed Cycle Cooling (SW)/Nuclear Services and Decay Heat Seawater (RW) flow element tap was not bounded by the lengths of straight pipe recommended by the instrument manufacturer. This determination was based on the review and comparison of piping drawings and the flow element instruction manual. The lack of sufficient straight pipe can reduce flow measurement accuracy, which in turn could impact assumptions made in plant calculations, as well as compliance with test requirements in Section XI of the ASME Boiler and Pressure Vessel Code. PC 97-3295 documented this condition.

The original RW system design did not include flow measurement capability but was modified in 1987 to allow the use of annubar flow elements. Under MAR 87-07-21-01, one spool piece in each RW loop was modified to allow flow element installation. When determining the locations for the flow element taps, vendor catalog information was used rather than an installation or instruction manual. Data contained in the vendor catalog disagreed with information contained in the instruction manuals.

As part of the extent of condition evaluation, the licensee determined that the configuration of the SW and DC (Decay Heat Closed Cycle Cooling) system flow elements also did not satisfy the manufacturer's recommendations for installation. PCs 97-5977 and 97-5978 documented the inadequate installation of the DC and SW flow instruments.

The corrective actions involved the fabrication and installation of a new spool piece in the SW/RW train. The new spool piece was made of an alloy that does not require the inner lining used on the other spool pieces. This will permit the use of ultrasonic flow indication without subjecting the piping to corrosion from salt water. The annubar elements in the DC trains have been replaced with ultrasonic flow instruments and an ultrasonic flow instrument has been installed in the SW system.

The inspectors concluded that the licensee's corrective actions were appropriate and acceptable. Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

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

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

E8.34 <u>(Closed) LER 50-302/97-39-00: Unqualified Material Left in Reactor Building During Construction Could Affect Post-LOCA Cooldown Capability (92903)</u>

On June 2, 1997, the licensee discovered a Styrofoam-like substance between the Reactor Building (RB) steel liner plate and the concrete incore pit/letdown cooler structure while preparing the RB liner for coating application. The initial review of the condition concluded that no operability or reportability concerns were evident. A subsequent engineering evaluation concluded that the polystyrene was not a problem with respect to fire protection or 10 CFR 50, Appendix R considerations. However, on November 8, 1997, further engineering evaluation determined that the substance was not qualified for a post-accident environment in the RB. This issue was reported to the NRC and documented in PC 97-3828.

The licensee attempted to locate a document that described the use, or need, for the polystyrene. It was used to form the three-inch separation between the incore pit/letdown cooler structure and the wall of the RB during construction. No drawings, specifications, or any design basis document was found. The use of polystyrene inside the RB was not described in the Final Safety Analysis Report, the Improved Technical Specifications, the Enhanced Design Basis Report, or any other licensing basis document. The existence of polystyrene inside the RB had not been analyzed, and the physical and chemical behavior in a post-Loss of Coolant Accident (LOCA) environment was not known, therefore the licensee decided to remove it.

Several concerns with the existence of polystyrene inside the RB were identified. The most notable was the dislodging and transportation of the polystyrene to the RB sump, thus potentially affecting the ability to mitigate the consequences of a LOCA. To resolve this issue, the licensee sent several samples of the Styrofoam-like substance offsite to be analyzed. The results indicated that the specific gravity of the Polystyrene after 24 hours of exposure to a 285°F post-LOCA environment was less than that of the water inside the RB (0.76 grams/cubic centimeter vs. 0.94). The 285°F value was derived from the temperature of the saturated steam for the post-LOCA environment inside containment.

The inspectors interviewed engineering personnel, reviewed the corrective actions and Operability Concern Resolution (OCR) Report 97-009, and concluded that the licensee's resolution of this issue was acceptable. Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

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

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

E8.35 (Closed) URI 50-302/96-201-02: Net Positive Suction Head for Building Spray Pump (92903)

This issue was raised during the IPAP inspection in July of 1996. The concern was with the margin between the net positive suction head available (NPSHa) under accident conditions and the NPSH required (NPSHr) for the 1B Building Spray Pump (BSP-1B). The margin for BSP-1B taking a suction from the RB sump by calculation was only 0.1 feet greater than the NPSHr, as taken from vendor curves for the specified flow rate. This issue subsequently became a design issue addressed in the NRC's Confirmatory Action Letter.

As part of the corrective actions, the licensee sent four impellers to Ingersall-Dresser where a modification was performed which decreased the NPSHr, without impacting head/capacity performance. The modifications to reduce NPSHr included the following: 1) the impeller eye diameter was increased to match the casing approach more closely; 2) the leading edges of the inlet vanes were cut back to open flow passages; 3) the inlet vanes were smoothed and sharpened; and, 4) the impeller eye area was polished. New NPSH curves were provided by the vendor and all four impellers were tested in the vendor's flow loop for NPSH characteristics. Two of the four BSP impellers were installed in the plant and tested in accordance with post modification testing. Both vendor and in-plant testing demonstrated that the pumps' head/capacities were unaffected by the modification. The post modification results for NPSH margins for the BSPs from the RB sump were found to have increased to 2.01 feet for BSP-1A and 1.28 feet for BSP-1B.

After interviewing engineering personnel and reviewing the design change package, along with the vendor's impeller test results, the inspectors concluded that the new NPSH margins were adequate to assure cavitation free operation of the BSPs during accident conditions. This issue is considered closed.

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

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

E8.36 (Closed) LER 50-302/97-24-00; Feedwater Valv Do Not Meet the Requirements for Main Steam Line or Feedwater Line Break Event Mitigation Due to Inadequate Design and Calculation Errors (92903)

There were several issues addressed in this LER. The first issue involved differential pressure (dP) across feedwater suction valves FWV-14/15 that could prevent the valves from closing in response to a signal from the EFIC system. The design calculation for the maximum dP for FWV-14/15 included incorrect and non-conservative plant response assumptions following a Main Steam Line Break (MSLB) or a Feedwater Line Break (FWLB). Based on those assumptions, the maximum dP for the suction valves was calculated to be 324 pounds psid at a 25% power level. Preliminary calculations using expected plant conditions indicated that the dP for a FWLB could be as high as 405 psid.

The second issue involved several deficiencies between various design documents, the FSAR, and calculations for closure time of the low load block valves FWV-31/32. The low load block valves close in less than 66 seconds compared to the FSAR Chapter 14 accident analysis of 34 seconds.

The third issue involves the FSAR Chapter 14 analysis not reflecting the current analysis of record. The FSAR analysis used non-conservative Moderator Temperature Coefficients (MTC), the analysis of record assumes two High Pressure Injection (HPI) pumps where the original FSAR only assumed one, and the FSAR references an obsolete computer code. Also, the conclusion that the reactor would remain subcritical for Case III could not be substantiated. Case III is the analysis in which all four steam lines in the non-seismic portion of the steam system are assumed to break during a seismic event and credit is not taken for the Main Steam Isolation Valves (MSIVs).

The fourth issue involved the maximum dP calculations for the main feedwater system isolation valves. These calculations assumed that the Main Feedwater Pumps (MFPs) are tripped and the only pumps providing head pressure are the feedwater booster pumps. The analysis of record credits MFPs tripping during a MSLB while no credit was taken in the current FSAR analysis for the MFPs to trip. The event mitigation for MSLB and FWLB in the FSAR Chapter 14 analysis was dependent on FWV-14/15 closing, as well as the feedwater low load block valves FWV-31/32, closing within 34 seconds. The low load block valves were designed to close within 66 seconds. Therefore, feedwater isolation cannot be performed within the assumed time of the original FSAR analysis of 34 seconds, concurrent with the failure of suction valve FWV-14 or 15 to close. The slower closure time of FWV-31/32 has been previously evaluated and met the acceptance criteria for Chapter 14 MSLB but was not incorporated into the FSAR. Conservative MTC and HPI assumptions were analyzed during recent cycle specific analyses but were also not incorporated into the FSAR.

The licensee's corrective actions included: 1) re-evaluating the FSAR Chapter 14 MSLB with appropriate bounding values for input parameters; 2) documenting in the FSAR the slow closing low load block valves; 3) modifying and testing the MFP suction valves' actuators; and 4) revising the FSAR, the Enhanced Design Basis Documents, the Improved Technical Specifications, and the Single Failure Topical Design Basis Document with the new analysis, which credits all as-built safety-related systems and response times. The inspectors considered the licensee's corrective actions to be appropriate and acceptable. Although this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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, 50.59 Evaluations, Procedure Adequacy, Reportability, and Corrective Actions That Are Subject to Enforcement Discretion.

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

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

E8.37 <u>(Closed) Restart Design Issue D.I.1: High Pressure Injection Pump Recirculation to Make-up Tank</u>

<u>(Closed) LER 50-302/97-08-00: Potential of HPI Pump Recirculation Capability Resulting in Possible MUT Overflow or Possible Pump Failure</u>

<u>(Closed) IFI 50-302/96-17-02: Potential for HPI Recirculation Resulting in MUT Overflow</u>

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

The inspector reviewed the licensee's closure documentation for their Restart Issue D-1 which encompasses all of the above items. The inspector reviewed the licensee's modification package for the design change to the high pressure injection (HPI) pump recirculation line (MARs 96-11-02-01 {mechanical} and 96-11-02-02 {electrical}) and verified the installation of the components in the plant. The inspector reviewed the MAR Functional Test and witnessed portions of the testing.

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

The original design issue was identified by the licensee early in 1996 and involved operation of the HPI pumps with suction being supplied from the Low Pressure Injection (LPI) pumps drawing from the reactor building sump during a small break Loss of Coolant Accident. With this lineup established, the make-up tank (MUT), which is the normal HPI suction source, is isolated from the HPI pump suction by closure of outlet check valve MUV-65, due to the approximate 200 psig suction pressure to the HPI pumps from the discharge of the LPI pumps. The MUT could also be isolated by closing manual Valve MUV-64 in the same line per instructions contained in EOPs. Normal HPI pump minimum recirculation flow is directed to the MUT. This could pressurize the isolated MUT in this scenario, creating the potential of releasing reactor coolant into the Auxiliary Building (AB) from the RB sump, via a MUT relief valve. In an October 28, 1996 letter to the NRC, the licensee identified this item as one of the eight significant design issues they committed to resolve prior to restarting the plant. They then documented the issue as an unanalyzed condition in LER 50-302/97-08-00. Although this LER was significantly beyond the required 30 day reporting requirement, the licensee identified the cause of the delay and has corrected their reportability process as discussed in IR 50-302/97-17. The LER also discussed another unanalyzed condition with the existing recirculation line in that the two line isolation valves were installed in series, did not have safety-related power supplies and were therefore not single failure proof. A single failure of either valve to open could have resulted in the loss of an adequate recirculation path for all three HPI pumps which could damage the pumps in a SBLOCA that required minimal HPI flow.

The licensee had considered procedurally throttling the LPI pump discharge to the HPI pumps to reduce the LPI discharge pressure to below the MUT pressure, allowing some of the MUT water to go to the HPI pump suction. The licensee elected instead to install a new recirculation line that would allow the HPI pump recirculation to go directly to the RB sump during the susceptible SBLOCA scenario. The inspector considered this a good decision which simplified the procedural burden on the operators, and addressed both issues in the LER.

The inspector observed that the licensee's modification added new piping, which tapped off the existing combined HPI pump recirculation line upstream of the existing series isolation valves, and routed recirculation flow to the RB sump. The piping contained manual isolation, vent, drain, and containment isolation valves to facilitate venting and local containment leak rate testing. It also contained an array of four solenoid operated valves arranged in two series sets. Each set had two valves in parallel, powered from different electrical trains to meet single failure criteria. Isolation of the containment penetration created by the line open to the RB atmosphere was provided by a check valve in containment and the downstream pair of solenoid valves outside containment. The new line was designed to only be used in an accident scenario so the solenoid valves were to be normally deenergized and closed. Controls for the solenoid valves were provided on the main control board. The previously existing recirculation line was not altered and will still function routinely as before.

The inspector reviewed the MARs that implemented this modification and determined they adequately resolved the technical recirculation path concern. The inspector did not identify any discrepancies during the field verification of the system installation.

The inspector reviewed the MAR functional test procedure (TP) written to validate the performance of the modification. The inspector noted that although the work on this MAR was largely completed by November, the TP was not issued until December 31, immediately prior to the scheduled test performance. That evening, an on-shift operator questioned various portions of the test. This resulted in a more detailed review on the morning of January 1, 1998. One operations concern for visually monitoring flow into the sump was agreed to be done by the review team but was not incorporated into the TP. The inspector discussed the need to incorporate actions that were perceived as necessary into the TP to ensure they were actually performed. The team agreed to incorporate the step and the inspector verified it was in the final TP. The inspector considered the team's original plan a poor resolution of a valid concern.

The review also revealed that a poor safety assessment (SA) had been done by the licensee for the test. The SA was dated December 19, 1997 and was not an adequate reflection of the TP as issued on December 31, 1997. Specifically, the test was modified in the intervening time to delete the use of a second HPI pump for LTOP concerns, but the SA was written assuming two pumps were running. The SA also assumed a specific plant configuration that was not required per the TP. Specifically, based on the TP statement that LTOP requirements needed to be met throughout the test, the SA author assumed and delineated plant conditions that had the pressurizer PORV available and operable. However, the PORV had been declared inoperable and isolated via its block valve on November 13, 1997 for unacceptable leakage. LTOP contingency requirements for this, such as defeating the automatic swap to the borated water storage tank (BWST), were complied with, but the PORV was not and had not been in the status assumed in the SA when it was written. The TP also contained actions allowing opening of the BWST valves which conceptually defeated the LTOP contingency action intent. The SA did not analyze this because it was written assuming the PORV was operable. The SA also contained errors in the numbers used for the LTOP requirements. The inspector and licensee management also questioned how an SA could be completed on a draft TP 12 days prior to it being issued. Although the specifics of these errors were not safety significant, the inspector considered this as another example of a SA not matching a test procedure, similar to the example cited in VIO 50-302/97-17-01: Failure to Conduct an Adequate Unreviewed Safety Question Evaluation for a Modification Functional Test. The licensee initiated PC 98-0024 on the problems and agreed to address this current problem in the scope of their response and corrective action to that VIO. The inspector verified that the licensee revised the TP and reperformed the SA prior to completing the test. The inspector considered the discovery of the error to be a very good example of questioning attitude on the part of the operator performing a non-required review of the test. The operator had identified several concerns other than those detailed above that initiated a thorough management review of the test. The inspector observed several other examples of poor attention to detail in the MAR TP development paperwork in that forms were not fully completed and boilerplate language was used excessively.

The licensee identified the cause of the original unanalyzed design condition as a design error because their original analysis did not consider the full range of break sizes and did not address the SBLOCA scenario discussed earlier. This cause was consistent with the causes for many items the licensee had corrected during this outage and the inspector considered it an appropriate determination. In LER 50-302/97-08-00 the licensee assigned several programmatic corrective actions to their MCAP II program, which was already in progress. The inspector considered the assignment of these commitments to MCAP II to be appropriate but did not verify them for completion because MCAP II has already been the subject of inspections.

Although the original design deficiency of this item is a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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.

The licensee closed their restart issue D-1 on this effort with open items for emergency and normal operating procedure revisions, local containment leak rate testing, and flow print updates. The inspector verified these were tracked for completion by licensee processes and considered this acceptable for closure of this issue. Consequently, these items are closed to the above NCV.

c. Conclusions

The inspector concluded that the licensee had adequately modified and tested the plant to resolve a significant design issue with high pressure injection pump recirculation. However, the licensee's test procedure was not completed and issued in a timely manner to allow for thorough review. The test safety assessment did not adequately reflect the test in that it erroneously assumed plant conditions not required by the test, was issued 12 days prior to the test issuance, and contained errors in LTOP parameters.

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

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

E8.38 (Closed) Restart Design Issue D.I.8: Generic Letter (GL) 96-06 (Thermal Overpressure Protection for Containment Piping, Penetrations and Coolers)

(Closed) LER 50-302/97-04-01: Thermal Relief Valves Inside Containment Do Not Meet Requirements For A Design Basis Accident

(Closed) LER 50-302/97-12-00: Industrial Cooling (CI) System Penetrations Not Designed for Containment Isolation

(Closed) IFI 50-302/96-08-02: Reactor Building Cavity Cooling Piping Thermal Relief Protection

a. Inspection Scope (92903)

The inspector reviewed the licensee's closure documentation for their Restart Issue (D-8 Series) which encompasses the corrective actions for all of the above items. The licensee committed to complete all modifications required to address GL 96-06 containment integrity concerns prior to startup from the current outage. The inspector verified installation of new components by plant walkdowns and MAR review.

GL 96-06 was issued by the NRC on September 30, 1996, and supplemented on November 13, 1997, to address the three main issues of concern: 1) Potential for water hammer on cooling water systems serving the containment air coolers following a high energy line break; 2) Potential for two-phase flow in containment air cooler cooling water systems during similar scenarios; and 3) Thermally induced over pressurization of isolated water-filled piping sections in containment. The licensee responded to these issues in letters dated January 27, 1997, April 30, 1997, and October 17, 1997.

The licensee's preliminary plans to address GL 96-06 were previously inspected and documented in Inspection Report 50-302/96-17.

b. Observations and Findings

The licensee's reviews concluded that no action was required to address the first two GL 96-06 concerns because water hammer and two phase flow could not occur in the containment air coolers. This was because a nitrogen overpressure was maintained on the on nuclear services closed cycle cooling system (SW) that serves these components. Consequently, the licensee's efforts and the inspector's review focused on the over pressurization of isolated piping concern. The licensee reviewed all potentially isolated and fluid filled containment penetrations for potential over pressurization from post-accident containment heat up. They appropriately screened out lines that had compressible fluids or were open to a surge or vent volume. The licensee also reviewed penetrations with existing relief valves to ensure the relief valves were adequately sized and determined that relief valves in penetrations for the containment air coolers were inadequate and replaced them.

The remaining 13 susceptible penetrations were modified by addition of expansion chambers outside containment and rupture discs in containment.

The inspector observed that the licensee had subdivided their GL 96-06 effort under Restart Issue D-8 into 4 sub-issues - A through D. The inspector reviewed the licensee's correspondence and closure documentation and verified the field installation for each of these sub-issues as follows.

D-8A: Provide Thermal Relief Capacity for SW System Inside Containment. This sub-issue culminated in one MAR, 96-10-04-02, which addressed all components requiring new or upgraded relief valves. This MAR added or replaced 28 SW, 4 CI, 5 DC and 4 SC system relief valves. The majority of the MAR was previously inspected and closed in IR 50-302/97-17 as closure for a separate issue with inappropriately removed relief valves tracked under IFI 50-302/95-15-05. The remaining MAR items were addition of new relief valves to the CI and SW systems for GL 96-06 concerns. Temporary MAR (TMAR) 96-07-16-01T had previously installed relief valves on the CI system as an immediate action to address the issues in IFI 50-302/96-08-02, Reactor Building Cavity Cooling Piping Thermal Relief Protection. The CI valves from the TMAR were incorporated in permanent MAR 96-10-04-0-2. This fully resolved the concern identified on IFI 50-302/96-08-02. The MAR also fully resolved the remaining relief valve concerns identified by a third party engineering review documented under PC 97-0055 and in LER 50-302/97-04-01. Most of the affected valves in this population were previously inspected for IFI 50-302/95-15-05 as mentioned earlier, but the relief valves in the three nuclear services closed cycle cooling system (SW) lines to the containment air coolers were specifically done to address the concerns from this restart issue. The three reliefs on the air cooler cooling coils were replaced for inadequate sizing. The three on the motor coolers were also all replaced, although two had been removed by a generic MAR several years ago and the third was inadequately sized. The inspector verified these were included in the MAR and verified their installation in the SW system. The MAR also contained the design basis for the relief valve set points. The set points were based on a maximum 55 psig back pressure from the potential containment pressurization following an accident. The inspector briefly reviewed the licensee's calculations and did not identify any discrepancies. Based on the inspector's review and verifications, the licensee adequately addressed the concerns identified in Restart Issue D-8A.

D-8B: Water-hammer and Two-phase Flow Evaluation and Component Material Capabilities. During their reviews, the licensee discovered another problem they documented in PC 97-0055 in that the SW quick disconnect fittings for the control rod drive coolers were not originally rated for post-LOCA containment atmosphere. The licensee had their reactor vendor analyze the materials to confirm that components would not fail from exposure to the post-accident environment. This sub-issue, also contained the licensee's justification, was why water hammer and two-phase flow were not concerns for the SW system. The inspector briefly reviewed the licensee's justifications and did not identify any discrepancies.

D-8C: Provide Thermal Relief Protection for Fluid-filled Containment Penetration Piping. The licensee identified thirteen containment penetrations susceptible to the isolated piping over pressurization concerns described in GL 96-06. The inspector reviewed their penetration screening effort against complete penetration lists and did not identify any problems with their conclusions. The 13 susceptible penetrations were modified by addition of expansion chambers with the exception of penetration 314 and 318 for steam generator secondary side drains. These two lines are normally drained of fluid so the licensee elected to install a rupture disc inside containment in the event an isolation valve leaked by its seat. The 12 expansion chambers used on the other 11 lines were an integral and non-isolable part of the piping boundary and provided a air-filled surge volume that could be utilized in an over pressurization event via a rupture disc. In the event of a containment valve isolation and heat up of the piping and fluid, the pressure would be relieved into the expansion chamber via an integral rupture disk, but still be contained from escaping to atmosphere, thus preserving containment integrity. The expansion volume of the chamber was normally filled with air. All 12 chambers were installed outside containment, close to the containment wall and were connected by stainless steel tubing to the affected penetration piping. The chambers were installed via MAR 96-10-04-01, Reactor Building Penetration Expansion Chambers. The inspector verified the installation and valve alignments for each chamber and the two rupture disks in containment. The inspector observed that the chambers contained low point drain valves as well as a drain of the chamber volume that could be used to detect leakage past the rupture disk. These were verified to be incorporated into a monthly surveillance check. The inspector identified one concern on the walkdown that had been previously reported in IR 50-302/97-14 in that supports for the small bore stainless steel piping connecting the chambers to the protected piping would be beneficial to prevent damage. The inspector observed two chambers where this piping was susceptible to being damaged from impact by a tool cart or stepped on by workers. The inspector did not identify any other concerns and considered this issue adequately resolved.

D-8D: Industrial Cooling (CI) Piping. The licensee discovered during their reviews that the closed loop CI piping in containment was not seismically qualified. Although not clearly an issue under GL 96-06, the licensee considered that a failure of the closed loop CI piping during a seismic event could jeopardize containment integrity. They upgraded the piping inside containment to Seismic Class I by analysis and the addition of supports via MAR 97-06-09-01. The piping penetrating the reactor building containment and the piping n the Auxiliary Building were already designed to Seismic Class I requirements. The inspector reviewed the licensee's documentation and verified the supports were installed in containment on affected air coolers. The inspector considered this adequate to resolve this sub-issue.

Most of the licensee's screening and design effort for this issue were validated or performed by third party engineering firms to provide independent review and verification of their actions. The results of this review were recorded in licensee Calculation Document M97-0039. The inspector reviewed this effort and concluded it was another example of the licensee's improved standards applied to engineering work.

The inspector's assessment of the adequacy of the licensee's corrective action efforts and restart package closure documentation identified some discrepancies. The licensee's documentation was not logically arranged in that it did not clearly define the problem, develop corrective actions based on a root cause effort or engineering investigation, and provide objective proof these actions were completed. The licensee's actions were scattered amongst various engineering reviews and MARs which made it difficult to ensure the problem was adequately resolved. This indicated to the inspector that the licensee's corrective actions were not driven by a PC in the corrective action process. The inspector also identified numerous discrepancies with tracking of open items for MARs and deficiencies with the MAR return to service process that justifies to Operations that the work is done and the system is ready to be restored. These observations occurred late in the report period and several similar examples were identified by the licensee in parallel. Consequently, review of that concern is ongoing and will be discussed in a subsequent report.

Although the design deficiencies of this item discussed in the LERs are a noncompliance with regulatory requirements, for the reasons discussed in Inspection Report 50-302/97-21, 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 concluded Design Issue 8 on Generic Letter (GL) 96-06 concerns was adequately resolved. Consequently, these items are closed. The licensee implemented a unique solution to one concern by the expansion chamber design and had fully incorporated the intent of GL 96-06. However, the licensee's Restart closure packages were poorly organized compilations of raw data that did not effectively justify the resolution of a safety concern. The inspectors considered this another example of a previously observed problem with corrective action not being driven by a precursor card in the corrective action system.

The inspector assessed the licensee's 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 - Good
- Compliance with Regulations - Good
- Operator Performance - N/A

IV. Plant Support

S1 Conduct of Security and Safeguards Activities

S1.1 (Closed) VIO 50-302/96-07-02: Failure to Complete Screening Elements for Fitness for Duty Personnel

a. Inspection Scope (81502)

The inspector reviewed corrective actions to Violation 50-302/96-07-02 to determine the adequacy of the licensee's corrective actions with respect to their failure to complete screening elements for the Fitness for Duty (FFD) staff.

b. Observations and Findings

The inspector reviewed Access Control Procedure (ACP) 104, Revision 1, dated October 1, 1996, to evaluate the licensee's corrective action documented in a letter to the NRC dated October 4, 1996. ACP 104 had been revised to include the positions of individuals on the FFD staff that would be required to complete screening elements every three years. The Medical Review Officer and the FFD Administrative Assistant were appropriately rescreened to satisfy the requirements stated in 10 CFR 26, Subpart B, Section 2.3. A review of the remaining FFD personnel was conducted by the licensee. Five additional individuals were discovered that did not meet the requirements. The licensee subsequently cleared those individuals. Also, the licensee determined that two couriers required the additional rescreening.

A training session was conducted on November 15, 1996, with those personnel who were responsible for administering the FFD program to ensure the requirements were understood. The licensee's corrective actions were considered adequate to close Violation 50-302/96-07-02.

Upon an annual FFD audit (QPA 97-0032, dated June 2-30, 1997), the licensee's Quality Assessments (QA) determined the licensee misinterpreted the provisions stated in 10 CFR 26, Subpart B, Section 2.3. This section states in part, "Appropriate background checks and psychological evaluations shall be completed prior to assignment of any tasks associated with the administration of the program, and shall be conducted at least once every three years." QA determined that the licensee's misunderstanding of 10 CFR 26, Subpart B, Section 2.3 was reflected in ACP 104, Revision 1. ACP 104, Revision 1, Section 6.1.3.2 stated in part, "Fitness for Duty Program personnel who have been granted unescorted access authorization to Crystal River 3 will meet these requirements while they maintain unescorted access authorization."

Upon discussion with licensee representatives, the inspector determined that ACP 104 was inadequate, which contributed to the licensee's failure to rescreen two FFD personnel every three years, since they were currently badged and held unescorted access to Crystal River 3.

The licensee revised ACP 104 on August 14, 1997, to delete the provisions that badged individuals would not be required to be rescreened. The two FFD personnel who were not rescreened earlier, were subsequently screened on July 1 and 2, 1997.

This non-repetitive, licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the Enforcement Policy. This violation is identified as NCV 50-302/97-19-03, Failure to Rescreen FFD Personnel under the Provisions of 10 CFR 26.

c. Conclusions

One NCV was identified with respect to the licensee's failure to rescreen FFD personnel under the provisions of 10 CFR 26.

S2 Status of Security Facilities and Equipment

S2.1 Protected and Vital Area Barriers

(Closed) LER 50-302/97-S01-00: Failure to Have a Protected Area Barrier in Place

(Closed) LER 50-302/97-S02-00: Failure to Compensate for a Protected Area Zone That Was Placed in Access Mode Due to Maintenance

a. Inspection Scope (81700)

The inspector reviewed LER 50-302/97-S01-00, concerning a security breach in the Protected Area via the Circulating Water System (CWS) and LER 50-302/97-S02-00, regarding an uncompensated security zone that had been placed out of service for maintenance.

b. Observations and Findings

On January 20, 1997, a security officer, on routine rounds, questioned an opened waterbox manway. Upon further investigation, the licensee discovered a previously unidentified protected pathway into the protected area. Upon discovery, the pathway was posted with a security officer. The breach remained posted for approximately 7.5 hours. The licensee determined that security was not compromised during this event.

Section 3.1 of the licensee's Physical Security Plan (PSP) states in part that the Protected Area is located within the Owner-Controlled Area and is enclosed by physical barriers.

Root cause analysis performed by the licensee determined that the licensee failed to identify and understand the security function of plant components affected by infrequently performed maintenance activities. The inspector noted that in 1985, the Amertap condenser cleaning system was installed; however, an inadequate pathway review was conducted. In 1989, a security evaluation failed to identify the pathway that is the subject of this discussion. Additionally, a 1994 unprotected pathway event only addressed the specific conditions of that event, rather than a total review of all security barrier components. This unprotected pathway discovered on January 20, 1997, could possibly have been discovered as early as 1985. The licensee determined that the following procedures were inadequate:

- OP-604, Circulating Water System, did not address that security notification was necessary if less than two circulating pumps were operating.

- CP-113A, Work Request Initiation, and CP-113B, Work Package Control, did not address recognition of security barriers status, which may be altered due to changing plant conditions or maintenance activities.

- SS-201, Security Force Personnel General Orders, Duties, and Responsibilities, did not address this pathway.

The inspector reviewed the licensee's corrective action, which was complete at the time of this inspection:

- Rather than using ty-wraps as a means of providing security to the plant's waterboxes, the licensee had placed substantial chains and security controlled locks on all waterboxes.

- A Security Barrier Component List (SBCL) was established, to include all boundary valves and pipes with runs greater than 11 inches in diameter which pass from either an unprotected area into a protected area or from the protected area into a vital area.

- The identified components in the SBCL and the necessary security actions have been incorporated into security procedure SS-201, Security Force Personnel General Orders, Duties, and Responsibilities, Revision 31.

- Procedure OP-604 had been revised to have security notified when less than two circulating pumps are in operation.

- Procedures CP-113A and 113B have been revised to have security unlock the waterboxes when maintenance is required and to ensure barriers are in place at that time.

- Security conducted periodic briefings with plant personnel, with respect to security conditions that are related to plant components.

This licensee-identified and corrected violation is being treated as a NCV, consistent with Section VII.B.1 of the Enforcement Policy. This violation is identified as NCV 50-302/97-19-04, Failure to Have a Protected Barrier in Place. LER 50-302/97-S01-00 is closed.

On May 30, 1997, the licensee discovered that the Perimeter Intrusion Detection System (PIDS) was in access mode for a zone without compensatory measures in place and reported the event to the NRC (LER 50-302/97-S02-00). The licensee's PSP states in Section 6.1.1.1.1 that in the event of a partial alarm system failure, the following compensatory measures will be taken: station a member of the security force at the inoperable alarm location.

The zone was immediately placed in the secure mode; however, the zone had been left uncompensated for approximately 2.5 hours. The zone had been placed in access, due to maintenance work that began around 09:30 a.m. hours. However, upon completion of that work, the CAS operator failed to place the zone back in the secure mode. The error was later discovered by the same CAS operator.

The inspector reviewed the licensee's immediate and long term corrective actions, which were complete at the time of this inspection. Protected and vital area searches were conducted. No abnormalities were discovered, nor were there any unexplained alarms during the event. The uncompensated zone had closed circuit television operating at the time of the event. During shift turnover briefings, the importance of self-checking was emphasized. The process of placing and returning protected area zones from the access to secure mode was incorporated in SS-205, Alarm Station Operations. Additionally, the inspector noted the following measures which have been implemented to prevent reoccurrence:

- A status sheet for turnover purposes, which outlined equipment out of service, had been instituted.

- Alarm Station Operators have to inform the Shift Supervisor when security equipment is taken out of service.

- Three classes with Alarm Station Operators and their supervisors on lessons learned were conducted.

- Two all hands meetings with the security force was conducted.

This non-repetitive, licensee-identified and corrected violation is being treated as a NCV, consistent with Section VII.B.1 of the

Enforcement Policy. This violation is identified as NCV 50-302/97-19-05, Failure to Compensate a Zone.

c. Conclusions

Through observation, document review, and discussion with licensee representatives, the inspector determined that a NCV for failure to have a protected barrier in place occurred. The licensee's failure to compensate a zone in access also resulted in a NCV.

S8 Miscellaneous Security and Safeguards Issues

S8.1 Actions on Previous Inspection Findings (92904)

(Closed) VIO 50-302/96-03-13: Unescorted Visitor Personnel Within the Protected Area

(Closed) VIO 50-302/96-09-02: Unescorted Visitor Personnel Within the Protected Area

In responses dated June 20, 1996, and November 4, 1996, respectively, the licensee documented corrective actions to these violations. The inspector verified that revised Security Procedure SS-207, Plant Entry and Exit Requirements, Revision 14, clearly outlined the process to transfer escort responsibilities among badged employees. During the corrective action process, the licensee reduced the escort/visitor ratio. The licensee also modified their Facility Access Log, visitor/escort questionnaire, and escort video to reflect the importance of visitor control. Two-sided badges had been developed by the licensee that identified an individual as an escort and outlined key escort control rules. The corrective action is considered adequate to close these two violations.

(Closed) VIO 50-302/96-07-01: Failure to Protect Safeguards Information

(Closed) EEI 50-302/97-03-02: Failure to Protect Safeguards Information

(Closed) EA 97-161 (01013): Failure to Protect Safeguards Information

(Closed) EA 97-161 (01023): Failure to Protect Safeguards Information

In responses dated October 4, 1996, and June 23, 1997, respectively, the licensee documented corrective actions to these violations. The inspector verified the following corrective actions were adequate and had been completed:

- An enhanced training session for those individuals responsible for the protection and control of Safeguards Information (SGI).
- SGI training for badged personnel was completed January 1997.
- A single-point of control for SGI inside the protected area and at

the Emergency Operations Center on Venable Street was established May 1997.

- Newly badged individuals received SGI briefings at General Employee Training and Access Control briefings.
- A continuing downgrade of existing SGI (as of December 17, 1996, 242 SGI documents had been declassified).
- The Personnel Access List (badged individuals allowed access to SGI), had been evaluated quarterly.
- Training for Security personnel on Security regulations with respect to SGI.
- Security is now part of the plant's Corrective Action Program (CAP), instead of having an internal tracking system of problems and events.

The inspector toured the SGI room and found it adequately controlled, documents appropriately locked, and access control measures in place. Through discussion with licensee representatives and a review of the Safeguard Event Logs, the inspector determined that no other loss of control or failure to protect SGI incidents had occurred. SGI that needs to be updated or revised is typed in the SGI room on a stand alone computer. Corrective action is considered adequate to close these two violations.

<u>(Closed) EA 97-012, VIO A(1) (01013): Failure to Implement Procedure for Central and Secondary Alarm Station Operations and Inadequate Procedure for Implemented Compensatory Measures</u>

In a response dated March 27, 1997, and supplemental response of May 21, 1997, the licensee outlined their corrective action. The inspector reviewed Security Procedure SS-205, Alarm Station Operations, Revision 17, dated October 22, 1997, to determine adequacy. The procedure clearly reflected the current security computer equipment and detailed the operational instructions. The inspector reviewed SS-208, Compensatory Measures, Revision 0, dated March 15, 1997. This procedure superseded SS-303, Compensatory Measures for Pre-Planned Maintenance. The inspector verified that SS-208 utilized an officer at a fixed post for a degraded vital area door, rather than allowing a roving patrol to compensate, as originally stated in SS-303. Corrective action is considered adequate to close this violation.

<u>(Closed) EA 97-012, VIO A(2) (01023): Failure to Respond to a Protected Area Alarm</u>

The licensee responded to this violation on March 27, 1997, and supplemented the response on May 21, 1997. The inspector reviewed records and verified that alarm station operator simulator training was completed on March 14, 1997. On routine observation of the central alarm station (CAS) and secondary alarm station (SAS) the inspector determined that the intelligent multiplexers (IMUXs) had an audible alarm to inform the operators when the cabinets had been opened. A configuration management procedure was developed. SS-307, Plant Security Computer, Revision 0, dated May 20, 1997, addressed post maintenance testing requirements. The inspector interviewed several CAS/SAS operators to determine their knowledge of alarm information that is displayed on the JC 6000. All operators were knowledgeable and understood their duties with respect to alarms and required response. Corrective action is considered adequate to close this violation.

<u>(Closed) EA 97-012, VIO A(3) (01033): Failure to Assess More than One Protected Area Alarm</u>

In responses dated March 27, 1997, and May 21, 1997, the licensee stated that a software change had been implemented on December 12, 1996, that allowed for automatic call-up for appropriate cameras during multiple protected area alarms, rather than having to manually call-up each camera. The inspector observed the licensee break three zones to test the assessment capability. All three zones were displayed automatically, showing the zones that had alarmed. Through discussion with licensee representatives, the inspector determined that Security now participates in the security modification process by attending Design Review Board meetings on a regular basis. Corrective action is considered adequate to close this violation.

<u>(Closed) EA 97-012, VIO A(4) (01043): Failure to Have a Protected Area Barrier in Place</u>

In responses dated March 27, 1997, and May 21, 1997, the licensee documented their proposed corrective actions: 1) ty-wraps on waterboxes to have controls in place prior to opening; 2) labels on waterboxes to remind workers to contact security, and 3) revised work order packages to ensure security receives notification. The inspector verified that all three corrective action commitments were in place at the time of this inspection. Due to another waterbox event, which identified a previously undiscovered pathway into the protected area (see Section S2.1 for more information), the ty-wraps have now been replaced with chains and security controlled locks. Corrective action is considered adequate to close this violation.

<u>(Closed) EA 97-012, VIO A(5) (01053): Failure to Secure the Arms Repository</u>

The licensee responded to this violation on March 27, 1997 and May 21, 1997, and identified the following corrective actions:

- The armory was posted with a security officer on December 16, 1996.

- Security equipment was secured in locked cabinets within the armory on December 20, 1996.

- The door lock to the armory was modified with a double key lock.

- An armory cage was installed April 30, 1997, as an enhancement to provide protection of the firearms and ammunition.

Through observation and discussion with licensee representatives, the inspector verified the above stated corrective actions were completed. Corrective action is considered adequate to close this violation.

(Closed) EA 97-012, VIO B (01063); Failure to Submit Physical Security Plan Changes Within Two Months

The licensee's corrective actions specified in responses to the NRC dated March 27, 1997, and May 21, 1997, stated that Administrative Instruction (AI) 800, Conduct of Plant Security, would be revised to reflect the guidance in Generic Letter 95-08. The inspector reviewed and evaluated AI-800, Revision 32, dated December 16, 1997. Guidance from Generic Letter 95-08 was appropriately incorporated so that the licensee may now produce an accurate evaluation and submittal of 10 CFR 50.54(p) plan changes. The inspector reviewed PSP changes 6-10 through 6-17 to determine if the provisions of 10 CFR 50.54(p)(2) had been met. All plan changes were submitted to the NRC within two months of implementing the changes. Corrective action is considered adequate to close this violation.

(Closed) IFI 50-302/96-18-08; Adequacy of the Licensee's Corrective Actions for Security Audit Findings

The inspector reviewed the third quarter security audit, conducted during the period of July 9, 1996, through November 15, 1996. All issues identified were appropriately dispatched in the corrective action system and answered appropriately. Corrective action reviewed by the inspector was adequate and timely. The review of this additional information closes this IFI.

(Closed) VIO 50-302/97-03-01; Failure to Maintain Control of a Security Badge (EA 97-008)

The licensee's response dated May 16, 1997, documented their corrective action to this violation. The implementation of biometrics on July 24, 1997, should preclude this type of event from occurring in the future, because badges are now allowed to be taken offsite. The inspector reviewed the Security Event Logs from May 16, 1997 to July 24, 1997, to determine if other similar events had occurred. There were no other documented events that security badges had been taken offsite during this time period. Corrective action is considered adequate to close this violation.

(Closed) VIO 50-302/97-15-01; Failure to Terminate Access of a Contract Employee

The inspector reviewed the licensee's corrective action which was documented in a response dated November 5, 1997. In order to better determine when access control receives a request to terminate an individual's badge, on September 26, 1997, the licensee had purchased a metered and dated time stamp. Security can now log in when the request was received. The inspector reviewed access termination forms received by access control for the last 30 days (August - September 1997). No additional examples of non-compliance were identified. The inspector reviewed the newly established electronic version of SS-207, Plant Entry and Exit Requirements. Computer access to SS-207 is readily available to supervisors. A form in SS-207, containing appropr. .e termination information, is then transferred to access control, who downloads everyday and makes the appropriate changes. Additionally, the inspector reviewed The Daily Bulletin, dated November 26, 1997, which described the new electronic form and its availability. Corrective action is considered adequate to close this violation.

V.Management Meetings

X1 Exit Meeting Summary

The inspection scope and findings were summarized on January 5, 1998. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

X3 Management Meeting Summary

X3.1 On December 2, 1997, a meeting was held at NRC Headquarters to conduct working level technical discussions with the licensee relating to seismic qualification of mechanical and electrical equipment (A46 issue) and large bore piping and pipe support calculations. A separate meeting summary was issued on December 11, 1997.

X3.2 On December 11 and 12, 1997, meetings were held in Crystal River to discuss the licensee's progress toward readiness for restart and the status of several Engineering issues. A separate meeting summary was issued on December 18, 1997.

PARTIAL LIST OF PERSONS CONTACTED

Licensees

R. Anderson, Senior Vice President, Energy Supply
J. Baumstark, Director, Quality Programs
J. Cowan, Vice President, Nuclear Operations
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, Nuclear Operations
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

P. Fillion, Reactor Inspector, Region II (December 1 through 10, 1997)
E. Lea, Project Engineer, Region II (December 1 through 5, December 15 through 19, 1997)
M. Miller, Reactor Inspector, Region II (December 1 through 12, 1997)
R. Schin, Reactor Inspector, Region II (December 1 through 12, 1997)
L. Stratton, Physical Security Specialist, Region II (December 15 through 19, 1997)
M. Thomas, Reactor Inspector, Region II (December 1 through 12, 1997)
J. York, Reactor Inspector, Region II (December 1 through 12, 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 81502: Fitness for Duty for Power Reactors
IP 81700: Physical Security Program for Power Reactors
IP 92901: Followup - Operations
IP 92902: Followup - Maintenance
IP 92903: Followup - Engineering
IP 92904: Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
NCV	50-302/97-19-01	Closed	Failure to Follow Procedure for the Use of Immediate Work Copy Changes to Procedures. (Section 07.2)
NCV	50-302/97-19-02	Closed	Failure to Validate Operating Curves. (Section E8.24)
NCV	50-302/97-19-03	Closed	Failure to Rescreen FFD Personnel Under the Provisions of 10 CFR 26. (Section S1.1)
NCV	50-302/97-19-04	Closed	Failure to Have Protected Barrier in Place. (Section S2.1)
NCV	50-302/97-19-05	Closed	Failure to Compensate a Zone. (Section S2.1)

Closed

Type	Item Number	Status	Description and Reference
NCV	50-302/97-19-01	Closed	Failure to Follow Procedure for the Use of Immediate Work Copy Changes to Procedures. (Section 07.2)
URI	50-302/97-07-03	Closed	Reactor Building Liner Plate Degradation. (Section E8.1)
LER	50-302/97-16-00	Closed	Reactor Building Coatings Not Included in Sump Calculation. (Section E8.2)
LER	50-302/97-18-00	Closed	High Energy Line Break Could Result in Loss of Chilled Water to Control Complex Ventilation System. (Section E8.3)
IFI	50-302/97-14-03	Closed	Follow-up on Verification of ASME Sec. XI Valve Testing. (Section E8.5)
VIO	EA 96-365 VIO B Ex. 4 (02013)	Closed	Use of Unverified Calculations to Support Modifications. (Section E8.6)

VIO	50-302/95-21-03	Closed	Failure to Isolate the Class 1E From the Non-Class 1E Electrical Circuitry for the RB Purge Valves. (Section E8.7)
LER	50-302/95-25-00 50-302/95-25-02	Closed	Personnel Errors By Architect Engineer Result in Operation Outside Design Basis Due to Inadequate Safety/non-Safety Related Circuit Isolation. (Section E8.7)
IFI	50-302/96-201-17	Closed	Coordination of SLUR and Fuse Protection. (Section E8.8)
LER	50-302/96-19-00 50-302/96-19-01 50-302/96-19-02	Closed	Classification of Transfer Switch Causes Potential for Loss of Power to ES Status Lights. (Section E8.9)
LER	50-302/97-23-00	Closed	Design Engineering Process Allows Installation of Vital Bus Inverters Containing an Unanalyzed Trip Circuit. (Section E8.10)
LER	50-302/97-32-00	Closed	Inadequate Electrical Isolation on a Safety Related Power Supply Due to a Design Error. (Section E8.11)
LER	50-302/97-21-00 50-302/97-21-01	Closed	Loss of A Battery Leads to the Inability to Bypass ES Actuation Signals as a Result of Inadequate System Knowledge. (Section E8.12)
VIO	EA 97-330 (01013)	Closed	Unreviewed Safety Question Involving Added EDG Protective Trips. (Section E8.14)
LER	50-302/97-05-00	Closed	Unanalyzed Condition Regarding Small Break LOCA and Emergency Feedwater. (Section E8.15)
LER	50-302/95-09-00	Closed	Minimal Release During Sulfur Dioxide Delivery Causes Actuation of Toxic Gas Monitor Resulting in Control Room Emergency Ventilation Actuation. (Section E8.16)
LER	50-302/96-04-00	Closed	Control Complex Habitability Envelope (CCHE) Control Dampers Found Damaged and Leaking. (Section E8.17)

LER	50-302/97-22-00	Closed	Calculation Errors Associated With 50-302/97-22-01Control Complex Habitability Envelope Unfiltered Air Inleakage Could Allow Operator Dose Limits to be Exceeded. (Section E8.18)
VIO	50-302/96-09-05	Closed	Failure to Incorporate Design Information into Operations Procedures. (Section E8.19)
VIO	50-302/96-09-07	Closed	Inadequate Corrective Action for Implementation of EFIC Task Force Recommendations. (Section E8.20)
IFI	50-302/96-201-15	Closed	Verification of Motor Starting Data. (Section E8.23)
URI	50-302/96-201-03	Closed	Operating Curves 16, 17, and 18 in OP-103B are not Validated by Licensee. (Section E8.24)
NCV	50-302/97-19-02	Closed	Failure to Validate Operating Curves. (Section E8.24)
LER	50-302/97-30-00	Closed	Installation Error Resulted in a Containment Isolation Check Valve Disc Sticking in The Open Position. (Section E8.25)
VIO	50-302/97-14-04	Closed	Failure to Adequately Test HPI Valves MUV-23, 24, 25, and 26 Power Selector Switches. (Section E8.26)
VIO	EA 96-365 VIO A (01062)	Closed	Inadequate 50.59 Evaluation for Post-LOCA Boron Precipitation Control. (Section E8.28)
IFI	50-302/95-15-04	Closed	Code Requirement for Thermal Relief Valves on Decay Heat Removal Heat Exchangers. (Section E8.31)
LER	50-302/97-15-00 50-302/97-15-01	Closed	Non-Conservative Differential Pressure Rating for Letdown Line Inboard Containment Isolation Valves Prohibiting Safety Function Performance. (Section E8.32)
LER	50-302/97-28-00	Closed	Flow Element Accuracy Uncertainty May Be Greater Than Assumed Due to a Design Error. (Section E8.33)

LER	50-302/97-39-00	Closed	Unqualified Material Left in Reactor Building During Construction Could Affect Post-LOCA Cooldown Capability. (Section E8.34)
URI	50-302/96-201-02	Closed	Net Positive Suction Head for Building Spray Pump. (Section E8.35)
LER	50-302/97-24-00	Closed	Feedwater Valves Do Not Meet the Requirements for Main Steam Line or Feedwater Line Break Event Mitigation Due to Inadequate Design and Calculation Errors. (Section E8.36)
LER	50-302/97-08-00	Closed	Potential of HP. Pump Recirculation Capability Resulting in Possible MUT Overflow or Possible Pump Failure. (Section E8.37)
IFI	50-302/96-17-02	Closed	Potential for HPI Recirculation Resulting in MUT Overflow. (Section E8.37)
LER	50-302/97-04-01	Closed	Thermal Relief Valves Inside Containment Do Not Meet Requirements For A Design Basis Accident. (Section E8.38)
LER	50-302/97-12-00	Closed	Industrial Cooling (CI) System Penetrations Not Designed for Containment Isolation. (Section E8.38)
IFI	50-302/96-08-02	Closed	Reactor Building Cavity Cooling Piping Thermal Relief Protection. (Section E8.38)
VIO	50-302/96-07-02	Closed	Failure to Complete Screening Elements for Fitness for Duty Personnel. (Section S1.1)
NCV	50-302/97-19-03	Closed	Failure to Rescreen FFD Personnel Under the Provisions of 10 CFR 26. (Section S1.1)
NCV	50-302/97-19-04	Closed	Failure to Have Protected Barrier in Place. (Section S2.1)
NCV	50-302/97-19-05	Closed	Failure to Compensate a Zone. (Section S2.1)

VIO	50-302/96-03-13	Closed	Unescorted Visitor Personnel Within the Protected Area. (Section S8.1)
VIO	50-302/96-09-02	Closed	Unescorted Visitor Personnel Within the Protected Area. (Section S8.1)
VIO	50-302/96-07-01	Closed	Failure to Protect Safeguards Information. (Section S8.1)
EEI	50-302/97-03-02	Closed	Failure to Protect Safeguards Information. (Section S8.1)
VIO	EA 97-161 VIO A (01013)	Closed	Failure to Protect Safeguards Information. (Section S8.1)
VIO	EA 97-161 VIO B (01023)	Closed	Failure to Protect Safeguards Information. (Section S8.1)
VIO	EA 97-012 VIO A(1) (01013)	Closed	Failure to implement Procedure for Central and Secondary Alarm Station Operations and Inadequate Procedure for Implemented Compensatory Measures. (Section S8.1)
VIO	EA 97-012 VIO A(2) (01023)	Closed	Failure to Respond to a Protected Area Alarm. (Section S8.1)
VIO	EA 97-012 VIO A(3) (01033)	Closed	Failure to Assess More than One Protected Area Alarm. (Section S8.1)
VIO	EA 97-012 VIO A(4) (01043)	Closed	Failure to Have a Protected Area Barrier in Place. (Section S8.1)
VIO	EA 97-012 VIO A(5) (01053)	Closed	Failure to Secure the Arms Respository. (Section S8.1)
VIO	EA 97-012 VIO B (01063)	Closed	Failure to Submit Physical Security Plan Changes Within Two Months. (Section S8.1)
IFI	50-302/96-18-08	Closed	Adequacy of the Licensee's Corrective Actions for Security Audit Findings. (Section S8.1)
VIO	50-302/97-03-01	Closed	Failure to Control a Security Badge (EA 97-008). (Section S8.1)
VIO	50-302/97-15-01	Closed	Failure to Terminate Access of a Contract Employee. (Section S8.1)

LER	50-302/97 S01-00	Closed	Failure to Have a Protected Area Barrier in Place. (Sections S2.1, S8.1)
LER	50-302/97-S02-00	Closed	Failure to Compensate for a Protected Area Zone. (Sections S2.1, S8.1)

Discussed

Type	Item Number	Status	Description and Reference
IFI	50-302/97-17-05	Open	Resolution of Improved Technical Specification Setpoint Program Deficiencies Prior to Entry Into Mode 4. (Section O7.4)
IFI	50-302/97-11-04	Open	Corrective Actions for approximately 4000 Precursor Ca. ds Not Tracked to Completion. (Section O8.1)
VIO	EA 95-126 VIO I.D.1 (05013)	Open	Design Controls Failed to Ensure Adequate Safety Margin for HPI Pumps for Certain LOCA Scenarios. (Section O8.2)
VIO	EA 95-126 VIO I.C.1 (03013)	Closed	Failure to Take Adequate Corrective Actions for Operator Concerns Regarding OP-103B, Curve 8, for MUT Pressure/Level Limits. (Section O8.2)
VIO	EA 95-126 VIO I.C.2 (04013)	Closed	Corrective Actions for an Inadequate Curve 8 Were Also Incorrect. (Section O8.2)
VIO	EA 95-126 VIO II.B (08014)	Closed	Failure to Take Adequate Corrective Actions for Tank Volumes/Level/Suction Point. (Section O8.2)
IFI	50-302/97-14-01	Open	Review of Operational Procedures Prior to Restart. (Section O8.3)
VIO	50-302/97-14-07	Open	Failure to Follow Foreign Material Exclusion Procedure Requirements. (Section M1.2)
NCV	50-302/97-21-01	Closed	Examples of Noncompliances in Design Control, 50.59 Evaluations, Procedure Adequacy, Reportability,

			and Corrective Actions That Are Subject to Enforcement Discretion. (Sections E8.2-E8.3, E8.9-E8.12, E8.17-E8.18, E8.25, E8.32-E8.34, E8.36-E8.38)
VIO	EA 96-365, 96-465, 96-527, VIO B (02013)	Open	Error in Design Calculations for Service Water Heat Loads. (Section E8.4)
LER	50-302/95-25-01	Closed	Personnel Errors by Architect Engineer Result in Operation Outside Design Basis Due to Inadequate Safety/non-Safety Related Circuit Isolation. (Section E8.7)
VIO	50-302/97-14-13	Open	Failure to Take Corrective Actions to Identify and Correct the Design Weaknesses Associated with Adequacy of the Past 10 CFR 50.59 Review for Positioning of DHV-34 and DHV-35 During Normal Operation. (Section E8.13)
VIO	EA 96-365, 96-465 96-527, VIO B Ex. 1 (02013)	Closed	Failure to Update Applicable Design Documents to Incorporate EFW Design Information. (Section E8.15)
URI	50-302/95-02-02	Open	Control Room Habitability Envelope Leakage. (Section E8.18)
VIO	50-302/97-02-03	Open	Adequacy of Procedures to Take the Plant from Hot Standby to Cold Shutdown from Outside the Control Room. (Section E8.21)
VIO	50-302/97-16-03	Open	Failure to Design and Install Radioactive Waste Disposal System Piping as Described in the FSAR. (Section E8.22)
LER	50-302/97-38-00	Open	An Engineering Oversight Resulted in Operation Outside of the Design Basis for the Waste Disposal System. (Section E8.22)
LER	50-302/97-13-00	Open	Emergency Diesel Generator Room Could Exceed Maximum Design Temperature During Operation Due to Inadequate Room Cooling. (Section E8.27)

LER	50-302/97-19-00 50-302/97-19-01	Open	Elevated EDG Supply Air Temperatures Due to EDG Radiator Discharge Air Recirculation Effect. (Section E8.27)
LER	50-302/97-27-00	Open	Failure to Add Antifreeze to the Diesel Generator Coolant Radiators May Render Emergency Diesel Generator Inoperable During Sub-Freezing Temperatures. (Section E8.27)
VIO	50-302/97-17-01	Open	Failure to Conduct an Adequate Unreviewed Safety Question Evaluation for a Modification Functional Test. (Section E8.37)

LIST OF ACRONYMS USED

AB - Auxiliary Building
ACP - Access Control Procedure
AI - Administrative Instruction
ANSS - Assistant Nuclear Shift Supervisor
AP - Abnormal Procedures
BSP - Building Spray Pump
BWST - Borated Water Storage Tank
CCHE - Control Complex Habitability Envelope
CFR - Code of Federal Regulations
CFT - Core Flood Tank
CREVS - Control Room Emergency Ventilation System
CR3 - Crystal River Unit 3
CT - Current Transformers
CWS - Circulating Water System
DH - Decay Heat
DHP - Decay Heat Pump
DHV - Decay Heat Valve
DIR - Design Input Requirements
DNPO - Director, Nuclear Plant Operations
DP - Differential Pressure
EA - Enforcement Action
ECCS - Emergency Core Cooling System
EDG - Emergency Diesel Generator
EFIC - Emergency Feedwater Initiation and Control
EFP - Emergency Feed Pump
EFPY - Effective Full Power Year
EFW - Emergency Feedwater
EOC - Extent of Condition
EOP - Emergency Operating Procedure
ES - Engineered Safeguards
ESAS - Engineered Safeguards Actuation System
FCN - Field Change Notice
FFD - Fitness For Duty
FME - Foreign Material Exclusion
FMEA - Failure Modes and Effects Analysis
FPC - Florida Power Corporation
FSAR - Final Safety Analysis Report
FWLB - Feedwater Line Break
GL - Generic Letter
HELB - High Energy Line Break
HPI - High Pressure Injection
IFI - Inspection Follow-up Item
IMUX - Intelligent multiplexers
IN - NRC Information Notice
IOC - Interoffice Correspondence
IPAP - Integrated Performance Assessment Process
IR - Inspection Report
ITS - Improved Technical Specifications

IWCC - Immediate Work Copy Change
JCO - Justification for Continued Operation
Kv - Kilovolts
Kw - Kilowatts
LAR - License Amendment Request
LER - Licensee Event Report
LOCA - Loss of Coolant Accident
LOOP - Loss of Offsite Power
LPI - Low Pressure Injection
LTOP - Low Temperature Overpressure Protection
MAR - Modification Approval Record
MCAP - Management Corrective Action Plan
MFP - Main Feedwater Pump
MOV - Motor Operated Valve
MSIV - Main Steam Isolation Valve
MSLB - Main Steam Line Break
MTC - Moderator Temperature Coefficients
MUP - Make-up Pump
MUT - Make-up Tank
MUV - Make-up Valve
MWST - Miscellaneous Waste Storage Tank
NCV - Non-Cited Violation
NEP - Nuclear Engineering Procedure
NOTES - Nuclear Operations Tracking and Expediting System
NPSH - Net Positive Suction Head
NQA - Nuclear Quality Assessments
NRC - Nuclear Regulatory Commission
NRR - Office of Nuclear Reactor Regulation
NSAT - Nuclear Safety Assessment Team
NSM - Nuclear Shift Manager
NSS - Nuclear Shift Supervisor
OCR - Operability Concerns Resolution
ODCM - Offsite Dose Calculation Manual
OI - Operations Instruction
OP - Operating Procedure
OTSG - Once Through Steam Generator
PC - Precursor Card
PCSC - Precursor Card Screening Committee
PDR - Public Document Room
PIDS - Perimeter Intrusion Detection System
PM - Preventive Maintenance
PMRG - Plant Modification Review Group
PMT - Post Maintenance Test
PORV - Power Operated Relief Valve
PRC - Plant Review Committee
PSID - Pounds Per Square Inch Differential
PSIG - Pounds Per Square Inch Gauge
PSP - Physical Security Plan
PT - Potential Transformer
QA - Quality Assessments
QC - Quality Control
QPA - Quality Program Assessment

QPS - Quality Programs Surveillance
RB - Reactor Building
RCBT - Reactor Coolant Bleed Tank
RCDT - Reactor Coolant Drain Tank
RCP - Reactor Coolant Pump
RCS - Reactor Coolant System
REA - Request for Engineering Assistance
RG - Regulatory Guide
RW - Nuclear Services and Decay Heat Sea Water
SA - Safety Assessment
SBCL - Security Barrier Component List
SBLOCA - Small Break Loss of Coolant Accident
SCBA - Self Contained Breathing Apparatus
SEL - Security Event Log
SER - Safety Evaluation Report
SGI - Safeguards Information
SLUR - Second Level Undervoltage Relays
SP - Surveillance Procedure
SRO - Senior Reactor Operator
SRP - Standard Review Plan
SRR - System Readiness Review
SRST - Spent Resin Storage Tank
SS - Security Procedure
SSFI - Safety System Functional Inspection
STA - Shift Technical Advisor
SW - Nuclear Services Closed Cycle Cooling
TP - Test Procedure
TS - Technical Specification
TSCRN - Technical Specification Change Request Notice
UHS - Ultimate Heat Sink
URI - Unresolved Item
USQ - Unreviewed Safety Question
USQD - Unreviewed Safety Question Determination
UT - Ultrasonic Test
VIO - Violation
WCS - Work Control Center Supervisor
WDS - Waste Disposal System
WGDT - Waste Gas Decay Tank
WR - Work Request

SUMMARY OF EGDG-1A TEST RUNS

Type of Run	Start Date/Time	Stop Date/Time	Parameter(s) of Interest
Unloaded 30 minute run per OP-707	12/7/97 1:52 pm	12/7/97 2:05 pm	Run stopped due to excessive clutch slippage
Unloaded 30 minute run per OP-707	12/7/97 4:13 pm	12/7/97 4:16 pm	Run stopped due to oil leak in fan compartment
Unloaded 30 minute run per OP-707	12/8/97 6:20 am	12/8/97 7:02 am	Completed 30 minute run
Loaded run per MAR 97-05-15-01 TP-4. Loaded at 2625 - 2825 kw for 24 hours. Loaded at 3100 - 3175 kw for 14 hours. Loaded at 3300 - 3375 kw for 20 hours. Loaded at 3325 - 3375 kw for 2 hours.	12/8/97 11:05 am	12/11/97 3:10 am	Closed breaker 3209 at 11:22 am on 12/8/97 and opened at 3:09 am on 12/11/97. Breaker 3209 opened for full load reject test.
Hot condition fast start Loaded 2625 - 2825 kw for 1 hour	12/11/97 3:16 am	12/11/97 5:34 am	Closed breaker 3209 from 3:35 am until 5:27 am on 12/11/97
Fast start Loaded 2625 - 2825 kw for 1 hour	12/11/97 10:12 am	12/11/97 12:29 pm	Closed breaker 3209 from 10:28 am until 12:24 pm
Unloaded run per OP-707	12/14/97 12:42 am	12/14/97 1:00 am	Run to support PM-123.
Loaded run per OP-707 at approximately 2725 kw	12/14/97 4:36 am	12/14/97 10:30 am	Closed breaker 3209 from 5:36 am until 10:27 am
Slow start per SP-354A	12/16/97 2:31 am	12/16/97 9:41 am	Closed breaker 3209 from 3:00 am until 9:38 am on 12/16/97
Fast start per SP-354A	12/16/97 11:22 pm	12/17/97 1:30 am	Closed breaker 3209 from 11:37 pm on 12/16/97 until 1:30 am on 12/17/97
Slow start per OP-707 for equipment reliability and vibration testing on right angle gear drive	12/17/97 5:30 am	12/17/97 7:57 am	Closed breaker 3209 from 5:43 am until 7:54 am on 12/17/97.