EXECUTIVE SUMMARY

Crystal River 3 Nuclear Station
NRC Inspection Report 50-302/97-11

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covered a 5-week period of resident inspection; in addition, it included the results of announced inspections by regional reactor inspectors and a visiting resident inspector.

Operations

The licensee's oversight of reactor coolant system draindowns was very good, but an Inspector Follow-up Item (IFI 50-302/97-11-01) was opened due to unreliable level indications and poor control of their configuration. The use of a new Infrequent Evolution procedure was a good initiative because the licensee's management representative performed a valuable role in identifying and prioritizing the problems encountered (Section 01.2).

A Violation (VIO 50-302/97-11-02) was identified when a portion of the Once Through Steam Generators was restored to service without being controlled by a procedure, resulting in a Reactor Coolant System indicated level decrease caused by failure to secure the ventilation (Section 01.3).

The inspectors concluded that Operations ownership, communications, and performance remained a challenge to the licensee, but licensee management was aggressively pursuing the causes of the problems in an effort to improve performance. The Operations Leadership Plan was a comprehensive and self-critical initiative to focus on Operations' problems and develop appropriate goals for the future (Section 04.1).

A Non-Cited Violation (NCV 50-302/97-11-03) was identified for four corrective action procedures that did not require that all corrective actions for conditions adverse to quality be documented in quality records (Section 07.1).

A weakness was identified in the licensee's corrective action program in that there was no overall procedure or guidance describing what processes were acceptable for tracking and documenting corrective actions for conditions adverse to quality (Section 07.1).

The inspectors concluded that the licensee's process for assuring the completion of corrective actions for grade A and B Precursor Cards (PCs) was adequate. For Grade C and D PCs, previous procedures did not require tracking or documenting corrective actions. The inspectors opened an Inspector Follow-up Item (IFI 50-302/97-11-04) for further NRC review of approximately 4000 level C and D PCs that were closed without tracking the completion of corrective actions. The licensee stated plans to conduct a Quality Assurance (QA) audit of the corrective actions for these PCs (Section 07.1).

The inspectors assessed that QA audits and assessments of corrective actions overall were good. These audits and assessments had reviewed a broad scope of areas and had many findings. These findings were well described in the

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reports, well documented in Precursor Cards, and followed up in subsequent QA audits (Section O7.1). While closing two open items, the inspector concluded the licensee's safety analysis group exhibited a good level of skepticism when evaluating and determining a completed 1996 safety evaluation was inadequate. A problem found with inaccurate Design Input Record information was considered another example of the already reported weaknesses in the licensee's old design control process (Section O8.4).

Maintenance

A Non-Cited Violation (NCV 50-302/97-11-05) was identified for failure to review and comply with work instructions and ensure a clearance was obtained prior to removal of an auxiliary steam valve (Section M1.1).

The inspector reviewed the licensee's root cause analysis and corrective actions and found that they addressed the immediate problem. Troubleshooting was continuing, using MP-531, to identify and address the problem with the high bearing temperature. The resolution to this issue will be addressed as part of the restart issue for the building spray pumps (Section M1.2.

Engineering

The licensee established a full, baseline database of all known Once Through Steam Generator (OTSG) conditions and verified that the OTSGs were in very good condition for future operation (Section E1.1).

The licensee made progress in its efforts to resolve an issue concerning net positive suction head for the emergency core cooling system pumps when the spent fuel pool was running in recirculation to the borated water storage tank (Section E1.2).

The Modification Approval Record packages reviewed were technically adequate and were being implemented in accordance with licensee requirements and NRC regulations (Section E1.3).

Current design control procedures generally provided adequate controls for implementation of the licensee's design control process (Section E1.3).

The majority of the PEERE evaluations reviewed by the inspector were appropriately implemented and completed. However, an example of a modification made to critical characteristics of a safety related system was found to have been inappropriately completed as a PEERE evaluation, resulting in a violation. Weaknesses in the program existed which allowed the process to be inappropriately implemented, as was the case with the building spray pump impellers. This modification was identified as requiring the increased review and approval associated with the MAR process (Section E3.1).

The Nuclear Quality Assessments section has been active and effective in identifying continued weaknesses and areas for improvement in the licensee's engineering activities and design control process (Section E7.1).

The inspectors concluded that the licensee's System Restart Readiness Reviews were well organized and clearly documented. Many potential problems were identified, and these problems were adequately evaluated and entered into the Precursor Card system for tracking of corrective actions. Restart items were clearly identified. The reviews appeared to be thorough but were not fully comprehensive. An NRC Safety System Functional Inspection is scheduled to provide a more detailed review of this area prior to restart (Section E7.2).

A Violation (VIO 50-302/97-11-07) was identified for the licensee's removal of the reactor coolant system water quality requirements from the Final Safety Analysis Report (Section E8.4).

A Non-Cited Violation (NCV 50-302/97-11-08) was identified for an inadequate 50.59 evaluation for an abnormal procedure revision (Section E8.6).

The licensee had an effective program to meet the intent of Generic Letter 96-01, Testing of Safety-Related Logic Circuits (Section E8.8).

The inspectors concluded that the licensee was in the process of completing the implementation of an effective Class 1E DC Power FMEA program (Section E8.9).

Based on the limited review performed by the inspector, no problems were noted with the licensee's commercial grade item dedication process (Section E8/10).

Plant Support

The Biometrics hand geometry system was implemented, allowing security badges to be taken offsite. The inspector concluded the licensee effectively prepared and executed the change (Section S1.1).

The inspector concluded that the number of degraded fire protection features was high but significant action had been taken to reduce the number of open fire prevention related maintenance work orders. The material condition of the fire protection components was satisfactory and the fire brigade equipment was properly stored and well maintained. Implementation of the surveillance and test procedures in the fire protection area was satisfactory (Section F2.1).

Although revisions had to be made on two of the test procedures reviewed to ensure they met the appropriate test objectives, the licensee had already identified the discrepancies and was tracking their correction (Section F2.2).

The plant procedures did not require an annual physical examination for each fire brigade member. Physicals for some fire brigade members were required every four years. The current revisions of the pre-fire plans and fire hazards analyses were not up to date (Section F3.1).

Implementation of the procedures for the control of ignition sources and transient combustibles was good. General housekeeping was also good, considering the large amount of work in process as the result the long term maintenance and modification outage. An effective program was in place to meet the compensatory measures required for degraded or inoperable fire protection equipment (Section F3.1).

The fire brigade organization and training met the requirements of the site procedures, and implementation of the training program was very good. Performance by the fire brigade during an observed drill was good (Section F5.1).

Adequate coordination and oversight were provided over the facility's fire protection program; however, the CR-3 Fire Protection Plan had not been revised to conform to the recent reorganization of the facility's management structure that was in place at time of this inspection (Section F6.1).

The QA audits conducted in late 1996 and in 1997 were detailed and comprehensive and identified a significant number of fire protection program discrepancies. Corrective action was being implemented to resolve these issues (Section F7.1).

The licensee's initial evaluations for fire protection related NRC Information Notices were weak, and appropriate corrective actions had not been initially identified (Section F8.2).

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																			
	04.1	07.1	08.1	08.2	08.4	M8.1	E1.2	E1.3	E7.1	E7.2	E8.1	E8.2	E8.3	E8.4	E8.5	E8.6	E8.7	E8.8	E8.9	R8.1
Management Oversight	G	A		G	A	A		A	G	G		G	A	I	G		G	G	G	
Engineering Effectiveness		A	A	A	G		G	A		A	A	G	G		G			G	G	A
Knowledge of Design Basis		A	A		A		G	A		A	A	G	A		G	I		G	G	A
Compliance With Regulations	A	A		A	G	A	G	A	G	A			A	I	G	I	A	G	G	
Operator Performance	A			A	G											I				

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

Section 04.1: Operator Performance and Communication Observations

Section 07.1: Corrective Action Program Effectiveness

Section 08.1: (Closed) VIO 50-302/95-16-03: Inadequate Procedures for Operation of the Makeup Pump 1A Cooling Water

(Closed) LER 50-302/95-010-01: Inadequate Procedure Causes Low Cooling Water Flow to Makeup Pump Resulting in Operation Outside the Design Basis

Section 08.2: (Closed) VIO 50-302/96-20-01: Failure to Adhere to Reactor Coolant System Cooldown Limits

Section 08.4: (Closed) IFI 50-302/96-03-15: HPI Flow Indicator 50.59 and Tech Spec Bases Change

(Closed) LER 50-302/96-07-01: HPI Line Break With Loss of Battery Could Result in Reliance on Inadequate Accident Mitigation Instrumentation

Section M8.1:	(Closed) VIO 50-302/96-20-02: Failure to Follow Procedure AI-400C for Review and Development of Maintenance Procedure PM-191
Section E1.2:	NPSH Concern with ECCS Pumps
Section E1.3:	Design Control Process
Section E7.1:	Quality Assurance Audits and Surveillances
Section E7.2:	System Restart Readiness Reviews
Section E8.1:	(Open) URI 50-302/96-201-04: Nonsafety-Related Positioners on Safety-Related Valves
Section E8.2:	(Closed) IFI 50-302/96-201-12: Conduit Sizing Criteria - Jamming Ratio Not Considered
Section E8.3:	(Open) VIO 50-302/96-09-05: Failure to Incorporate Design Information into Operations Procedures
Section E8.4:	(Closed) URI 50-302/97-02-02: Deletion of Water Quality Requirements from the FSAR
Section E8.5:	(Closed) Violation 50-302/97-02-04: Failure to Conduct TS Logic Testing
Section E8.6:	(Closed) URI 50-302/97-05-02: 50.59 Safety Evaluation does not Address Operation of the Atmospheric Dump Valves from the Remote Shutdown Panel During an Appendix R Fire Event
Section E8.7:	(Closed) URI 50-302/97-05-04: LER and Violation not Supplemented by Date Stated in Licensee Responses
Section E8.8:	(Open) NRC Generic Letter 96-01: Testing of Safety-Related Logic Circuits
Section E8.9:	DC System Failure Modes & Effects Analysis (FMEA)
Section R8.1:	(Closed) Restart Item RMG 29/30: Seismic Mounting of HR Rad Monitor (FPC Restart Issue D-19)

Report Details

Summary of Plant Status

The unit remained in Mode 5 throughout the inspection period, continuing in the outage that began on September 2, 1996. The reactor coolant system (RCS) remained vented to atmosphere and filled to a normal pressurizer level with once-through steam generator (OTSG) nozzle dams installed to support OTSG eddy current tube inspections, tube end repairs, and tube plugging until July 25, 1997. On July 27, 1997, the RCS was drained to a midloop condition to perform radiographic inspections of high pressure injection nozzle welds. On July 28, 1997, the RCS was refilled up to a reduced inventory condition to remove the nozzle dams and on July 29, 1997, was then filled to a normal pressurizer level, all vent openings were closed, and a nitrogen overpressure was placed on the pressurizer. Both OTSG secondary sides remained completely drained until July 31, 1997, when the B OTSG was refilled and made available as a natural circulation heat sink following main steam isolation valve refurbishment. The A OTSG was filled on August 10, 1997. The main generator rotor was removed and shipped offsite to a vendor for required modifications and repairs on August 3, 1997.

Work on some major physical modifications related to the licensee's restart efforts continued or commenced this report period. On August 3, 1997, the A Emergency Diesel Generator was removed from service to commence a complete replacement of the radiator and upgrades to cooling airflows. Other ongoing modifications included the Feedwater Pump 7 Backup Diesel Power Supply and overpressurization chambers for containment penetration isolations to address concerns in NRC Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions. Modifications to the Emergency Feedwater (EFW) cavitating venturis and EFW motor-operated cross-tie Valve EFV-12 were completed.

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, use of procedures, log reviews, system lineup verifications, and review of clearance tagging processes. Significant observations are discussed in the following paragraphs.

O1.2 Reactor Coolant System Draindown Controls

a. Inspection Scope (71707)

The inspector reviewed the licensee's process and observed their performance of RCS draindown activities to reduced inventory that was performed July 26, 1997, through July 29, 1997, to support injection line nozzle radiographic testing and to remove OTSG nozzle dams. A specific problem caused by OTSG ventilation on July 28, 1997, is discussed in Section O1.3.

b. Observations and Findings

The licensee drained the RCS on July 27, 1997, to a low level of 129 feet, 4 inches, which was below their mid-loop definition of 129 feet, 6 inches, to perform radiography of two High Pressure Injection (HPI) nozzle welds for scheduled inspections. They then refilled the RCS to approximately 131 feet 6 inches (reduced inventory is less than 132 feet) to remove the OTSG nozzle dams. On July 29, 1997, the Reactor Coolant Pump (RCP) J-legs were then refilled; the RCS was restored to a normal pressurizer level; remaining primary vent openings were closed, and a nitrogen overpressure was established on the pressurizer.

The inspector observed that the licensee had again assigned a single, accountable Operations individual to coordinate the draindown activities. The licensee also was testing a newly developed draft Administrative Instruction (AI) 550, Infrequently Performed Test and Evolutions (IPTE), necessitating a member of management to be assigned for continuous oversight of the evolution. The inspector noted quality assurance auditors monitoring the evolution and several other members of licensee management observing to ensure licensee expectations were being implemented. This resulted in effective and consistent pre-job briefings and apparent good preparation for the draindown. The inspector observed that the licensee did use an effective, simple operator aid showing relative RCS levels and reference points which addressed earlier inspector draindown concerns documented in Inspection Report (IR) 50-302/97-08. However, the preparation and performance of the draindown and refill evolution revealed several notable problems with RCS level indications. First, a 7/8 inch error on the tygon tube level indication tape reference was found by surveyors benchmarking elevations in the reactor building. The licensee initiated Precursor Card (PC) 97-5514 and Request for Engineering Assistance (REA) 97-0887 to disposition the difference formally. Several minor but unexplained oscillations in the two tygon level indicators, a failure of the tygons to track level change, and slow response of the tygons to level changes were experienced in both this draindown and the previous evolution in June. The oscillations often would occur in only one of the two tygons, indicative of a venting problem. The licensee was always able to be fixed by flushing the level indicators, but the number of problems raised concerns about the reliability of the indicators.

Another problem the inspector observed on July 27, 1997, was the lack of configuration control and awareness by the operators as to where the tygons were aligned. The two tygon tubes could be aligned to either the A or B OTSG or the reactor vessel. Prior to the draining on July 27, 1997, the shift supervisor stated the tygons were both lined up to indicate reactor vessel level. However, after verifying the lineup following one of the aforementioned slow responses of the tygon tubes, the licensee confirmed that one of the tubes was aligned to the A OTSG. The inspector determined that the licensee had not verified the tygon lineup prior to commencing the draindown and that their procedures did not require it. The licensee had assumed the lineup was correct, as it was left following the June draindown. The licensee also informally tracked the alignment of tygon level indication by the use of yellow sticky notes on the control room television monitor aligned to view the tygons. This contributed to the shift supervisor's misunderstanding as to where the indicators were aligned. The IPTE management representative overseeing the draindown also noted these problems and stopped the draining until the cause of the level indicator problems was corrected. He noted that the position of the common root isolation valves to both the tygon and control board level indicators had not been reverified since the June draining. The management representative considered it prudent to reverify and tag these valves prior to each draining evolution and he initiated PC 97-5615, which summarized these problems for corrective action and a critique of the evolution was held by operations management. The inspector reviewed the critique and found it thorough and self-critical.

Another problem occurred during alignment of the RCS for filling and venting on July 30, 1997, when radioactive waste disposal (WD) system isolation valve WDV-405 was found open. Although normally open during power operation, it was expected to be closed per the RCS procedure in use. Being open resulted in aligning the Waste Gas Header to the pressurizer and RCS level indication vent path, which caused a significant and unexpected perturbation in indicated RCS level on the installed main control board instruments but not the tygon hose level indicators which remained vented to atmosphere. The safety consequence was minimal since it only indicated level changed, but it was another example of poor status control of equipment that resulted in uncertainty in indicators important to Operations. The licensee initiated PC 97-5612 to perform a root cause investigation and develop corrective action. Their preliminary review had not determined the cause of the mispositioned valve.

Both the inspector and licensee considered that these problems needed to be understood and resolved by Engineering and Operations and the alignment controlled prior to the next RCS draindown. The licensee was appropriately addressing the problems but had not resolved them yet. Consequently, the inspector identified these problems with level indication as Inspector Follow-up Item IFI 50-302/97-11-01, RCS Reduced Inventory Level Indication Problems.

c. Conclusions

The inspectors concluded that the licensee's oversight of the RCS draindown was very good, but that the level indication system was unreliable and poorly controlled. The use of the IPTE procedure was a good initiative. The licensee's management representative performed a valuable role in identifying and prioritizing encountered problems.

O1.3 RCS Level Indication Perturbation During Reduced Inventory Operation

a. Inspection Scope (71707)

On July 28, 1997, control room (CR) operators noticed movement of indicated RCS level while health physics technicians (HPTs) were helping to restore the OTSGs after various maintenance activities had been completed. The inspectors interviewed various licensee personnel in order to understand the occurrence.

b. Observations and Findings

The primary evolution being conducted at the time was an RCS fill and vent using Operating Procedure (OP)-301, Operation of the Reactor Coolant System. After the CR operators noticed movement of indicated RCS level, they contacted the HPTs who were to secure the OTSG ventilation. The HPTs reported that they had just secured the ventilation connected to the A OTSG upper hand-holes after the lower diaphragms and manways were in place. The RCS tygon tube indication remained within two inches of expected level, but the control board RCS level indication dropped approximately eight inches. Based on no detection of change in reactor building sump level and tygon tube indication remaining stable, CR operators believed there was no change in actual RCS level. But as a conservative measure, decay heat pump (DHP-1A) flow was reduced from 3000 gallons per minute (gpm) to the allowed decay heat flow (2500 gpm) based on lowest indicated level.

After discussions with Health Physics (HP) management, the inspectors determined that a controlled procedure was not actually used during this portion of OTSG restoration. HP guidelines were used instead for OTSG system outages. These guidelines were developed by HP for HPTs, to document the planning and scheduling efforts undertaken in preparation for OTSG system outages, with the goal of educating HPTs on the various tests and maintenance activities to be conducted on the OTSGs. Another goal of the guidelines was to complement the radiation work permits (RWPs) written to control work activities by providing management direction and expectations to the HPTs. The inspectors also determined that during the pre-job brief, the securing of the OTSG ventilation had been discussed but was not assigned to any particular person, and therefore was not performed when necessary.

c. Conclusions

This particular portion of OTSG restoration was conducted without being controlled by a procedure (OP-301 did not have explicit instructions to cover this evolution), and despite guidelines available to the HPTs conducting the evolution and what appeared to be a reasonably thorough pre-job brief, the securing of the OTSG ventilation failed to occur and resulted in an RCS indicated level decrease. Consequently, the inspectors considered this inadequate procedural guidance a violation and identified it as VIO 50-302/97-11-02, Inadequate Procedural Guidance for Quality-Related Work.

O4 Operator Knowledge and Performance

O4.1 Operator Performance and Communication Observations

a. Inspection Scope (71707)

The inspectors reviewed examples of Operations performance to assess the operators questioning attitudes and communications practices. Licensee management has focused on improving performance in these areas and Operations Readiness was identified as a restart restraint item on the NRC Restart List. The licensee recently developed an improvement plan, entitled "The Operations Leadership Plan," to focus attention and develop corrective actions to address each deficient area. The inspector reviewed the scope and goals of the plan.

b. Observations and Findings

Coordination and communications improvement between Operations and other site groups remained a significant priority with licensee management. More initiatives such as relocating the Shift Manager (SM) and Shift Supervisor offices and redefining their expectations, increasing management observation and oversight of shift activities, and refining Operation's processes such as clearances were evidence of this. The inspectors also observed several corrective action system PCs, submitted by operators, that indicate progress was being made in the area of ownership and questioning attitude.

However, the inspectors continued to observe coordination and performance problems which indicated the licensee has not yet fully corrected this area. Examples included an inadvertent trip of a reactor protection system channel on July 7, 1997, that was not logged by the shift supervisor, an operator assuming a high pressure injection flow meter erroneously indicated flow because it had not been vented without verifying that to be true, and a site drain (SD) pump motor that automatically started and ran uncoupled from the pump after removal of a clearance on July 30, 1997.

Clearances continued to be a source of operator performance problems, although the opportunities for error have sharply increased due to a rise in workload. On August 14, 1997, during a review of open clearances, Operations discovered that blue tags hung on power cords to the OTSG Nozzle Dam Instrument Panel could not be located. The panels were removed from service on approximately July 28, 1997, but the clearance was never removed. Neither the tags nor the cords could be located, and the licensee could not determine who inappropriately removed the tagged cords without authorization. The licensee appropriately initiated a root cause investigation via grade B level PC 97-5982. Several other problems were found by the licensee such as tags left hanging when the clearance was removed or tags removed but not signed for removal. The inspector did observe that the licensee was being very self-critical with tagging problems and was expecting PCs to be initiated on any problem, even if discovered within a normal clearance process second check. This was a significant departure from the previous licensee practice of only considering a discrepancy as a problem if all the barriers and checks in the system had been defeated. The licensee conducted an Operations department standdown on July 30, 1997, to focus attention on the above clearance problems and reiterate management expectations.

The Operations Leadership Plan was a comprehensive effort to improve Operations ownership and performance. The inspector noted the scope was very large and included many ambitious goals. The four main areas included Organizational Structure, Human Performance Improvements, Process Enhancements, and Facilities Upgrade. The majority of the elements had yet to be implemented but the inspector determined that the Operations management had developed a complete plan which included all known problems and weaknesses as well as appropriate goals for the future. The licensee was in the process of developing performance indicators to measure their progress implementing the plan. The inspector will evaluate the licensee's implementation in future inspections of outstanding restart items.

c. Conclusions

These observations caused the inspectors to conclude that Operations ownership and communications remained a challenge to the licensee, but licensee management was aggressively pursuing the causes of the problems in an effort to improve performance. The Operations Leadership Plan was a comprehensive and self-critical initiative to focus on Operations' problems and develop appropriate goals for the future.

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

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

O6 Operations Organization and Administration

O6.1 The following organizational changes were announced on July 18, 1997, effective mid-August:

- John Holden became Site Director, reporting to John Paul Cowan, Vice President, Nuclear Production. Individuals reporting to John Holden were:
  - Chip Pardee, Director, Nuclear Plant Operations
  - Bruce Hickle, Director, Restart,
  - Ken Lancaster, Manager, Projects, and
  - Dave Daniels, Manager, Corrective Action Program (Nuclear, Safety Assessment Team (NSAT))

- Mike Rencheck was hired from outside Florida Power Corporation and became the Engineering Director, the position previously held by John Holden. Mr. Rencheck reported directly to Roy Anderson, Senior Vice President, Nuclear Operations.

O6.2 The Nuclear Regulatory Assurance department's title was changed to Nuclear Regulatory Compliance, with Walter Pike as the manager.

O7 Quality Assurance in Operations

O7.1 Corrective Action Program Effectiveness

a. Inspection Scope (40500)

The inspectors reviewed the licensee's corrective action procedures and sample completed corrective actions for identified problems to verify that they complied with requirements of 10 CFR 50, Appendix B, and the licensee's approved Quality Assurance Program as described in the Final Safety Analysis Report (FSAR), Chapter 1.7. Inspectors reviewed samples of closed PCs to verify that corrective actions taken complied with licensee procedures, fully addressed problems, and were both effective and timely. In addition, inspectors reviewed recent licensee Quality Assurance Audits and Surveillances addressing this area. Also, inspectors discussed procedures and corrective actions with licensee engineering, operations, quality assurance, regulatory compliance, and licensing personnel.

b. Observations and Findings

1) Corrective Action Procedures

The inspectors reviewed procedures and document files for processes that included corrective actions for conditions adverse to quality. During this review, the inspectors noted that the licensee had no overall procedure or guidance describing what processes were acceptable for tracking and documenting corrective actions for conditions adverse to quality. The procedure for the primary corrective action program, Compliance Procedure (CP)-111, Processing of Precursor Cards for Corrective Action Program, Rev. 58, dated August 12, 1997, described a program that accepted completed PC forms addressing any known or suspected conditions, then graded the PCs A, B, C, or D based on safety significance. CP-111 listed four other programs to which grade D PCs could be closed: Work Requests (WRs), Modification Approval Records (MARs), Design Change Notices (DCNs), and Plant Equipment Equivalency Replacement Evaluations (PEEREs). The inspectors found that a licensee Quality Assurance (QA) Surveillance was underway to verify that WRs, MARs, DCNs, and PEEREs satisfied the regulatory requirements for corrective action processes.

The inspectors noted that the CP-111 list (i.e., WRs, MARs, DCNs, and PEEREs) did not include other programs that were routinely used for tracking and documenting corrective actions for conditions adverse to quality, such as the Nuclear Operations Tracking and Expediting System (NOTES), the Restart Management System for identified restart items, and training conducted by the training department. Further, some managers stated that they felt that the PC computerized system was too cumbersome, and these managers preferred to use the NOTES system for tracking important items. Inspectors found that 45 NOTES items were being used by 17 different managers or supervisors to track corrective actions for PCs. Also, violation responses, Licensee Event Reports (LERs), and restart corrective action items were, by procedure, tracked by the NOTES system. However, completed files for restart items, violation responses, and LERs were not required to be sent to Records Management. The procedure for LERs required feedback of LER corrective actions into the PC system; however, procedures for violation responses and restart items did not require that all corrective actions be fed back into the PC system. The inspectors reviewed the files for restart items and violation responses and determined that they contained documentation of corrective actions for some conditions adverse to quality that would not be duplicated in other programs that did send completed documents to records management. Examples of such corrective actions included counselling of operations and engineering individuals, training of all operations and engineering personnel, benchmarking of FSAR updating procedures against other utilities, and periodic NSAT monitoring of the qualifications of personnel performing root cause evaluations.

The inspectors noted that CP-111 required that corrective actions for grade A, B, and C PCs be documented and sent to Records Management but did not require that corrective actions for grade D PCs be documented in quality records. Grade D PCs were to include less important conditions adverse to quality and also other issues that were not conditions adverse to quality. Inspectors also noted that prior to Rev. 57 of CP-111, dated June 9, 1997, tracking or documentation of corrective actions for grade C or D PCs had not been required. The lack of resolution documents for grade C PCs had been identified in Quality Programs Surveillance (QPS)-97-0015, dated February 7, 1997, and documented in PC 97-1032, dated March 11, 1997.

Inspectors identified that four corrective action procedures were inadequate, in that they did not require that all corrective actions for conditions adverse to quality be documented in quality records. The procedures did not require that completed documents be sent to Records Management for microfilming and placement into secure and fire resistant storage. The four procedures were:

- CP-111, Precursor Card Program, Rev. 58, dated August 12, 1997, did not require records of corrective actions for level D precursor cards be sent to Records Management.

- CP-214, Regulating Correspondence Process and Validation, Rev. 0, dated June 9, 1997.

- Nuclear Operations Department (NOD)-10, Processing of Nuclear Operations Term Commitments (NOTES System), Rev. 6, dated January 31, 1996, which both described processes for documenting corrective actions to NRC violations, did not require that documentation of these corrective actions be sent to Records Management.

- NOD-57, Restart Management, Rev. 1, dated May 1, 1997, did not require that completed restart item packages go to Records Management.

In response to this issue, the licensee initiated PC 97-5995 and expressed plans to revise the procedures. The inspectors identified no corrective actions that the licensee had failed to pursue or complete and assessed that these procedural deficiencies had minor safety significance. The inspectors identified this issue as Non-Cited Violation NCV 50-302/97-11-03, Corrective Action Procedures Failed to Require Quality Records.

The inspectors noted that the licensee's corrective action procedures and practices included some inefficiencies. Many corrective actions were tracked and recorded in multiple programs; for example, in a PC and also in a violation, LER, or restart item. Grade D PCs included conditions adverse to quality (that would require quality records) and included many issues or questions that were not conditions adverse to quality (that would not require quality records).

2) Precursor Card Corrective Actions

The current backlog of open PCs numbered approximately 2200, of which 125 were graded as significant conditions adverse to quality (grade A or B). There were approximately 250 open Problem Reports (PRs) remaining from the previous corrective action program, which was in effect prior to mid-November, 1996. Approximately 1300 of the open PCs were identified during the System Readiness Review Program (SRR), which is discussed in section E7.2 of this report. The initial, problem identification phase of the SRR was recently completed and the licensee was refocusing resources on the analysis and resolution of the identified findings. The inspectors noted that ten PCs, of the twenty-two 1997 level B PCs closed, were closed by transfer of the issues to PCs which were still open. This indicated that the number of closed items was not a clear indicator of the licensee's progress in resolving issues.

a) Grade A and B PCs

The inspectors reviewed a sample of PCs which were graded A or B. Grade B PCs 97-2942 and 97-1530 identified reduced Emergency Diesel Generator (EDG) capacity due to the degraded cooling air flow conditions of the EDG radiator. This licensee identified problem was discovered during the analysis to support EDG capacity upgrade modifications. The technical evaluations to support operability and reportability were adequate. An LER was initiated to report past inoperability of the EDGs. The Suspected Design Base Issue (SDBI) evaluation appropriately identified this as a design base issue. EDG capacity was adequate for present Mode 5 requirements and a modification was in progress to correct the radiator cooling problem. A root cause analysis and associated corrective actions had not yet been determined. This PC remained open and the processing of this issue in the corrective action program was adequate.

Grade B PC 97-5696 addressed the licensee's inadequate implementation of the instrument calibration program which resulted in numerous in-plant instruments exceeding their calibration grace period. Related grade B PCs which were closed and transferred to this PC for resolution included PCs 97-0985, 97-0986, 97-0987 and 97-1060. An NRC violation (VIO 50-302/97-01-04) was previously identified for this condition. Although the PC was not completed due to outstanding corrective actions, the scheduled actions adequately addressed the determined root causes and identified examples. Appropriate responsibility was assigned for verifying the implementation of corrective actions. The processing of this issue in the corrective action program was adequate.

Grade B PC 97-2633 involved a configuration control problem with not removing a red tag from a previous clearance on the air for valve MUV-253. This PC was closed and later reopened when the licensee noted an additional seven PCs dealing with configuration management problems. This trend caused the licensee to perform a root cause analysis. This root cause identified 46 incidents of configuration management control problems. The inspectors reviewed the root cause analysis and noted there were several standdowns to discuss configuration management from an operational and engineering prospective. The tagging Procedure CP115, Nuclear Plant Tags and Tagging Orders, changed a check list to include a column for tags returned (removed) and the licensee had a training secession on the procedure. This issue appeared to be adequately resolved, however, the licensee will need to monitor the area.

Grade B PC 97-2754, dated April 17, 1997, addressed the fact that previous PRs and PCs had identified failures to provide timely corrective action to identified deficiencies in the engineering area, but there had not been improvement in this condition. Further, the PC stated that results of a QA review indicated a significantly increasing trend in overdue corrective action steps. This PC was assigned a grade of B and closed to previous PC 97-3159. Grade B PC 97-3159, which was still open, included a Root Cause Report dated June 13, 1997. The Root Cause Report concluded that untimely resolution of corrective action assignments has been a station-wide and ongoing problem. It included recommended corrective actions. One of the corrective actions, which was completed, was a statement of expectations for meeting due dates that was signed by the members of the Corrective Action Review Board. The inspectors noted that the Root Cause Report reasonably identified a number of contributing factors and recommended actions to both correct the problem and to review other processes (i.e., WRs) to determine the extent of condition. The inspectors noted that this open PC indicated a licensee recognition of a significant problem with the corrective action process.

In general, the documentation of information in the PC packages was inconsistent and required further review with the plant staff to verify the justifications in technical evaluations. No standard format was specified or noted for the documentation of corrective actions or the boundaries of the SDBI reportability evaluation within the PC process. For example, the SDBI evaluation in PC 97-1530 concluded there was a design basis issue and specified "no actions" under conclusion recommendations. The PC root cause evaluation had not been performed and no corrective actions were specified although a modification was in progress to resolve the issue.

b) Grade C and D PCs

The inspectors estimated that approximately 4000 grade C and D PCs had been closed in late 1996 or 1997 when the revision of CP-111 in effect did not require documentation of corrective action completion. With a brief review of those items, inspectors identified several PCs that described conditions adverse to quality and that would not have been tracked as a MAR, WR, DCN, or PEERE. One example was PC 97-2337, dated March 10, 1997, which stated that an additional step was needed in the relatching procedure for the turbine-driven EFW pump overspeed trip. The PC stated inappropriately that this was not an operability issue because the valve is periodically verified during Surveillance Procedure (SP)-349B to be latched at times the system is required to be operable. The PC was assigned a grade of D and closed based on assigning one individual to correct OP-450, another individual to correct Performance Testing Procedure (PT)-350, and a third individual to correct the plaque near the overspeed trip.

Another example, PC 97-3246, dated May 13, 1997, stated that several instances of valve "preconditioning" have been discovered during an Inservice Testing (IST) program review in response to NRC information Notice 97-16, dated April 4, 1997. This preconditioning involved stroking the valve several times before measuring the stroke time for the surveillance procedure. The PC stated that addressing "preconditioning" issues is not an American Society of Mechanical Engineers (ASME) Section XI requirement. It further stated that addressing this issue is a proactive initiative to keep up with industry trends before regulatory requirements force compliance. The PC stated that an extent of condition review was completed, which identified eight valves that were being "preconditioned." Nuclear Engineering Programs and Nuclear Operations were to revise the related procedures. Additionally, reviews for preconditioning in areas outside ASME Section XI valve stroke time testing were to be performed by four different individuals and tracked by four different NOTES items. Based on this information, the PC was graded C and closed.

A third example, PC 97-0993, dated February 21, 1997, stated that a Structural Maintenance Rule walkdown had identified a number of missing or damaged cable tray covers. The PC was graded C and closed based on a statement that the electric shop was putting together a walkdown team to identify cable tray cover deficiencies in all plant locations, and then would designate a plan of action to make appropriate resolutions.

A fourth example, PC 96-5314, dated November 25, 1996, stated that previous reclassification of the boric acid pumps from ASME Code Class 3 (safety related) to ASME Code Class 4 (non safety related) and removal of them from the Inservice Inspection (ISI) program may not be consistent with the licensing basis or with Technical Specification 4.0.5. The PC was graded C and closed based on assignment of mechanical design engineering to track resolution of the issues.

To further review the licensee's completion and documentation of corrective actions for these closed PCs, the inspectors opened Inspector Follow-up Item IFI 50-302/97-11-04, Corrective Actions for Approximately 4000 Precursor Cards Not Tracked to Completion. In response to this issue, the licensee stated plans to conduct a QA audit of the corrective actions for these closed PCs and initiated PC 97-5994.

The inspectors also reviewed the following grade C or D PCs:

- Grade C PC 97-2100 identified a wrong part installed in a radiation monitor. This issue was adequately resolved and appropriate corrective actions were implemented to prevent recurrence.

- Grade D PC 96-5791 involved the need for a drawing change on a safety related system. The PC was opened and closed on the same day. There was no indication in the PC file that the drawing had been changed. However, the licensee did produce documentation that the change had taken place.

- Grade C PC 97-0124 involved an unusual tagging order sent to the site safety committee where the information given was incorrect. The corrective actions appeared appropriate and the apparent cause proper. However, there was no record of verbal counselling of the individual involved.

- Grade D PC 97-0242 involved the potential to affect adversely the Emergency Core Cooling System (ECCS) system following draining for maintenance due to the formation of a void in the system. This item was improperly graded D when it should have been higher. (Improper grading of PCs was the subject of Violation 50-302/97-07-01.) The item was appropriately identified as a restart issue.

- Grade C PC 97-2188 involved some dye penetrant indications that were found on a butt weld on a stainless steel pipe. The apparent cause and the disposition were adequate.

3) <u>Quality Assurance Audits and Surveillances</u>

Inspectors reviewed portions of six QA audits and eight QA surveillances conducted during 1997 that addressed areas related to corrective actions for conditions adverse to quality. The inspectors assessed that the QA audits and assessments of corrective actions overall were good. They looked at a broad scope of areas and had many findings. These findings were well described in the reports, well documented in Precursor Cards, and followed up in subsequent QA audits.

c. Conclusions

The inspectors concluded that the licensee's process for assuring the completion of corrective actions for grade A and B PCs was adequate. For Grade C and D PCs, previous procedures did not require tracking or documentation of corrective actions. The inspectors opened an Inspector Follow-up Item (IFI) for further NRC review of approximately 4000 level C and D PCs that were closed without tracking the completion of corrective actions. The licensee stated plans to conduct a QA audit of the corrective actions for these PCs.

Inspectors identified a non-cited violation for four corrective action procedures that did not require that all corrective actions for conditions adverse to quality be documented in quality records.

The inspectors identified a weakness in the licensee's corrective action program in that there was no overall procedure or guidance describing what processes were acceptable for tracking and documenting corrective actions for conditions adverse to quality.

The inspectors assessed that QA audits and assessments of corrective actions overall were good. They looked at a broad scope of areas and had many findings. These findings were well described in the reports, well documented in Precursor Cards, and followed up in subsequent QA audits.

The inspectors assessed the licensee's performance, relative to corrective action program effectiveness, 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

O8 Miscellaneous Operations Issues

O8.1 (Closed) VIO 50-302/95-16-03: Inadequate Procedures for Operation of the Makeup Pump 1A Cooling Water (FPC Restart Issue O-8A)

(Closed) LER 50-302/95-010-01: Inadequate Procedure Causes Low Cooling Water Flow to Makeup Pump Resulting in Operation Outside the Design Basis

a. Inspection Scope (92901)

The inspector reviewed the licensee's corrective actions including the evaluation of reduced cooling flow to Makeup Pump (MUP)-1A, determination of component maximum flow values, procedure modifications to prevent recurrence, and results of DC and SW system flow balances. Furthermore, the inspector interviewed engineers involved in the evaluations and independently verified implementation of selected corrective actions.

b. Observations and Findings

The licensee's evaluations were thorough. The inspector reviewed the licensee's evaluation for operability of MUP-1A which was documented by REA 950627. The REA was well written and concluded that MUP-1A would have been capable of performing its safety function. This conclusion was based on a maximum Ultimate Heat Sink (UHS) temperature of 92 degrees F which was supported by historical data. The evaluation further determined that operator action may have been necessary to restore cooling water flow to the pump within 20 minutes for the design basis UHS temperature of 95 degrees F.

The inspector initially questioned an assumption used in the evaluation for the motor cooler heat transfer performance with reduced cooling water flow. The licensee provided additional information which adequately supported the assumption during a telephone call on August 5, 1997.

The inspector reviewed PT Procedures PT-136A, SW System Flow Balance, Rev. 0, PT-136B, DC System Flow Balance and EGDG KW Loading, Rev. 0, and OP-408, Nuclear Services Cooling System, Rev. 84. The procedure changes specified as corrective actions were performed. The inspector reviewed the changes and determined they were adequate to prevent recurrence of the low cooling flow to MUP-1A. The inspector also verified that all SW and DC flows measured during the flow balances were within the required range. Based on the licensee's corrective actions, these items are closed.

c. Conclusions

The licensee's corrective actions were thorough and accurate. The evaluation to document historical operability of MUP-1A was well written.

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

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

08.2 (Closed) VIO 50-302/96-20-01: Failure to Adhere to Reactor Coolant System Cooldown Limits

a. Inspection Scope (92901)

The inspector reviewed the corrective actions developed in response to the Violation of February 5, 1997, in a letter dated March 7, 1997.

b. Observations and Findings

The inspector reviewed the licensee's corrective action, as delineated in the response of March 7, 1997. A fracture analysis was performed for the reactor pressure vessel by Framatome Technologies and reviewed and approved by the licensee on July 2, 1996. The inspector reviewed the analysis and concluded that it was acceptable, performed in accordance with the ASME Section XI requirements.

The inspector verified that the involved operators had been counseled, as documented in the corrective action documentation dated February 1, 1996. The lesson plan for plant cooldown, Reactor Operator Training (ROT) 4-54, last revised on May 14, 1997, included a summary of the event and detailed the corrective actions taken and the correct methodology for maintaining the plant cooldown within limits.

A Short Term Instruction (STI), 96-004, was issued on January 17, 1996 to provide guidance for correctly performing monitoring of cooldown rates. The STI instructed the operators to use the decay heat removal system cooler outlet temperature instead of T_{cold} when no reactor coolant pumps are in operation. SP-422, RCS Heatup and Cooldown Surveillance was revised on April 27, 1996, to provide the same guidance. At that time, the STI was cancelled.

In response to REA 93-677, which was issued to provide guidance on limits for wapping from one DH system train to the other train, specific guidance was provided to ensure brief temperature drops that occur when swapping DH trains are fully addressed. The inspector verified that this guidance was included in OP-404, Decay Heat Removal System, Revision 107, issued July 25, 1997.

A Technical Specification interpretation was developed and incorporated in OP-202, Plant Heatup, and OP-209, Plant Cooldown, regarding the use of average reactor coolant temperature for defining modes. The inspector reviewed both of the procedures and verified that the guidance was clearly delineated.

Procedure OP-404, Decay Heat Removal System, was revised February 23, 1996, to provide the operators instructions for maintaining and adjusting cooldown rates. The inspector reviewed the procedure and assured that this revision was issued.

c. Conclusions

The corrective actions taken in response to VIO 50-302/96-20-01 were sufficient and warrant closure of this item.

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

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

O8.3 (Closed) LER 50-302/93-002, Supplement 2 and Supplement 3: Switchyard Cable Failure Caused Degraded Voltage of Class 1E Electrical Busses and Actuation of Emergency Diesel Generators (92902)

The inspector reviewed the licensee's corrective actions detailed in Supplement 2, dated November 8, 1995, and Supplement 3, dated May 10, 1996. In addition to the corrective actions for the original LER and Supplement 1, which were addressed in IR 50-302/95-009, the licensee detailed additional modifications to be made to the switchyard, which necessitated coordinating outages between the nuclear plant and the adjoining fossil units. In July 1996, the modifications were completed to the switchyard, replacing breaker control and power cables for both the 230 kV and 500 kV switchyards. The inspectors reviewed the documentation of the completed modifications. This work was not performed using nuclear plant controls, but was accomplished by the licensee's Energy Controls Office. No outstanding corrective actions remain on this issue. This LER is closed.

O8.4 (Closed) IFI 50-302/96-03-15: HPI Flow Indicator 50.59 and Tech Spec Bases Change

(Closed) LER 50-302/96-07-01: HPI Line Break With Loss of Battery Could Result in Reliance on Inadequate Accident Mitigation Instrumentation

a. Inspection Scope (92901)

This Inspector Follow-up Item was part of the NRC restart list and was tracked by the licensee under their restart issue number R-14. The inspector reviewed the licensee's actions to address the original inspector's concern that a pending change to add new, low range HPI line flow indicators received an adequate safety evaluation and the Technical Specification bases were changed as planned. The inspector also reviewed the related LER 96-07 and verified the original problem had been corrected.

b. Observations and Findings

The inspector observed that the licensee reevaluated the 1996 original safety evaluation as part of their Restart Item R-14 review. They determined the original evaluation was inadequate because it did not fully address electromagnetic and radio frequency interference (EMI/RFI) potential associated with the upgrade of the flow indicators from analog to digital indicators. Although the original design change did test the new indicators for EMI/RFI problems, the licensee determined it was qualitative testing and inadequate to support the original safety evaluation conclusion of no new failure possibilities. The licensee verified that subsequent third party testing resolved these specific concerns for the instruments and was documented in April of 1997. The licensee updated the safety analysis and concluded the modification did not create an unreviewed safety question. The inspector did not identify any concerns with the licensee's resolution. The licensee and inspector verified that the Technical Specification Bases were updated as required. The inspector also verified by surveys that operators were familiar with the bases change and the correct use of the Technical Specification for an inoperable low range instrument.

The inspector noted one discrepancy when verifying the indicators in the field versus the Design Input Record (DIR) for MAR 96-02-09-01 which installed the new flow meters. The DIR was not updated when the physical location of the new instruments on the control board was changed during the MAR installation for train separation concerns. Consequently the DIR didn't match the configuration in plant but the MAR installation instructions, plant drawings, and instrument labels were correct so a loss of the desired configuration control did not occur. The licensee initiated PC 97-5597 to document and correct the discrepancy via a Field Change Notice to the MAR. The inspector considered this appropriate.

The original concern of the LER was the inadequacy of the installed instrumentation and that was corrected by the addition of the narrow range channels. The inspector verified the change was reflected in the FSAR and that operators were familiar with the change and the reason for it. Subsequent to the issuance of the LER, numerous actions have been taken by the licensee during the current shutdown to correct engineering design processes. These have been previously inspected and the inspector determined they also adequately address the root cause of this LER.

c. Conclusions

The inspector determined the licensee's completed restart item fully addressed the original concerns in both open items. Consequently, IFI 96-03-15 was closed and LER 96-07 was closed, which included both Revision 00 and 01. The inspector concluded the licensee's safety analysis group exhibited a good level of skepticism when evaluating the original safety evaluation. The inspector concluded the inaccurate DIR was another example of the already reported weaknesses in the licensee's old design control process.

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

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

II. Maintenance

M1 Conduct of Maintenance

M1.1 Replacement of ASV-15

a. Inspection Scope (62707, 92902)

The inspector reviewed work being performed on Auxiliary Steam Valve (ASV)-15, problems encountered while performing the work, and the licensee's assessment and initial corrective actions to address the problem.

b. Observations and Findings

On July 9, 1997, two maintenance technicians were assigned WR 344344 to cut ASV-15 out of the auxiliary steam system and weld in a replacement valve. During the pre-job briefing, the maintenance supervisor informed the technicians that the clearance had not been obtained but setup for the task could continue. During setup for the work, the technicians identified that the replacement valve was not a direct replacement. The WR did not include instructions for installing a valve that was not a direct replacement. The maintenance shop returned the WR to the planning department for rework. The maintenance supervisor met with the Chief Nuclear Operator (CNO) assigned to perform clearances. The CNO informed the maintenance supervisor that the clearance could not be hung as requested, as only a single isolation had been requested. Procedure CP-115, Nuclear Plant Tags and Tagging Orders, stated that a double isolation was required in the circumstances that existed on the system, unless the Nuclear Shift Manager (NSM) approved a single isolation. Neither the maintenance supervisor nor the CNO pursued the matter with the NSM.

The task was carried over to the back shift maintenance schedule; however, it was not worked due to a lack of manpower. The next morning, the task was again assigned to the day shift to complete. On July 10, 1997, one of the two scheduled master mechanics was absent from work. Since the master mechanics perform the pre-job briefings for routine assignments, the remaining master mechanic was performing all briefings on that morning. The master mechanic provided the technicians with the corrected work package and provided a short pre-job briefing, without conducting a thorough work package review. After the technicians left the shop area, the master mechanic reviewed the work package and realized that no clearance was included. By the time he reached the field, the technicians had already removed ASV-15, without signing onto a clearance, as required by the work request.

The inspector reviewed the licensee's root cause analysis and made comparisons to the conclusions reached during the NRC inspection of the event. The licensee identified four inappropriate actions and three contributing factors. The licensee concluded that the master mechanic failed to verify a clearance had been obtained prior to releasing the workers to remove the valve, the two workers failed to self-check or verify the work request was ready prior to physical removal of the valve from the system, and the pre-job briefing was not adequate. Contributing factors identified included work control center operators not resolving questions as to the proper way to tag the valve, the workers felt that they were working under time pressure due to the delays in beginning the task, and the original work package was not properly planned. The inspector concluded similar root causes, but noted that the licensee's original root cause determination did not address procedural aspects of the event.

Licensee Procedure CP-113A, Work Request Initiation and Work Package Control, Revision 21, Step 3.2.3, Responsibilities, stated that the work supervisors were responsible for ensuring the identified activities were completed per applicable procedures, approved work instructions and directives and to resolve work problems that did not require Work Request re-evaluation. Step 4.3.2.1 required that the work supervisor or designee ensure that a safety and/or pre-job briefing was performed with the work group and review the work package and signify, by signing and dating Part 3, that the activity could be performed as instructed. Step 4.3.2.4 requires that the person(s) performing the activity perform the activity in accordance with applicable procedures, approved work instructions, and/or management directives. Step 4.3.2.4 also requires that the person(s) performing the work enter the tag order number or clearance number on the Work Request. However, the procedure allowed the entry to be made at any time during work performance or during work package close out. The original root cause analysis did not address procedural requirements or weaknesses in CP-113A. The licensee revised the original root cause analysis to address these issues.

The licensee corrective actions for this event included developing a sitewide pre-job briefing administrative procedure to improve consistency and content, meeting with all maintenance personnel to clarify expectations and requirements for performing work and to discuss the event, scheduling a quality performance surveillance in October 1997 to evaluate performance, and recording NUPOST comments for CP-113A for the next revision, requiring that the clearance and RWP numbers be recorded on the WR prior to the start of work.

Technical Specification 5.6.1.1, Procedures, requires that written procedures be established and implemented for activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, including general procedures for the control of maintenance, repair, replacement, and modification work. The failure to adhere to CP-113A by reviewing the work package and assuring that a clearance was obtained prior to the beginning of the removal of ASV-15 is a violation. This licensee identified and corrected violation is being treated as a Non-Cited Violation, NCV 50-302/97-11-05, Removal of ASV-15 Without Reviewing and Complying with Work Instructions.

c. Conclusions

A number of weak work practices were exhibited during the performance of this task. The work supervisor and the technicians failed to perform an adequate review of the work package prior to beginning work. Neither maintenance nor operations personnel initiated a resolution to the concerns with the method of tagging of the component, the pre-job briefing was weak, and procedural controls did not prevent this type of event from occurring. In addition, the licensee initially failed to address procedural aspects of the event in the root cause analysis. Management response to this event was timely and proactive, including provisions for assessment of the results of the corrective actions in the future.

M1.2 Building Spray Pump 1B Post Maintenance Functional Test

a. Inspection Scope (61726, 62707)

The inspector reviewed work being performed on Building Spray Pump (BSP)-1B, problems encountered while performing the work, and the licensee's assessment and initial corrective actions to address the problem.

b. Observations and Findings

On July 13, 1997, BSP-1B was being run to satisfy the post-maintenance testing requirements of WR 338614, which was used to install the modified building spray pump impeller. During the test, the line bearing temperature reached 180°F, the upper limit allowed by the testing procedure. The test was terminated at approximately 3:20 p.m. as a result of the high bearing temperature. The maintenance personnel notified engineering and maintenance management of the problem. The decision was made to take an oil sample, per management direction, using licensee procedure, Preventive Maintenance (PM)-133, Equipment Lubrication and General Inspection, Revision 50. At that time, the licensee made a decision not to enter licensee Maintenance Procedure (MP)-531, Troubleshooting Plant Equipment, Revision 9.

Attempts to obtain a good oil sample through the upper vent port were unsuccessful, so the decision was made to drain the oil from the pump, per PM-133. A short pre-job briefing was held and the technicians were dispatched to obtain an oil sample. At the briefing, the supervisor did not address how to drain the oil. He considered this to be basic journeyman knowledge. At the pump, the technicians identified what they assumed was the lube oil drain plug and removed it. This plug was the drain line for the decay heat closed cycle cooling system (DC), which provides cooling for the pump lubricating oil. The maintenance supervisor notified the main control room and obtained permission to close the DC valves to the pump, DCV-117 and DCV-118. Closing these valves secured the leak and allowed the technicians to reinstall the plug.

The licensee initiated a root cause analysis, which was completed on August 1, 1997. The root cause analysis identified one inappropriate action and three contributing factors. The licensee identified that this was not the first time that the DC jacket cooling water plug was removed by mistake, approximately ten years earlier. Even though no documentation of an earlier event could be located, several members of the maintenance department remembered the earlier events. It was identified that the decay heat removal pumps were constructed with the lower jacket cooling water system. DHP-1B oil was flushed and drained twice in September 1996. During the first evolution pre-job briefing (WR 336742) it was identified that the drain plug was the DC drain and not the oil drain. A new WR (WR 338017) was issued to address this issue. This issue was addressed in the new WR, but was not added to any lessons learned system.

Contributing factors identified were an inadequate pre-job briefing, inattention to detail while reviewing component drawings, which showed that the lower plug was not the oil drain, and time pressure, since the job started at the end of a shift, necessitating the technicians to work overtime to complete the task.

c. Conclusions

The inspector reviewed the licensee's root cause analysis and corrective actions and found that they addressed the immediate problem. Troubleshooting was continuing, using MP-531, to identify and address the problem with the high bearing temperature. The resolution to this issue will be addressed as part of the restart issue for the building spray pumps.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) VIO 50-302/96-20-02: Failure to Follow Procedure AI-400C for Review and Development of Maintenance Procedure PM-191

a. Inspection Scope (92902)

The inspector reviewed the corrective actions developed in response to the Violation of February 5, 1997, in a letter dated March 7, 1997.

b. Observations and Findings

The inspector reviewed the revision of AI-400C, New Procedures and Procedure Change Processes, Revision 19, to assure that guidance was provided on obtaining appropriate technical reviews when more than one end-user department is required for implementation of the procedure. The inspector reviewed the maintenance study book entry, dated March 31, 1997, which discussed the importance of multi-discipline reviews for procedure development and revisions.

The inspector verified that AI-100, Facility Administrative Policies, Revision 20, was issued on June 23, 1997, to address requirements for single point accountability for complex tasks or multi-disciplined evolutions. The inspector reviewed the procedure Revision and determined that it fulfilled the requirements discussed in the violation response.

c. Conclusions

The licensee adequately addressed the concerns addressed in the Violation. This issue is closed.

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

III. Engineering

E1 Conduct of Engineering

E1.1 General Comments (37551)

During June and July the licensee conducted an extensive 100% baseline inspection of both OTSGs. This included 100% bobbin coil inspection and numerous other required and licensee initiated specific inspections such as 100% upper roll transition inspection. The scope of this effort expanded significantly due to a loose part found on the A OTSG upper tubesheet. The part was determined to be half of a 3/4 inch hex nut and had done significant impact damage to the tube ends. This caused the licensee to have to repair over 10,500 of the 15,531 tube ends on the A OTSG in order to complete the eddy current inspections. The licensee also repaired over 3,000 known damaged tube ends in the B OTSG that were left over from previous outage work and caused by a previous loose part problem. The licensee's root cause investigation was still ongoing for the A OTSG loose part at the end of this inspection period. The inspectors will review the results of this effort to verify the adequacy of the licensee's resolution as to where the hex nut originated and where the other half could be.

The results of the licensee's inspection and repair efforts were very favorable. 77 tubes in the A OTSG were plugged during this effort for a total of 151 plugged out of 15,531 tubes. 483 tubes were plugged in the B OTSG for a total of 634 plugged. The majority of plugged tubes in the B OTSG were due to first span intergranular attack (IGA) volumetric indications due to a known historic problem during initial operation.

Conclusions

The licensee established a full baseline data base of all known OTSG conditions, and verified that their OTSGs were in very good condition for future operation.

E1.2 NPSH Concern with ECCS Pumps

a. Inspection Scope (40500)

The inspectors reviewed the licensee's actions to resolve a concern regarding net positive suction head (NPSH) for the ECCS pumps when the spent fuel pool (SFP) was running in recirculation to the borated water storage tank (BWST).

b. Observations and Findings

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

c. Conclusions

The inspector concluded that licensee personnel were making progress in their efforts to resolve this issue prior to restart.

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

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

E1.3 Design Control Process

a. Inspection Scope (37550, 37551, 92903)

The inspectors reviewed selected MAR packages, for modifications that were being installed during the current outage, to: (1) determine the adequacy of the safety evaluation screening and the 10 CFR 50.59 safety evaluations; (2) verify that the modifications were reviewed and approved in accordance with Improved Technical Specifications (ITS) and applicable administrative controls; (3) verify the modifications were being installed as required by the licensee's procedures and had proper sign-offs; (4) verify that the FSAR Enhanced Design Basis Document (EDBD), drawings, and applicable procedures were being updated, and (5) verify that post modification testing requirements were adequately specified. In addition, field walkdown inspections were conducted to examine selected portions of the installations.

b. Observations and Findings

The MAR packages inspected all had the engineering design and field work portions completed. Most of the field installation and some of the post modification testing had also been completed. Some MARs were field completed and returned to service but not closed out. None of the MAR packages reviewed were fully completed and closed out. The following MARs were inspected:

1) MAR 96-10-05-01, EGDG Power Upgrade

The purpose of this MAR was to upgrade the turbocharger and intercooler for both emergency diesel generators (EGDGs) A and B. Each EGDG had a new nozzle ring installed in the turbine portion of the turbocharger and a new dual pass intercooler installed. Both EGDG A and B were adequately tested and returned to service.

2) MAR 96-07-15-01, EGDG Standby Keepwarm Systems (DL & DJ) Setpoint Changes

The purpose of this MAR was to increase the minimum temperature lube oil setpoint from 110 to 115 degrees. The increased setpoint provided an alarm and standby pump interlock that would prevent damage to the standby pump from overly viscous oil. Operations would be alerted before the 110 degree minimum temperature was reached and the pump damaged. The temperature set point for the low alarm lube oil water jacket cooling was increased from 115 to 130 degrees to ensure the water temperature was maintained above

120 degrees as recommended by the EGDG vendor. The inspector only reviewed the design package for this MAR.

3) MAR 97-04-06-01, ASV-204 Spring Pack Replacement

The purpose of this MAR was to replace the spring pack in the motor operated valve (MOV) operators with a different size. The new spring pack was sized to allow the torque switch to be set within the required thrust limits of the valve. The Movats functional test was completed. However, the valve stroke timing test had not yet been completed.

4) MAR 96-10-02-01, EFW Cavitating Venturis

The purpose of this modification was to install flow restricting devices on the discharge side of both EFW pumps. The modification should eliminate the potential for pump runout or inadequate net positive suction head while still permitting the minimum design flow requirement of 550 gpm into an OTSG pressurized to 1050 psig.

The inspectors noted that the 10 CFR 50.59 safety evaluation was limited to installation and testing with the plant shut down and EFW not required to be operable. During testing, the flow control valve for the turbine-driven EFW pump had oscillated excessively and caused the pump to trip. Prior to plant restart, the licensee needs to correct that deficiency and complete a 10 CFR 50.59 safety evaluation for all plant operating modes.

5) MAR 96-10-04-01, Reactor Building (RB) Penetration Expansion Chambers

This modification added expansion chambers to various containment penetration piping that was susceptible to overpressurization due to thermal expansion of fluid, as described in Generic Letter (GL) 96-06. Each expansion chamber installation included a rupture disk and tubing from the expansion chamber to the affected section of piping.

6) MAR 96-10-10-03, Emergency Feedwater Valve (EFV)-12 Installation

This modification replaced the EFW cross-tie isolation valve (a manual gate valve) with a motor operated parallel disc gate valve. This will allow operators to operate the valve from the control room in the event of a small break loss of coolant accident concurrent with a loss of B battery failure, so that the turbine-driven EFW pump can discharge through the A train EFW flow control valves to the OTSGs.

7) MAR 96-11-01-01, ASV-204 EFIC Automatic Opening Reinstallation

This modification reinstalled the automatic opening of ASV-204, the alternate steam admission valve to the turbine-driven EFW pump. This provided for automatic starting of the turbine-driven EFW pump in the event of a loss of power or other failure to ASV-5, the primary steam admission valve.

The inspectors also reviewed selected AIs, CPs, and Nuclear Engineering Procedures (NEP). These instructions and procedures provided the requirements for the licensee's design control process. The inspectors reviewed the instructions and procedures to verify that they had been updated to address previously identified weaknesses in the design control process. The instructions and procedures reviewed included the following:

- AI-602, MAR Work Package Preparation, Implementation, And Closure, Revision 13, dated June 30, 1997
- CP-213, Preparation of a Safety Assessment and Unreviewed Safety Question Determination (10 CFR 50.59 Safety Evaluation), Revision 3, dated July 3, 1997
- NEP-104, Interface Design Control, Revision 8, dated June 30, 1997
- NEP-210, Modification Approval Records, Revision 17, dated June 30, 1997
- NEP-212, Processing of Modifications Projects by Nuclear Projects, Revision 18, dated June 30, 1997
- NEP-213, Design Analysis/Calculations, Revision 10, dated March 31, 1997
- NEP-254, Plant Equipment Equivalency Replacement Evaluation, Revision 13, dated March 31, 1997
- NEP-261, Design Verification, Revision 5, dated March 31, 1997
- NEP-271, Modification Approval Records, Commercial Grade Work Requests, And Plant Equipment Equivalency Replacements, Revision 13, dated March 31, 1997

The inspectors found the MAR packages to be complete, including all required reviews and signatures. Also, the inspectors identified no discrepancies between the MAR packages and the FSAR, EDBD, or ITS. The 10 CFR 50.59 safety evaluations were thorough and technically adequate. Field installations inspected were in accordance with the requirements of the applicable MAR packages. Changes to NEPs to address previously identified weaknesses included, but were not limited to, the incorporation of Procedure CP-213 requirements, additional guidance regarding design inputs, and guidance regarding prompt revision to design basis documents following implementation of a plant modification. Current procedures generally provided adequate controls for implementation of the licensee's design control process. The inspectors noted that additional changes had been made to some of the NEPs subsequent to the design development and implementation of some of the MAR packages. These additional changes provided further enhancement to the design control process.

The inspectors noted that information contained in the working MAR packages reviewed was generally not organized very well, which caused some difficulty during the review. This observation was also expressed by some of the plant personnel who indicated that the process was somewhat cumbersome.

c. Conclusions

The inspectors concluded that the MAR packages reviewed were technically adequate and were being implemented in accordance with licensee requirements and NRC regulations. Current procedures generally provided adequate controls for implementation of the licensee's design control process. Additional changes made to some of the NEPs subsequent to the design development and implementation of some of the MAR packages provided further enhancements to the design control process.

The inspector assessed the licensee's performance, relative to the design control process, 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

E3 Engineering Procedures and Documentation

E3.1 Review of Plant Equipment Equivalency Replacement Evaluation Process

a. Inspection Scope (37551)

The inspector reviewed the process used by the licensee to evaluate replacement equipment, components, or parts with items having different characteristics. The process and a number of completed evaluations were assessed to determine that procedural guidance was clearly and appropriately defined and that field implementation was in accordance with the defined program.

b. Observations and Findings

Licensee Procedure, NEP-254, Plant Equipment Equivalency Replacement Evaluation, Revision 13, was reviewed by the inspector. The procedure addressed two types of parts to be used: equipment replacement and equipment equivalency replacement. Equipment replacement was defined as replacing an item with another item which is identical in all respects. Manufacturer part number changes that were administrative only, with no change to the part itself, were considered equipment replacements. The procedure required that in cases where equipment was purchased directly from a manufacturer with a 10 CFR 50, Appendix B design control program, manufacturer-initiated replacements may be provided, for which the manufacturer had performed an evaluation under the manufacturer's approved Appendix B program, provided there was no change in form, fit, or function. This was also considered to be an equipment replacement and did not require an evaluation.

Step V.A.2 defined the second type of equipment replacement, equipment equivalency replacement, as replacing an item with another item which was different from the licensee specified requirements, but which has been determined, through engineering evaluation, to be equivalent in that the replacement item fulfills the licensee required critical design characteristics and is equivalent in form, fit, function, and structural integrity. A note attached to that step stated that in the case where a critical design characteristic was changed or where the replacement resulted in operational, functional, or performance changes, the replacement, or that portion of the replacement which represented the change would require a modification or commercial grade work request. The licensee defined critical design characteristics as those properties or attributes established by the licensee which are essential to the physical and functional interfaces, qualification, and capability of equipment, components, and parts to perform their intended function. To use an equipment equivalency replacement, the licensee performed a formal review to determine the acceptability of the replacement.

A review was conducted of a sample of completed PEERE evaluations. The majority were conducted to allow use of a different material or component than was originally used in an application. PEERE 1497 was issued on April 10, 1997, to document modifications made to the building spray pump impellers. These modifications were made to increase the NPSH margin for the pumps. Changes to the impellers included increasing impeller eye diameter to match casing approach, cut back the leading edges of the inlet vanes to open flow passages, smooth and sharpen the inlet vanes, and polish the impeller eye area. The PEERE stated that the form of the impeller was modified slightly to improve $NPSH_r$ characteristics. The form change was the result of polishing and shaving portions of the suction side of the impeller. The PEERE also stated that the function would not change as total head, efficiency, and brake horsepower did not change as a result of the changes. The PEERE concluded that although NPSH changed, it was not a function of the pump but was a characteristic. Licensee Specification CS 3-30-2, Technical Specification for Centrifugal Pumps for Auxiliary System Service, was the original purchase specification for the building spray pumps. One of the licensee-specified critical characteristics for these pumps was identified as NPSH available. NEP-254 step V.2, Note 1, required that a replacement or part of a replacement that represented a change to FPC specified critical design characteristic required a MAR/Commercial Grade Work Request (CGWR) to implement.

The PEERE process, as defined in Licensee Procedure NEP-254, did not require the performance of a 10 CFR 50.59 evaluation, since any changes under this process were considered to be analogous to a non-design change. The licensee performed a 10 CFR 50.59 screen on April 10, 1997, due to the realization on the part of the design engineering personnel that this change would require a revision to FSAR Table 6-12 , Post-Accident NPSH Requirements, and FSAR Figure 6-11, Reactor Building Spray Pump Characteristic. The screen required that the licensee perform a Unreviewed Safety Question (USQ) determination, based on the required changes to the FSAR. This USQ determination (USQD) addressed only the plant in mode 5 and did not address any other modes of operation. A note was included in restoration requirements that an additional USQD would need to be performed prior to entering Mode 4. As a result of performing this modification outside of the normal MAR process, the only reviews that the USQD received were the design engineer, a design engineering supervisor, and a member of the safety analysis group. Neither the Plant Review Committee (PRC) nor any licensee management above the engineering supervisor were involved in the review and approval of this modification.

On April 24, 1997, the licensee issued PC 97-2034, to document as-found testing performed by the manufacturer on the pump impellers. At that time, it was identified that the $NPSH_r$ for the impellers did not meet the original specification. An Engineering evaluation performed by the licensee concluded that operability of the pumps was not affected by this finding, as the NPSH calculations contained overconservative assumptions, which accounted for the non-conservative as-found values obtained by the manufacturer. A Quality Performance Surveillance, (QPS)-97-0117, was completed on August 8, 1997. Based on the assumption that the changes were restoring the BSP function to the original design, the conclusion was reached that the PEERE process was the appropriate forum for the change. The PEERE evaluation included no statement that the BSP was being restored to original specification. The PEERE evaluation stated that the modification of the impellers was to gain additional margin over that originally calculated, to decrease the chance for pump degradation due to cavitation.

On August 4, 1997, the licensee issued PC 97-5688 to document a review performed by licensee management, which concluded that the changes to the BSP impellers were inappropriately completed under the PEERE process. Technical Specification 5.6.1.1, Procedures, requires that written procedures be established and implemented for activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, including general procedures for the control of replacement and modification work. The failure of the licensee to adhere to the requirements of Licensee Procedure NEP-254 by completing this modification as a MAR/CGWR is a violation. This is identified as Violation 50-302/97-11-06, Failure to Follow Licensee Procedure NEP-254, resulting in a modification of a safety related system without appropriate reviews and approvals.

c. <u>Conclusions</u>

The majority of the PEERE evaluations reviewed by the inspector were appropriately implemented and completed. However, an example of a modification made to critical characteristics of a safety related system was found to have been inappropriately completed as a PEERE evaluation, resulting in a violation. Weaknesses in the program existed which allowed the process to be inappropriately implemented, as was the case with the building spray pump impellers. This modification was identified as requiring the increased review and approval associated with the MAR process.

E6 Engineering Organization and Administration

E6.1 Mike Rencheck was appointed Director, Engineering, reporting to Roy Anderson, Senior Vice President, Nuclear Operations. John Holden has a new position of Site Director, as discussed in paragraph 06.1.

E7 Quality Assurance in Engineering Activities

E7.1 Quality Assurance Audits and Surveillances

a. Inspection Scope (40500)

The inspectors reviewed selected audits performed by the Nuclear Quality Assessments (NQA) section of the Quality Programs Department.

b. Observations and Findings

The inspectors noted that the NQA section had performed six audits since January 1997. These audits were integrated audits of activities which included aspects of operations, maintenance, engineering, and plant support. The inspectors reviewed the engineering and design control process aspects for the following audit reports:

Audit 97-01 Audit 97-04
Audit 97-02 Audit 97-05
Audit 97-03 Audit 97-06

During review of these audits, the inspectors noted that the NQA section initiated a number of precursor cards for engineering and design control activity discrepancies. Some of the findings from the audit reports included the following.

- A strong questioning attitude was not always evident in engineering documentation. (Audit 97-01)
- Engineering did not readily recognize when use of the corrective action system was appropriate. (Audit 97-01)
- Incorrect data was used as MAR design input. (Audit 97-02)
- Inadequate programs and procedures existed for implementing Technical Specifications (TS) and NUREG 0737 requirements for reactor coolant system leakage outside containment. (Audit 97-03)

- A significant challenge for Engineering was assuring timely resolution of identified problems. Corrective actions to reduce Engineering overdue corrective action backlogs were not effective. (Audit 97-04)

- Not all Engineering personnel understood which documents were design basis documents. (Audit 97-04)

- Additional management attention was warranted to assure timely resolution of fire protection equipment deficiencies. (Audit 97-04)

- Engineering programs and activities were effective in implementing the requirements of ANSI N45.2.11. (Audit 97-04)

- Requirements of the Fire Protection Plan were not translated to nuclear engineering procedures. (Audit 97-05)

- The quality of the supporting documentation for Improved Technical Specification changes required improvement. (Audit 97-06)

The inspectors noted that Engineering was taking corrective actions to address the various findings identified during the NQA audits.

c. <u>Conclusions</u>

The inspectors concluded that the NQA section has been active and effective in identifying continued weaknesses and areas for improvement in the licensee's Engineering activities and design control process.

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

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

E7.2 <u>System Restart Readiness Reviews</u>

a. <u>Inspection Scope (40500)</u>

The licensee was performing System Restart Readiness Reviews to provide assurance that all systems that were important to safety were designed, installed, operated, and maintained in accordance with their licensing and design bases. These reviews were to satisfy the extent of condition concerns related to previous engineering design and 10 CFR 50.59 violations. In addition, the reviews were to satisfy one of the conditions of an NRC Confirmatory Action Letter prior to plant restart. The inspectors reviewed the System Restart Readiness Reviews for three systems to assess the scope, depth, and quality of documentation of these reviews. The NRC plans to conduct a more thorough review of this area in a later Safety System Functional Inspection (SSFI) inspection prior to restart.

b. Observations and Findings

Inspectors reviewed the System Restart Readiness Reviews for three systems: Core Flood Tanks, Control Room Emergency Ventilation, and Communications. Inspectors found that the review packages were well organized and clearly documented. Many potential problems were identified, and they were adequately evaluated and entered into the PC system for tracking of corrective actions. Restart items were clearly identified. The System Restart Readiness Reviews focused on a list of system attributes (i.e.; fans, filters, ducting, isolation dampers) and seemed to be generally thorough but not fully comprehensive. For example, during the inspection, a licensee employee initiated a PC for a discovery of a single failure vulnerability in the service water (SW) system. If one of two operating SW pumps failed during an event, while two reactor building fan coolers were operating, and if the intake temperature was higher than 81 degrees F, then the service water system could exceed its design temperature limit. The licensee evaluated this issue to be reportable and made a 10 CFR 50.72 telephone report to the NRC. This single failure vulnerability had been identified during an extent of condition review for a previous violation, and had not been identified by the completed Service Water System Restart Readiness Review.

Inspectors also attended a management licensee review panel meeting for the Control Room Emergency Ventilation System Restart Readiness Review. Attendees included representatives from engineering, operations, and licensing, and provided an adequate breadth of experience for a multi-disciplined review. Attendees had reviewed the Control Room Emergency Ventilation System Readiness Review package prior to the meeting and came prepared with questions and comments. Most of the comments were editorial in nature and a few were substantive, but no significant problems with the review package were identified.

c. Conclusions

The inspectors concluded that the licensee's System Restart Readiness Reviews were well organized and clearly documented. Many potential problems were identified, and they were adequately evaluated and entered into the PC system for tracking of corrective actions. Restart items were clearly identified. The reviews appeared to be thorough but were not fully comprehensive. An NRC SSFI inspection is scheduled to provide a more detailed review of this area prior to restart.

The inspectors assessed the licensee's performance, relative to the System Restart Readiness Reviews, 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 Miscellaneous Engineering Issues

E8.1 (Open) URI 50-302/96-201-04: Nonsafety-Related Positioners on Safety-Related Valves

a. Inspection Scope (92903)

This Unresolved Item (URI) involved a concern identified by the NRC during the Integrated Performance Assessment Process (IPAP) inspection, where safety-related air operated valves (DCV-17, DCV-18, DCV-177, and DCV-178) used to control cooling water flow to the decay heat removal heat exchangers were connected to nonsafety-related positioners. The inspector initially followed up on the licensee's corrective actions for this item. This inspection effort was documented in IR 50-302/97-01.

b. Observations and Findings

Licensee corrective actions were documented in PR 96-0041 and PR 96-0220. Resolution of this issue was being tracked as licensee Restart Issues D-10 and R-7. The inspector reviewed the corrective actions that had been implemented to address this item. The inspector reviewed these corrective actions for compliance with the FSAR, TS, licensee topical design basis document (TDBD), and design control procedures.

The inspector noted that the licensee had implemented MAR 94-09-02-01, DC Cooling Instrument Enhancement, to address this issue. This modification addressed the NRC's concern regarding the nonsafety-related positioners on valves DCV-17, DCV-18, DCV-177, and DCV-178.

As discussed in the IPAP Inspection Report 50-302/96-201 (Appendix C, paragraph 3.1.5), the IPAP team questioned the design criteria in the Crystal River Unit 3 Topical Design Basis Document for the Single Failure Criteria, Revision 1, dated April 25, 1994. The inspector reviewed the TDBD and noted that the IPAP team questioned the applicability of the criteria included in the TDBD for single failure of nonsafety-related components. The TDBD stated that failures less than 1x 10E-6 should not be considered as credible. During this current inspection, the inspectors reviewed the licensee's documentation which provided the basis for the single failure criterion for nonsafety-related components contained in the TDBD. The inspectors held discussions with licensee personnel and raised questions regarding inconsistencies in the methodology used by the licensee in determining the failure frequency for nonsafety-related components. Licensee personnel indicated that they would review the Single Failure Criteria for Nonsafety-Related Components contained in the TDBD to determine if additional clarification was needed. The inspectors continued to question the methodology used by the licensee for determining the failure probability of a nonsafety-related component failure. This item remains open pending further review and discussions with the licensee regarding the design criteria for single failure of nonsafety-related components.

c. Conclusions

The inspector concluded that implementation of MAR 94-09-02-01 addressed the issue of nonsafety-related positioners on safety-related valves DCV-17, DCV-18, DCV-177, and DCV-178. However, the inspectors continued to question the methodology used by the licensee for determining the single failure probability of a nonsafety-related component. This item remains open pending further review and discussions with the licensee regarding the design criteria for single failure of nonsafety-related components.

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

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

E8.2 (Closed) IFI 50-302/96-201-12: Conduit Sizing Criteria - Jamming Ratio Not Considered

a. Inspection Scope (92903)

This IFI involved a concern that any cable installed under Electrical Design Criteria, page 5, paragraph IV.B, conduit sizing, did not consider the jamming ratio. Consequently, the licensee issued a PC to correct the appropriate design documents and investigate any cable installed after 1990 to determine if there was a jamming concern. The inspectors reviewed the licensee's corrective action for this IFI.

b. Observations and Findings

The inspectors verified that the licensee had issued PC 96-3488 and completed their investigation of cables installed after 1990. Four cables with three single conductors were installed, MUC-253, MUC-259, MUC-265, and MUC-271. The inspector verified by walkdown that the cables were very short (less than 20 feet) and had very few bends. Two cable had three 15 degree bends and two cables had two 15 degree bends. The calculated jam ratio was 2.91, which was within the licensee's design requirements. The inspector noted that the licensee was in the process of revising the engineering documents and Design Criteria to include the recommendations of IEEE Standard 690-1984, A9.2.4.4, Critical Jamming Ratio. The inspector did not identify any cable jamming concerns during this review. This IFI is closed.

c. Conclusions

The inspector concluded that the licensee had or was in the process of implementing appropriate corrective action to address this concern.

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

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

E8.3 (Open) VIO 50-302/96-09-05: Failure to Incorporate Design Information into Operations Procedures

a. Inspection Scope (92903)

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 to change 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 inspector followed up on the licensee's corrective actions by reviewing procedure changes, internal licensee correspondence, and interviewing engineering personnel.

b. Observations and Findings

The inspector noted that the corrective actions being implemented to address this violation (VIO) were being tracked under licensee Restart Issue OP-27. The inspector 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 inspector included licensee interoffice correspondence (IOC) NOE97-0228, dated March 14, 1997, which corrected the Design Input Record (DIR) for MAR 95-01-07-01 and enhancements to various Nuclear Engineering Procedures.

During further review of the licensee's response, the inspector noted that the 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 inspector 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 inspector 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 inspector that these questions required further review to determine if appropriate documentation has been submitted to NRR for review.

The inspector informed the licensee that this item will remain open, pending inspector verification that appropriate documentation has been submitted to NRR for review describing the changes and clarifications made to the licensing and design basis for Valve MUV-64.

c. Conclusions

The inspector concluded that the corrective actions for this Violation had been completed by the licensee but there were questions regarding whether the licensee had provided appropriate documentation for the changes and clarifications to the licensing and design basis for Valve MUV-64. This item remains open and will be reviewed further during subsequent NRC inspections.

The inspector 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 - Good
- Knowledge of the Design Basis - Adequate
- Compliance with Regulations - Adequate
- Operator Performance - N/A

E8.4 (Closed) URI 50-302/97-02-02: Deletion of Water Quality Requirements from the FSAR

a. Inspection Scope (92903)

This URI involved a concern regarding the deletion of FSAR Tables 4-10, 4-11, and 9-3 from the FSAR via FSAR Amendment No. 23. Deletion of these FSAR tables resulted in the removal of the RCS water quality requirements from the FSAR.

b. Observations and Findings

During review of this URI, the inspector noted that the RCS water chemistry requirements were removed from the TS with the issuance of TS Amendment No. 149, which implemented the CR-3 ITS. In support of License Amendment No. 149, the licensee proposed, and the NRC approved (via the NRC safety evaluation report issued for TS Amendment No. 149) relocating the provisions for reactor coolant water chemistry from (at that time) TS 3.4.7 to the FSAR and appropriate plant procedures.

The inspector noted that the licensee's removal of the RCS water chemistry requirements from the FSAR was not in accordance with 10 CFR 50.71(e), which requires licensees to update the FSAR periodically, to assure that the information included in the FSAR is the latest material developed. Regulation 10 CFR 50.71(e) further states that this submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee pursuant to Commission requirement, since the submission of the original FSAR or, as appropriate, the last updated FSAR. The updated FSAR shall be revised to include the effects of all safety evaluations performed by the licensee in support of requested license amendments. The inspector discussed this item with the Chemistry Department Manager who indicated that a FSAR change was being prepared (for inclusion in the next FSAR update) to incorporate FSAR Tables 4-10, 4-11, and 9-3 (and thereby the RCS water quality requirements) back into the FSAR. The inspector informed the licensee that removal of the RCS water quality requirements from the FSAR was a violation of 10 CFR 50.71(e). Therefore, URI 50-302/97-02-02 was closed and a violation was identified as VIO 50-302/97-11-07, Deletion of Water Quality Requirements from the FSAR.

c. Conclusions

The inspector concluded that the licensee's removal of the RCS water quality requirements from the FSAR was a violation of 10 CFR 50.71(e). A violation will be identified for this issue.

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

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

E8.5 (Closed) Violation 50-302/97-02-04: Failure to Conduct TS Logic Testing

a. Inspection Scope (92903)

This Violation involved a failure to conduct required TS surveillance testing on safety related circuits. On April 12, 1996, the licensee, in response to GL 96-01, Testing of Safety Related Logic Circuits, identified several circuits that were not tested in accordance with TS requirements. The inspectors reviewed the licensee's corrective action (complying with the recommendations in GL 96-01) for this violation.

b. Observations and Findings

The inspectors verified that the licensee had implemented a very effective program to comply with GL 96-01 for testing all TS safety-related circuits. During this inspection the inspectors verified that the licensee's contractor had completed all the required reviews for testing TS safety-related logic circuits as discussed in Section E8.8 of this report. Generic Letter 96-01 was also discussed in detail in NRC Inspection Report 50-302/97-02, Section E8.13. The requirement for the close out of GL 96-01 was that all corrective action shall be completed prior to startup from the present outage. This issue was being tracked as Restart Issue R-1. The inspector verified that the corrective action for the specific examples listed in the violation had been satisfactorily implemented.

c. Conclusions

The inspectors concluded that the licensee has implemented appropriate corrective action to resolve this violation. The licensee's compliance with the recommendations in GL 96-01 for reviewing the TS requirement for testing all safety-related logic was considered quite good.

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

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

E8.6 (Closed) URI 50-302/97-05-02: 50.59 Safety Evaluation does not Address Operation of the Atmospheric Dump Valves from the Remote Shutdown Panel During an Appendix R Fire Event

a. Inspection Scope (92903)

This URI involved a concern regarding the adequacy of the 10 CFR 50.59 safety evaluation performed for a change made to the ITS Bases 3.7.4. This ITS Bases change involved the atmospheric dump valves (ADVs) and the associated backup nitrogen supply for the ADVs. The ITS Bases change did not address Abnormal Procedure (AP)-990, Shutdown Outside Control Room, which took credit for using the ADV backup nitrogen supply to operate the ADVs from the remote shutdown panel during an Appendix R fire event. The inspector followed up on the licensee's actions by reviewing procedure changes, internal licensee correspondence, and interviewing engineering, operations, and licensing personnel.

b. Observations and Findings

During followup of this item, the inspector noted that the licensee had initiated PC 97-2360 (dated March 31, 1997) to document a concern that remote control of the ADVs might not be available for long term cooling during an Appendix R fire event as indicated in Revision 9 to Procedure AP-990. The backup nitrogen supply used for remote operation of the ADVs was only designed to meet the 4-hour station blackout requirement, whereas, Appendix R required that a loss of offsite power be assumed for 72 hours. Revision 8 to Procedure AP-990 eliminated the local manual operation of the ADVs and, instead, directed that the backup nitrogen supply be aligned to allow remote operation of the ADVs. Revision 8 was issued November 12, 1993. Revision 8 was not consistent with the CR-3 Appendix R licensing and design bases, in that the licensing and design bases only took credit for local manual operation of the ADVs to cool down the plant to the point where the decay heat removal system could be initiated.

During further review of Revision 8 to Procedure AP-990, the inspector noted that the 10 CFR 50.59 safety evaluation for this revision was inadequate in that the 50.59 did not address the limitations of the ADV backup nitrogen supply during an Appendix R fire event and the 50.59 did not take into consideration the Appendix R licensing and design bases for use of the ADVs. The PC indicated that the apparent cause of failing to recognize the limitations of the backup nitrogen supply was due to a lack of understanding of the Appendix R requirements associated with control room evacuation. During further review of PC 97-2360, the inspector noted that the concern discussed in the PC was identified by the licensee during a system readiness review (SRR) that was being performed for the main steam system. The SRR was being performed as part of the licensee's corrective actions to address NRC Violation A (Severity Level II violation with six examples of inadequate implementation of 10 CFR 50.59) discussed in NRC enforcement action (EA) 96-365, EA 96-465, and EA 96-527 dated March 12, 1997. The licensee described the corrective actions for the above Violation in letters dated April 11, 1997, and June 16, 1997. The inspector discussed this issue with licensee personnel and informed the licensee that the inadequate 50.59 evaluation for Revision 8 to Procedure AP-990 was a violation of NRC requirements. This URI was closed, and the issue changed to a non-cited violation based on the criteria described in the NRC enforcement policy (NUREG-1600) for violations identified due to previous escalated enforcement action. This issue was identified as NCV 50-302/97-11-08, Inadequate 50.59 Evaluation for Revision 8 to Procedure AP-990.

c. Conclusions

The inspector concluded from reviewing PC 97-2360 and the licensee's responses to Violation A of EA 96-365, EA 96-465, and EA 96-527, that this URI was closed, and the issue changed to a NCV based on the criteria described in the NRC enforcement policy (NUREG-1600) for violations identified due to previous escalated enforcement action.

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

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

E8.7 (Closed) URI 50-302/97-05-04: Licensee Event Report and Violation not Supplemented by Date Stated in Licensee Responses

a. Inspection Scope (92903)

This URI involved the licensee's failure to provide supplemental responses for LER 50-302/95-025-01 and VIO 50-302/95-21-03. This item also identified a weakness in the licensee's internal commitment tracking process which contributed to the supplemental response due dates being missed.

b. Observations and Findings

The licensee initiated PC 97-2413 to address this issue. The LER and VIO have been supplemented (LER 50-302/95-025-02 was submitted on May 16, 1997, and VIO 50-302/95-21-03 was supplemented on July 9, 1997). In addition, corrective actions taken or planned by the licensee included formalizing the LER and Notice of Violation (NOV) regulatory correspondence review and approval process to ensure that commitments would be captured and tracked. Other actions taken included, but were not limited to, issuance of CP-214, Regulatory Correspondence Process and Validation, separately listing commitments for each LER or NOV submittal as an attachment to the document to identify the commitment clearly, along with implementation of an extent of condition review of previously docketed regulatory correspondence to determine if other commitments have been missed. The inspector discussed the status of the corrective actions with licensee personnel who indicated that some of the corrective actions had not been completed yet. The inspector noted that the extent of condition review was still in progress at the conclusion of this inspection. Resolution of this issue was being tracked by the licensee as Restart Issue OP-32.

c. Conclusions

The inspector concluded from discussions with licensee personnel and reviewing selected documentation that the corrective actions being taken or planned by the licensee were satisfactory to address this issue. This URI 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 - N/A
- Knowledge of the Design Basis - N/A
- Compliance with Regulations - Adequate
- Operator Performance - N/A

E8.8 (Open) NRC Generic Letter 96-01; Testing of Safety-Related Logic Circuits

a. Inspection Scope (37550, 92903)

The scope of this inspection was the followup of incomplete items identified during the initial inspection conducted May 5-9, 1997 and documented in NRC Inspection Report 50-302/97-07. The following items were not previously completed and remained open for further NRC examination:

- Reactor Protection System review and validation.
- Closure of the eight open PCs.
- Submittal of the final GL 96-01 Report by the contractor.
- Final review, approval, and closure by the licensee of all GL 96-01 documents.

During this inspection, the inspectors continued to examined the licensee's actions to date relative to the testing of TS safety-related logic circuits described in GL 96-01.

b. Observations and Findings

During the initial GL 96-01 inspection, the inspectors reviewed the licensee's program. The GL 96-01 program was being managed by a licensee's project engineer and was being implemented by an offsite contractor and an onsite engineer for review and validation of the initial offsite report. The onsite engineer provided for each system or function a final report that identified the results of the completed validation review, corrective action implemented, and open items (PCs). The onsite engineer had completed all the work except for review and validation of the Reactor Protection System (RPS).

During this inspection the inspector verified that the contractor's onsite engineer had completed the review and validation for the RPS. Two problems were identified concerning RPS trip function testing where Procedures SP-110A, B, C, and D required revision. These concerns were documented in PC No. 97-2051 and PC No. 97-2053. This concern was identified as Restart Issue R-01F for tracking and close out.

The inspector verified that the contractor completed all GL 96-01 work by reviewing their final submittal letters and reports dated July 1997. The inspector also verified that all comments concerning GL 96-01 from the independent reviewer had been satisfactorily resolved. The inspector concluded the GL 96-01 program was well managed, thorough, and very effective in identifying the logic testing problems.

The contractor's onsite engineer was in the process of finalizing the "close out package". The following GL 96-01 items, tracked as Restart Issues R-01, 01A, 01B, 01C, 01D, 01E, 01F, and 01G, remain open. These open items consist of PCs that were in the process of having corrective action implemented but not yet complete.

c. Conclusions

The inspector concluded that the licensee had completed work implementing an effective program to meet the intent of GL 96-01. The onsite contract engineer had been effective in reviewing, validating, and identifying deficiencies (PCs). However, the GL 96-01 program had not been completed at this time. The inspectors assessed the licensee's performance, with respect to the licensee's response to GL 96-01, 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

E8.9 DC System Failure Modes & Effects Analysis (FMEA)

a. Inspection Scope (37550, 92903)

The scope of this inspection was the followup of incomplete items identified during the initial inspection conducted May 5-9, 1997 and documented in NRC Inspection Report 50-302/97-07. The following items were not previously completed and remained open for further NRC examination:

- The Class 1E 120 VAC Vital Bus FMEA
- Open Precursor Cards
- Final Submittal by the Contractor
- Final review, approval, and closure of all FEMA documents

During this inspection, the inspectors continued to examine the licensee's actions to date relative to the DC System FMEA.

b. Observations and Findings

During the initial FMEA inspection, the inspectors reviewed and verified that the licensee's program was performed for a specific accident scenario of Loss of Coolant Accident (LOCA)/Loss of Offsite Power (LOOP)/Loss of DC safety-related power. The DC Power FMEA included the examination of 22 systems that were powered from the safety-related Class 1E 125/250VDC system. Also included was the 120 VAC Vital Bus powered from the Class 1E inverters. The DC Power FMEA project was managed by a licensee's engineering manager and implemented by offsite contractors.

The inspector reviewed and verified that the contractors satisfactorily completed the last remaining FMEA part, the Class 1E 120 VAC Vital BUS System fed from the inverters. The inverters were powered from the 125/250 VDC safety-related batteries. The contractor's 120 VAC FMEA submittals included (1) Miscellaneous Circuits and RG 1.97 Variables, dated June 10, 1997, and 2) Engineered Safeguards Actuation System

c. Conclusions

The inspector concluded that the licensee had completed work implementing an effective program to meet the intent of GL 96-01. The onsite contract engineer had been effective in reviewing, validating, and identifying deficiencies (PCs). However, the GL 96-01 program had not been completed at this time. The inspectors assessed the licensee's performance, with respect to the licensee's response to GL 96-01, 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

E8.9 DC System Failure Modes & Effects Analysis (FMEA)

a. Inspection Scope (37550, 92903)

The scope of this inspection was the followup of incomplete items identified during the initial inspection conducted May 5-9, 1997 and documented in NRC Inspection Report 50-302/97-07. The following items were not previously completed and remained open for further NRC examination:

- The Class 1E 120 VAC Vital Bus FMEA
- Open Precursor Cards
- Final Submittal by the Contractor
- Final review, approval, and closure of all FEMA documents

During this inspection, the inspectors continued to examine the licensee's actions to date relative to the DC System FMEA.

b. Observations and Findings

During the initial FMEA inspection, the inspectors reviewed and verified that the licensee's program was performed for a specific accident scenario of Loss of Coolant Accident (LOCA)/Loss of Offsite Power (LOOP)/Loss of DC safety-related power. The DC Power FMEA included the examination of 22 systems that were powered from the safety-related Class 1E 125/250VDC system. Also included was the 120 VAC Vital Bus powered from the Class 1E inverters. The DC Power FMEA project was managed by a licensee's engineering manager and implemented by offsite contractors.

The inspector reviewed and verified that the contractors satisfactorily completed the last remaining FMEA part, the Class 1E 120 VAC Vital BUS System fed from the inverters. The inverters were powered from the 125/250 VDC safety-related batteries. The contractor's 120 VAC FMEA submittals included (1) Miscellaneous Circuits and RG 1.97 Variables, dated June 10, 1997, and 2) Engineered Safeguards Actuation System

(ESAS) dated June 26, 1997. In addition, the inspector verified that the following PCs were satisfactorily closed and the appropriate corrective action was implemented: 1) 97-0238, 2) 97-0491, 3) 97-1351, and 4) 97-1353. The following PCs have not been verified closed: 1) 97-0292, 2) 97-0294, 3) 97-0487, 4) 97-0489, 5) 97-0492, 6) 97-1352, 7) 97-1871, 8) 97-1870, 9) 97-2468, and 10) 97- 2485. Two PCs, 97-4244 and 97-4354, identified with the 120 VAC inverters, were recently opened and were under evaluation by the licensee.

c. Conclusions

The inspectors concluded that the licensee was in the process of completing the implementation of an effective Class 1E DC Power FMEA program. The contractors had completed the last segment, 120 VAC Vital Bus fed from inverters, in a satisfactory manner. Several PCs remained open and the licensee had not completed their final review and closeout. The inspectors assessed the licensee's performance, with respect to this DC Power FMEA program, in the five areas of continuing NRC concern:

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

E8.10 Commercial Grade Dedication Process

a. Inspection Scope (37551)

The inspector reviewed the process that the licensee uses for dedicated commercial grade items for use in safety related applications.

b. Observations and Findings

The inspector reviewed a sampling of commercially procured items that had been dedicated for use in safety related applications. The procedure controls which implement the dedication process were contained in the Nuclear Procurement and Storage Manual. Section 6.3, Commercial Method, provides detailed instructions, including the requirement for Nuclear Procurement Engineering Services to complete a Safety-Related Procurement Checklist to ensure the adequate stipulation of applicable requirements and must complete a Functional Analysis/Critical Characteristics Review (FA/CCR) form to indicate any special requirements necessary for replacement items.

These forms were normally reviewed by the Procurement Engineer, who resolved any discrepancies with the applicable design engineering personnel. A receipt inspection plan was developed, based on the completed FA/CCR form.

The inspector reviewed a number of dedication packages and determined that the licensee was complying with the approved program.

c. Conclusions

Based on the limited review performed by the inspector, no problems were noted with the licensee's commercial grade item dedication process.

IV. Plant Support

R8 Miscellaneous RP&C Issues

R8.1 (Closed) Restart Item RMG 29/30: Seismic Mounting of High Range (HR) Rad Monitor (FPC Restart Issue D-19) (92904)

This item concerned: 1) the seismic qualification of the mounting configuration of the HR Radiation Monitors, RMG-29 and RMG-30 and associated wiring, and 2) the routing of RMG-29 and RMG-30 safety related wiring through a non-safety related cabinet. The licensee initiated PR-96-0267 to document these items. Precursor Card 97-0132 was processed when the initial concern of safety related wiring passing through a non-safety related cabinet was not resolved by PR-96-0267 corrective action. In response to this issue the licensee:

- performed a seismic verification of the Main Control Board (MCB) backpanel (report #020-97-002-R);
- performed a calculation (Seismic Calculation S-97-0051) to document the structural qualification of the floor stand supporting RMG-29 and RMG-30; and
- upgraded the Integrated Control System Auxiliary Relay cabinet to safety related (material upgrade form #0116-97 and CIDP #97061001).

The inspector reviewed the above documentation, interviewed the licensee staff involved in the seismic evaluations, and inspected the physical layout of the radiation monitor floor stand and wiring layout. Seismic evaluations and material upgrades were adequate. The inspector identified two concerns with the physical installation. First, there was very little clearance between the radiation monitor modules and the panel of the MCB which they pass through. Secondly, the top cover of the support stand did not extend far enough to provide protection for the radiation monitor cables from falling components mounted above RMG-29 and 30. These concerns did not appear to be addressed in the seismic verification report. Through follow-up discussions with the licensee staff that performed the seismic walkdowns and a review of the seismic walkdown worksheets, the inspectors determined that these items had been evaluated and were not a concern.

- Calculations showed that the radiation monitor support stand and the MCB panel were both very stiff and would move the same amount with the floor. Therefore, there would be very little or no difference in their movement during a seismic event.

- The licensee staff judged that the electronic modules mounted above RMG-29 and 30 would not fall during a seismic event. This type of judgement call was allowed based on the Seismic Qualification Utility Group Generic Implementation Procedure.

The inrpector concluded that the licensee's evaluations and corrective actions were adequate. The staff performing the seismic verifications were knowledgeable and performed thorough walkdowns. Based on the licensee's corrective actions this restart item was closed.

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

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

S1 Conduct of Security and Safeguards Activities

S1.1 General Comments (71750)

The licensee implemented an exemption to their license on July 24, 1997, to use a Biometrics hand geometry system and allow security badges to be taken offsite. The inspectors observed that the licensee had prepared well in advance for this transition, had a lost badge policy that was well promulgated, and implemented the change without any problems. The inspector concluded the licensee effectively prepared and executed this change very well.

F2 Status of Fire Protection Facilities and Equipment

F2.1 Operability of Fire Protection Facilities and Equipment

a. Inspection Scope (64704)

The inspector reviewed maintenance's list of equipment out of service and the Fire Protection Impairment Log which listed the inoperable or degraded fire protection systems to assess the licensee's performance for returning degraded fire protection components to service. In addition, walkdown inspections were made to assess the material condition of the plant's fire protection systems, equipment, features and fire brigade equipment.

b. Observations and Findings

Maintenance and Operability of Fire Protection Equipment and Components:

As of August 4, 1997, there were 23 inoperable fire barrier penetration sealing devices and one inoperable sprinkler system within the safety related areas of the plant.

The inoperable sprinkler system provided protection for the 119' elevation of the Intermediate Building. This system was removed from service for corrective maintenance on July 14, 1997, and was returned to service on August 5, 1997. Appropriate compensatory measures, consisting of a continuous fire watch with backup fire suppression capability was implemented while this system was out of service. This system was out of service for 21 days. This exceeded the 14 days permitted by the Fire Protection Plan and will require a Special Report to be submitted to the NRC. The preparation of this report was in process at the conclusion of this inspection.

Of the 23 inoperable fire barrier penetration seal devices, 13 were placed out of service in 1997, 5 in 1996, 1 in 1994, 3 in 1993 and 1 in 1992. The licensee had implemented the appropriate compensatory measures for these inoperable and degraded fire barriers as required by the Fire Protection Plan. However, the use of compensatory measures for long term degraded fire protection components in lieu of correcting the deficiency was a poor practice. This issue was also addressed during a recent licensee's QA audit. PCs had been issued to perform an evaluation and implement appropriate corrective action for these issues (PCs 97-0285, 2873 and 3011). A subsequent inspection will be performed to determine if all of the required fire protection systems are operable prior to restart. This concern was identified as IFI 50-302/97-11-09, Correction of Fire Protection Discrepancies prior.

There were 108 fire protection related maintenance work requests outstanding during the QA fire protection audit in June 1997. This number had been reduced to approximately 58 by August 4, 1997. Forty of these work requests were issued in 1997, 6 in 1996, 3 in 1995, and 7 in 1989 through 1993. None of these deficiencies resulted in an NRC required fire protection system being out of service or inoperable. Most of the items were primarily associated with minor issues that did not effect the operability of the fire protection systems, such as small leaks in the fire protection piping system. The inspector concluded that significant progress had been made in recent months to reduce the maintenance backlog associated with fire protection components.

During the plant tours, the inspector noted that the material condition and maintenance of the operable fire protection systems were satisfactory.

Fire Brigade Equipment:

Fire brigade equipment was stored on mobile carts located in several areas of the plant. The fire brigade turnout gear was stored in a room in the maintenance shop adjacent of the Turbine Building. Each active fire brigade member was assigned dedicated fire brigade turnout gear, consisting of a coat, pants, boots, helmet, gloves, etc. The equipment was properly stored and well maintained.

c. Conclusions

The number of degraded fire protection features was high. Significant action had been taken to reduce the number of open maintenance work orders. The material condition of the fire protection components was satisfactory and the fire brigade equipment was properly stored and well maintained.

F2.2 Surveillance of Fire Protection Features and Equipment

a. Inspection Scope (64704)

The inspectors reviewed the following completed surveillance and test procedures to determine if the specified surveillance frequencies met the NRC guidelines.

- SP-190, Functional and Operability Test of Auxiliary Building Fire Detection Instrumentation and Reactor Building Purge and Exhaust Fan POC Detector Interlocks, Revision 10. Test Completed January 29, 1997.

- SP-363, Fire Protection System Test (Fire Pumps), Revision 29. Test completed December 3, 1996.

- SP-407, Fire Barrier Penetration Seals, Revision 26. Completed May 16, 1997.

- SP-408, Fire Protection Flow Test, Revision 10 (3 Year Fire Protection Hydraulic Performance Verification). Completed December 15, 1995.

- SP-501B, Halon System Functional Check, Revision 7 (Cable Spreading Room Suppression System). Tests performed March 21 and 22, 1997.

Also, the surveillance procedures and frequency specified by the Fire Protection Plan Section 1.6 for the NRC required fire protection components were reviewed.

b. Observations and Findings

The completed fire protection surveillance tests reviewed by the inspectors were appropriately completed and met the acceptance criteria. However, several procedural discrepancies and needed improvements were identified. For example, Procedure SP-408 did not require sufficient flow tests in all sections of the distribution system to verify the hydraulic performance of the fire protection water system which was the intent of the procedure. Procedure SP-363 did not address calibration requirements for the required test instruments to be used during the testing activities. The licensee had identified these discrepancies and revisions to these procedures were in process. The licensee states that these procedures were to be revised prior to their next scheduled performance. Verification that enhancements have been made to the fire protection surveillance procedures will be performed during the post restart inspection of the fire protection features and is identified as IFI 50-302/97-11-10, Post Restart Fire Protection Inspection to Validate Completion of Fire Protection Enhancement Items.

The surveillance requirements for the fire protection systems were contained in Fire Protection Plan, Section 6 and Table 6.0. The inspector reviewed the surveillance procedures available and verified that each surveillance requirement of the Fire Protection Report had been addressed by a site procedure and that the scheduled performance frequency met the NRC requirements.

c. Conclusions

Implementation of the surveillance and test procedures was satisfactory. Revisions were being made on two of the procedures reviewed to ensure they meet the appropriate test objectives.

F3 Fire Protection Procedures and Documentation

F3.1 Fire Protection Procedures Reviewed

a. Inspection Scope (64704)

The inspector reviewed the following procedures for compliance with the NRC requirements and guidelines:

- CR-3 Fire Protection Plan, Revision 13.
- CR-3 Pre-Fire Plan, Revision 4.
- CR-3 Fire Hazards Analysis, Revision 7.
- AI-1000, Good Housekeeping/Material Condition Program, Revision 33.
- AI-2100, Guidelines for Handling Use and Control of Transit Combustibles, Revision 7.
- AI-2205, Administration of CR-3 Fire Brigade Organization, Revision 12.
- AI-2210, Fire Watch Program.
- CP-118, Fire Prevention Work Permit, Revision 20. (Hot Work Permits)
- SP-809, Weekly Inspection - Fire Protection, Revision 7.

Plant tours and reviews were also performed to assess procedure compliance.

b. Observations and Findings

The procedures listed above were the principal procedures issued to implement the facility's fire protection program. These procedures contained the requirements for program administration, controls over combustibles and ignition sources, fire brigade organization and training, and operability requirements for the fire protection systems and features. In general, the procedures were well written and met the licensee's commitments to the NRC, except as follows:

CR-3 Pre-Fire Plans: During plant tours, the inspector noted that the Pre-Fire Plans contained several minor errors. Also, pre-fire plans had not been prepared for the diesel driven generator and 2,500 gallon diesel fuel tank erected west of the Turbine Building adjacent to the condensate storage tank. The licensee stated that these discrepancies had previously been identified and revisions to the Pre-Fire Plans were in process and were scheduled to be completed by September 26, 1997. This will be verified during a subsequent NRC inspection and is identified as IFI 50-302/97-11-09, Correction of fire protection discrepancies.

CR-3 Fire Hazards Analysis (FHA): Following a walkdown inspection of battery room 3A and 4160V switchgear room 3A, the inspector reviewed the FHA to determine if the large quantity of combustible Thermo-Lag fire barriers installed in these areas had been included in the combustible fire load calculations for these room. The current FHA, Revision 7, did not include this combustible material. However, this discrepancy had been identified by the licensee and the licensee was revising the FHA to address this issue. Revision 8 to the FHA will include the additional combustible fire loads resulting from the installation of the Thermo-Lag materials and the calculated fire loads for each plant area will conform to the actual as built plant conditions. Revision of the FHA was scheduled to be completed 90 days after restart. Revision enhancements to the FHA will be verified during a post restart inspection of the fire protection features and is identified as IFI 50-302/97-11-10, Post restart fire protection inspection to validate completion of fire protection enhancement items.

AI-2205, Administration of CR-3 Fire Brigade Organization: AI-2205, Section 4.6, requires each fire brigade member to complete a physical examination to meet the physical demands of fire brigade and fire fighting duties. The frequencies of the physical examinations were annually for members more than 54 years of age, every two years for members 40 through 54, and every four years for members 39 and younger. An annual physical examination is delineated by the NRC guidelines and industry practice. The licensee states that the current physical examination schedule for CR-3 met their licensing requirements and was established by the licensee's medical department.

The inspector performed plant tours and noted that implementation of the site's fire prevention program for the control of ignition sources, transient combustibles, and general housekeeping was good, considering that the plant was in a major longer term maintenance and modification outage. However, the following items were noted:

- On the 95' elevation of the Auxiliary Building an 8x8 foot operations office had been erected without extending the area fire detection or sprinkler systems to provide protection for this room. The licensee did not have an evaluation to address this issue. A Precursor Card was to be written to perform an evaluation to determine the required corrective action. This issue will be reviewed during a subsequent NRC inspection and is identified as IFI 50-302/97-11-09, Correction of fire protection discrepancies prior to restart.

- The fire door separating the control room from the operation office area was found blocked in the open position. Subsequent investigation found that this door had been blocked open since 1992. There was no evaluation to justify this configuration. On August 6, 1997, the licensee issued Precursor Card 97-5764 to address this issue. This issue will be reviewed during a subsequent NRC inspection and is identified as IFI 50-302/97-11-09, Correction of fire protection discrepancies prior to restart.

- West of the Turbine Building, approximately 12 feet west of the Miscellaneous Drain Tank and 6 feet northwest of the Condensate Tank, the licensee had erected a self-contained diesel driven emergency generator structure which, when operable, will have a 2,500 gallon diesel fuel tank. The fuel tank had not been erected at time of this inspection. However, the facility's pre-fire plans had not been revised to address the fire hazards associated with the generator and fuel tank. The licensee stated that the pre-fire plans would be revised to address this issue prior to placing any fuel in this tank. This issue will be reviewed during a subsequent NRC inspection and is identified as IFI 50-302/97-11-09, Correction of fire protection discrepancies prior to restart.

The compensatory fire watch program for degraded fire protection features utilized contract employees under the general oversight of the fire protection technical staff. The inspector reviewed recently completed data sheets, reviewed the fire watch routes, and evaluated the bar code reader device being used to validate that the fire watch patrols were performed within the specified frequency. This program was effectively implemented and was considered a program strength.

c. Conclusions

Most of the fire protection program implementing procedures met the NRC requirements. The current revisions of the pre-fire plans and fire hazards analyses were not up to date. Implementation of the procedures for the control of ignition sources and transient combustibles were good. General housekeeping was also good, considering the large amount of work in process due to the long term maintenance and modification outage. An effective program was in place to meet the compensatory measures required for degraded or inoperable fire protection equipment.

F5 Fire Protection Staff Training and Qualification

F5.1 Fire Brigade Organization, Training and Drill

a. Inspection Scope (64704)

The inspector reviewed the fire brigade organization, fire brigade training program and fire brigade drill participation for compliance with the NRC guidelines and requirements.

b. Observations and Findings

Fire Brigade Organization and Training

The organization and training requirements for the plant fire brigade were established by AI-2205, Administration of CR-3 Fire Brigade Organization. The fire brigade team leader was normally one of the Assistant Nuclear Shift Supervisors. The fire brigade members consisted of at least one non-licensed operator from operations and at least three personnel members from facility services. Normally, there was a sufficient number of qualified fire brigade member personnel on the site to staff two fire brigade teams.

Each fire brigade leader and fire brigade member was required to receive initial, quarterly and annual fire fighting related training. Each brigade member was required to participate in at least one drill per year. Current NRC guidelines and industry policy are to participate in at least two drills per year. This was not a licensing basis requirement at CR-3. However, based on a review of selected fire brigade personnel the average drill attendance was 2.8 in 1995, 2.2 in 1996 and 2.0 from January through June 1997.

The inspectors reviewed the training and medical records for 12 fire brigade members and verified that the training and medical records were up to date. However, as noted in Section F3, annual fire brigade physical examinations were not specified for all fire brigade members. Otherwise, the fire brigade training program was very good. There were four State of Florida certified fire fighting training instructors in the licensee's training organization. A well-equipped fire brigade training facility equipped with a three-story smoke tower and simulated power plant equipment were provided on site to perform the annual fire brigade training and practical fire training scenarios.

Fire Brigade Drill

During this inspection, the inspectors witnessed a fire brigade drill involving a simulated fire in an electrical cable tray in the "A" 480V switchgear room on the 124' elevation of the Control Building. The response of the fire brigade to the simulated fire, the brigade leader's direction and fire brigade members' performance were good. A critique to discuss the brigade performance was held following the drill. Several items were identified which were to be considered to enhance future fire brigade and plant personnel 's performance in the event of a fire.

c. Conclusions

The fire brigade organization and training met the requirements of the site procedures and implementation of the training program was very good. Performance by the fire brigade during a drill was good. However, annual physical examinations are not performed for all fire members with physicals for personnel less than 40 performed every four years.

F6 Fire Protection Organization and Administration

F6.1 Management and Administration

a. Inspection Scope (64704)

The licensee's management and administration of the facilities fire protection programs were reviewed for compliance with the commitments to the NRC and to current guidelines.

b. Observations and Findings

At the time of this inspection, the Director, Nuclear Engineering and Projects was designated the responsibility for implementing the facility's fire protection program. A fire protection engineer was assigned the task of coordinating the engineering functions associated with the design, modifications and maintenance, activities of fire suppression systems, fire barriers, and fire barrier penetrations. Two fire protection specialists were responsible for the implementation of the fire prevention portions of the program including reviews of the plant facility for identification and correction of fire hazards and housekeeping problems, and assessment of the fire brigade performance. The organizational structure of the fire protection organization did not meet the CR-3 Fire Protection Plan; however, the plan was being revised to conform to the actual functional organization. This will be verified during a subsequent NRC inspection and is identified as IFI 50-302/97-11-09, Correction of Fire Protection Discrepancies prior to Restart.

c. Conclusions

Adequate coordination and oversight were provided over the facility's fire protection program; however, the CR-3 Fire Protection Plan had not been revised to conform to the recent reorganization of the facility's management structure that was in place at time of this inspection.

F7 Quality Assurance in Fire Protection Activities

F7.1 Audit Reports

a. Inspection Scope (64704)

The following audit reports were reviewed:

- Audit 97-05, Fire Protection and Restart Readiness, conducted May 5 through June 2, 1997.
- Audit 97-06, Integrated Audit, conducted June 2-30, 1997.
- Audit 97-04, Integrated Audit, conducted March 31 through May 5, 1997.
- Audit 97-03, Integrated Audit, conducted March 3-31, 1997.
- Audit 96-Q4-RSTRT, Integrated Audit, completed on December 20, 1996.
- Audit 96-Q3-QPD, Integrated Audit, conducted July 19 through November 15, 1996.

b. Observations and Findings

Audit 97-05 performed an assessment of the facility's fire protection program and implementing procedures and performed plant inspections. Audits 97-06, 97-05, 96-Q4-RSTRT and 96-Q3-QPD were integrated audits that provided an assessment of several plant functional areas, including fire protection. The inspector reviewed the audit reports and interviewed the audit team leader for the 97-05 audit. These audits performed a detailed comprehensive review of the CR-3 facility's fire protection program and identified a number of program discrepancies. PCS were issued to identify these finding and to develop the appropriate corrective action. The principal discrepancies identified by the audits included the following:

- Nuclear Engineering Procedures did not include requirements for appropriate fire protection reviews. (PC 97-3647)
- Plant procedures did not require the Plant Review Committee to review the Fire Protection Plan and implementing procedures as stipulated by the FSAR. (PC 97-3739)
- Fire protection requirements were not being incorporated into procedures and existing procedures were not being kept up to date. (PCS 97-3739, 3647, 3648 and 3650)
- One hour fire barrier was not provided between step-up transformers. (PC 97-3351)

- Three hour fire barriers were not provided to separate the three fire pumps. All three fire pumps were in the same building and same fire area. (PC 97-3352)

- Emergency lighting for 4160V 3A switchgear room was not properly adjusted. (PC 97-3637)

- General housekeeping inspection requirements were not being performed as stipulated by Procedure AI-1000. (PC 97-3303)

- Examples of inadequate preventive maintenance on the fire protection equipment were identified. (PC 97-3395)

- The engineering records were not updated following installation of the Mecatiss fire barrier installations. (PC 97- 4095)

- Procedures in place did not effectively track the status of degraded fire barrier penetrations which resulted in the degraded fire barrier penetrations not being corrected in a timely manner. (PC 97-2873 and 3011)

- Housekeeping discrepancies associated with the inappropriate storage of combustible materials within the plant were identified. (PC 97-1490, 1555 and 1707)

- Resolutions of identified fire protection problems were not timely and follow-up of corrective actions were weak. (PC 97-0285)

- Fire watch training was inadequate. (PC 96-4395)

- Problems related to the Special Reports for inoperable fire protection components were identified. (PC 96-4685)

The corrective action on these items had not been fully completed. A subsequent inspection will be performed to determine if the required fire protection issues are operable prior to restart. This is identified as IFI 50-302/97-11-09, Correction of Fire Protection Discrepancies prior to Restart.

c. Conclusions

The QA audits conducted in late 1996 and in 1997 were detailed and comprehensive and identified a significant number of fire protection program discrepancies. Corrective action was being implemented to resolve these issues.

F8 Miscellaneous Fire Protection Issues

F8.1 (Open) URI 50-302/96-06-10; Justification for Removal of Thermo-Lag Protection from Source Range Instrumentation (64704)

Previously, an inspector found that the Thermo-Lag fire barrier had been removed from the cabling to the source range instrumentation without an adequate technical evaluation. The licensee's position was that boron concentration in the RCS could be determined from sampling the RCS. In the past, the NRC determined that the use of boron concentration indication was acceptable as an alternative method of monitoring reactivity following a fire, if adequately addressed by the facility's procedures. Alternative reactivity indication at CR-3 was not addressed by the licensee's evaluation or referenced by the Emergency Operating Procedures (EOPs).

During this inspection, the licensee provided additional information to the inspector on the use of source range instrumentation during fire events. For normal events and events in which the reactor coolant pumps were lost, the normal operating procedures and EOPs require boron sampling if source range instrumentation was not available. However, for events in which the reactor coolant pumps were not lost, Procedures EOP-02, Vital System Status Verification, Revision 3, Change 2, and EOP-10, Post Trip Stabilization, Revision 2, Change 2, did not provide guidance if source range instrumentation was not available. EOP-10, Step 3.5, states "When Intermediate Range flux lowers to 5E-10 amps Then verify SR (source range) energizes. Continue in this procedure." The procedure did not provide any guidance if source range instrumentation was not available. The procedure is dependent on the operator's knowledge to refer to the Technical Specification, Section 3.3.9 which required shutdown margin calculations to determine reactivity levels. This was not considered desirable for operations during emergency conditions.

This issue remains open pending additional evaluation during the NRC EOP CR-3 inspection scheduled for October 1997.

F8.2 Fire Protection Related NRC Information Notices (64704)

The inspector reviewed the licensee's evaluation for the following NRC Information Notices (IN):

- IN 88-04 and IN 88-04, Supplement 1, Inadequate Qualification and Documentation of Fire Barrier Penetration Seals.
- IN 88-56, Potential Problems with Silicone Foam Fire Barrier Penetration Seals.
- IN 92-18, Potential Loss of Shutdown Capability During a Control Room Fire.
- IN 92-28, Inadequate Fire Suppression System Testing.
- IN 94-28, Potential Problems with Fire Barrier Penetration Seals.
- IN 94-58, Reactor Coolant Pump Lube Oil Fire.
- IN 95-36, Emergency Lighting.

Conclusions

The licensee's initial evaluations for these INs were weak and appropriate corrective actions were not initially identified, as follows:

INs 88-04, 88-56 and 94-28: Records were not available to support the design of the fire barrier penetration seals installed at CR-3. At the time of this inspection the licensee was evaluating this issue to determine the appropriate action. The action taken by the licensee to resolve this issue will be reviewed during a subsequent NRC inspection. This is identified as IFI 50-302/97-11-09, Correction of Fire Protection Discrepancies prior to Restart.

IN 92-18: A recent licensee's evaluation identified 29 MOVs could potentially be damaged in the event of an Appendix R type fire. Precursor Card 97-3963 was issued to evaluate this condition and implement the appropriate corrective action. The licensee did not consider this issue to be outside the licensing basis for CR-3 and, therefore not reportable. Corrective action will be completed on 15 of these valves prior to restart from the current outage. Corrective action for the remaining valves will be completed at a later date.

IN 92-28: This information notice identified problems with excessive leakage from enclosures provided with gas fire suppression systems, stressed the need to maintain the concentration for gas fire suppression systems in order for the systems to function effectively and identified an approved testing method to validate a systems operability. The licensee concluded that the initial construction testing was adequate to assure that the Halon gas fire suppression system installation for the cable spreading room was satisfactory. The control building complex has a history of excessive leakage through the various penetrations of the control room boundary resulting in the failure to meet the habitability requirements for the control room during accident conditions. The cable spreading room is also located in the control building. The failure to perform testing to determine if the leakage through the various penetrations in the walls, floors, ceilings, etc. of the spreading room enclosure would render the gas suppression system inoperable was identified as a past program weakness.

IN 94-58: The oil collection systems for the reactor coolant pumps do not meet the NPC requirements. Modifications to these systems are in process and will be completed prior to the restart from the current outage. This issue is an item to be completed prior to restart and was identified as restart item D-11-B.

IN 95-36: The licensee in the past has used the vendor's recommendations for performing periodic testing of the 8-hour emergency battery powered lighting units. Procedures required the lighting units to be tested during the first and third quarter of each year to verify operability (lights illuminated) and during the second and forth quarter each year to be operated for three to five minutes. The licensee considered periodic 8-hour discharge testing to be destructive to the batteries. However, this testing program had not been totally successful, in that, a significant number of battery failures continued to be identified during the testing activities. Therefore, the licensee was considering a program to replace the battery units periodically. The final resolution of this issue will be reviewed during a subsequent NRC post restart inspection of the fire protection features and is identified as IFI 50-302/97-11-10, Post Restart Fire Protection Inspection to Validate Completion of Fire Protection Enhancement Items.

V. Management Meetings

X1 Exit Meeting Summary

The inspection scope and findings were summarized on August 15, 1997. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

Management Meeting Summary

X A meeting was held on July 21, 1997, at NRC Headquarters to discuss the Emergency Diesel Generator upgrade, Emergency Feedwater system and other technical issues. A separate meeting summary was issued on August 1, 1997.

X3.2 A meeting was held on July 30, 1997, in Region II to discuss implementation of a Security Improvement Plan. A separate meeting summary was issued on August 8, 1997.

PARTIAL LIST OF PERSONS CONTACTED

<u>Licensees</u>

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

NRC

J. Bartley, Resident Inspector, Farley (July 28 through 30, 1997)
K. Landis, Branch Chief, Region II (July 31 through August 1, 1997)
M. Miller, Reactor Inspector, Region II (July 14 through 18, July 28 through August 1, 1997)
W. Miller, Reactor Inspector, Region II (August [illegible] through 8, 1997)
L. Moore, Reactor Inspector, Region II (August 11 through 15, 1997)
S. Ninh, Project Engineer, Region II (July 16 through 18, 1997)
D. Orrik, NRR (August 4 through 6, 1997)
A. Qualantone, NRC Contractor (August 4 through 6, 1997)
R. Schin, Reactor Inspector, Region II (July 28 through August 1, August 11 through 15, 1997)
B. Sevario, NRC Contractor (August 4 through 6, 1997)
R. Speer, NRC Contractor (August 4 through 6, 1997)
L. Stratton, Physical Security Specialist, Region II (August 4 through 6, 1997)
M. Thomas, Reactor Inspector, Region II (July 14 through 18, July 28 through August 1, 1997)
F. Vangel, NRC Contractor (August 4 through 6, 1997)
J. York, Reactor Inspector, Region II (August 11 through 15, 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 64704:	Fire Protection Program
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92901:	Followup - Operations
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Type	Item Number	Status	Description and Reference
IFI	50-302/97-11-01	Open	RCS Reduced Inventory Level Indication Problems. (Section 01.2)
VIO	50-302/97-11-02	Open	Inadequate Procedural Guidance for Quality-Related Work. (Section 01.3)
IFI	50-302/97-11-04	Open	Corrective Actions for Approximately 4000 Precursor Cards Not Tracked to Completion. (Section 07.1)

VIO	50-302/97-11-06	Open	Failure to Follow Licensee Procedure NEP-254. (Section E3.1)
VIO	50-302/97-11-07	Open	Deletion of Water Quality Requirements from the FSAR. (Section E8.4)
IFI	50-302/97-11-09	Open	Correction of Fire Protection Discrepancies Prior to Restart. (Sections F2.1, F3.1 and F7.1)
IFI	50-302/97-11-10	Open	Post Restart Fire Protection Inspection to Validate Completion of Fire Protection Enhancement Items. (Sections F2.2 and F3.1)

Closed

Type	Item Number	Status	Description and Reference
NCV	50-302/97-11-03	Closed	Corrective Action Procedures Failed to Require Quality Records. (Section O7.1)
VIO	50-302/95-16-03	Closed	Inadequate Procedures for Operation of the Makeup Pump 1A Cooling Water. (Section O8.1)
LER	50-302/95-10-01	Closed	Inadequate Procedure Causes Low Cooling Water Flow to Makeup Pump Resulting in Operation Outside the Design Basis. (Section O8.1)
VIO	50-302/96-20-01	Closed	Failure to Adhere to Reactor Coolant System Cooldown Limits. (Section O8.2)
LER	50-302/93-02-02	Closed	Switchyard Cable Failure Caused Degraded Voltage of Class 1E Electrical Busses and Actuation of Emergency Diesel Generators. (Section O8.3)
LER	50-302/93-02-03	Closed	Switchyard Cable Failure Caused Degraded Voltage of Class 1E Electrical Busses and Actuation of Emergency Diesel Generators. (Section O8.3)
IFI	50-302/96-03-15	Closed	HPI Flow Indicator 50.59 and Tech Spec Bases Change. (Section O8.4)

LER	50-302/96-07-01	Closed	HPI Line Break With Loss of Battery Could Result in Reliance on Inadequate Accident Mitigation Instrumentation. (Section O8.4)
NCV	50-302/97-11-05	Closed	Removal of ASV-15 Without Reviewing and Complying with Work Instructions. (Section M1.1
VIO	50-302/96-20-02	Closed	Failure to Follow Procedure AI-400C for Review and Development of Maintenance Procedure PM-191. (Section M8.1)
IFI	50-302/96-201-12	Closed	Conduit Sizing Criteria - Jamming Ratio Not Considered. (Section E8.2)
URI	50-302/97-02-02	Closed	Deletion of Water Quality Requirements from the FSAR. (Section E8.4)
VIO	50-302/97-02-04	Closed	Failure to Conduct TS Logic Testing. (Section E8.5)
URI	50-302/97-05-02	Closed	50.59 Safety Evaluation does not Address Operation of the Atmospheric Dump Valves from the Remote Shutdown Panel During an Appendix R Fire Event. (Section E8.6)
NCV	50-302/97-11-08	Closed	Inadequate 50.59 Evaluation for Revision 8 to Procedure AP-990. (Section E8.6)
URI	50-302/97-05-04	Closed	LER and Violation not Supplemented by Date Stated in Licensee Responses. (Section E8.7)

Discussed

Type	Item Number	Status	Description and Reference
URI	50-302/96-201-04	Open	Nonsafety-Related Positioners on Safety-Related Valves. (Section E8.1)
VIO	50-302/96-09-05	Open	Failure to Incorporate Design Information into Operations Procedures. (Section E8.3)

LER	50-302/95-25-01	Closed	Inadequate Isolation of Safety/Non-Safety Related Circuits. (Section E8.7)
VIO	50-302/95-21-03	Open	Failure to Isolate the Class IE From the Non-Class IE Electrical Circuitry for the RB Purge and Mini-Purge Valves. (Section E8.7)
GL	GL 96-01	Open	NRC Generic Letter 96-01, Testing of Safety-Related Logic Circuits. (Section E8.8)
URI	50-302/96-06-10	Open	Justification for Removal of Thermo-Lag Protection from Source Range Instrumentation. (Section F8.1)

LIST OF ACRONYMS USED

ADV - Atmospheric Dump Valve
AI - Administrative Instruction
AP - Abnormal Procedures
ASME - American Society of Mechanical Engineers
ASV - Auxiliary Steam Valve
BSP - Building Spray Pump
BWST - Borated Water Storage Tank
CFR - Code of Federal Regulations
CGWR - Commercial Grade Work Request
CNO - Chief Nuclear Operator
CP - Compliance Procedure
CR - Control Room
CR3 - Crystal River Unit 3
DC - Decay Heat Closed Cycle Cooling System
DCN - Design Change Notice
DHP - Decay Heat Pump
DIR - Design Input Record
EA - Enforcement Action
ECCS - Emergency Core Cooling System
EDBD - Enhanced Design Basis Document
EDG - Emergency Diesel Generator
EFW - Emergency Feedwater
EGDG - Emergency Diesel Generator
EMI/RFI - Electromagnetic and Radio Frequency Interference
EOP - Emergency Operating Procedures
FA/CCR - Functional Analysis/Critical Characteristics Review
FHA - Fire Hazards Analysis
FMEA - Failure Modes and Effects Analysis
FPC - Florida Power Corporation
FSAR - Final Safety Analysis Report
GL - Generic Letter
gpm - Gallons Per Minute
HP - Health Physics

HPI	-	High Pressure Injection
HPT	-	Health Physics Technician
HR	-	High Range
IFI	-	Inspection Followup Item
IGA	-	Intergranular Attack
IN	-	Information Notice
IOC	-	Interoffice Correspondence
IPAP	-	Integrated Performance Assessment Process
IPTE	-	Infrequently Performed Test and Evolutions
IR	-	Inspection Report
ISI	-	Inservice Inspection
IST	-	Inservice Testing
ITS	-	Improved Technical Specifications
Kw	-	Kilowatts
LER	-	Licensee Event Report
LOCA	-	Loss of Coolant Accident
LOOP	-	Loss of Offsite Power
MAR	-	Modification Approval Record
MCB	-	Main Control Board
MOV	-	Motor Operated Valve
MP	-	Maintenance Procedure
MUP	-	Make-up Pump
MUV	-	Make-up Valve
NCV	-	Non-cited Violation
NEP	-	Nuclear Engineering Procedure
NOTES	-	Nuclear Operations Tracking & Expediting System
NOD	-	Nuclear Operations Department
NOV	-	Notice of Violation
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
OP	-	Operating Procedure
OTSG	-	Once Through Steam Generator
PC	-	Precursor Card
PDR	-	Public Document Room
PEERE	-	Plant Equipment Equivalency Replacement Evaluation
PM	-	Preventive Maintenance
PR	-	Problem Report
PRC	-	Plant Review Committee
psig	-	Pounds Per Square Inch Gauge
PT	-	Performance Testing Procedure
QA	-	Quality Assurance
QPS	-	Quality Programs Surveillance
RB	-	Reactor Building
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
REA	-	Request for Engineering Assistance
RG	-	Regulatory Guide
ROT	-	Reactor Operator Training

RP&C	-	Radiological Protection and Chemistry
RPS	-	Reactor Protection System
RWP	-	Radiation Work Permit
SD	-	Site Drain
SDBI	-	Suspected Design Basis Issue
SFP	-	Spent Fuel Pool
SM	-	Shift Manager
SP	-	Surveillance Procedure
SRR	-	System Readiness Review
SSFI	-	Safety System Functional Inspection
SW	-	Service Water
STI	-	Short Term Instruction
TDBD	-	Topical Design Basis Document
TS	-	Technical Specification
UHS	-	Ultimate Heat Sink
URI	-	Unresolved Item
USQ	-	Unreviewed Safety Question
USQD	-	Unreviewed Safety Question Determinatiun
VIO	-	Violation
WD	-	Waste Disposal
WR	-	Work Request

ENFORCEMENT DISPOSITION TABLE

This information is being provided for record purposes to close the identified Escalated Enforcement Issues (EEIs) and requires no response from the licensee. Following the evaluation of the licensee response to each apparent violation, a Notice of Violation (NOV) was issued on September 5, 1997. Based on the NOV issued, the EEIs are closed. The cited violations are identified in the NOV and are being tracked per the following Enforcement Disposition Table as Enforcement Actions (EAs). Each individual NOV has a specific NOV ID Number.

EEI NO.	TITLE	EA NO.	NOV ID NO.	TITLE
302/97-09-01	Unreviewed safety question involving added EDG protective trips	97-330	VIO 01013	Failure to perform an adequate safety evaluation for 1987 modification which added five protective trips to each EDG
302/97-09-02	Failure to update the FSAR to include added emergency diesel generator trips	97-300	VIO 01023	Failure to update the FSAR to describe the added EDG protective trips