U. S. NUCLEAR REGULATORY COMMISSION

REGION II

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Licensee:	Florida Power Corporation
Facility:	Crystal River 3 Nuclear Station
Location:	15760 West Power Line Street Crystal River, FL 34428-6708
Dates:	September 26 through November 6, 1999
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Enclosure

EXECUTIVE SUMMARY

Crystal River 3 Nuclear Station
NRC Inspection Report 50-302/99-07

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a six-week period of resident inspection; in addition, it includes the results of announced inspections by a radiation specialist and several regional reactor inspectors.

Operations

- Overall, operators performed very well during numerous significant refueling outage operating evolutions. Operators followed procedures and altered plant conditions methodically. Supervisory oversight of plant condition changes was thorough and consistent. Operational focus on shutdown reactor safety parameters was clear and consistent (Section O1.1).

- Rod drop testing was effectively controlled. Senior management oversight was continual. Operators closely monitored plant instrumentation and distractions were limited during the testing. Communications were complete and precise (Section O1.3).

- Preparations for refueling outage reactor coolant system inventory reductions were thorough. Dedicated oversight teams were established well in advance of the outage. Revised guidance, new operator aids, and a different initial draindown methodology were developed. The draindowns were closely supervised and operators were cognizant of all level indication instrument capability and readings. Level instrument performance was consistently accurate and stable, validating that previous concerns had been addressed. Temporary Instruction 2515/142 was completed to evaluate the licensee's analysis of Generic Letter 98-02 regarding reactor inventory control. The licensee's analysis was appropriate (Section O1.4).

- Fuel movement was precisely controlled. Communications were consistently effective and utilized 3-way techniques (Section O1.5).

- A non-cited violation was identified for incorrect attachment of the reactor plenum to the tripod lifting device. Detailed procedural guidance for attaching the plenum to the tripod was not followed by contract refueling personnel and the error was not detected by licensee refueling senior operators (Section O1.5).

- Two non-cited violations were identified for operator errors involving poor procedure adherence that resulted in inadvertent water level decreases in the spent fuel pool and reactor coolant system. Operators responded promptly to the events and terminated the draindowns prior to any impact on reactor coolant or spent fuel cooling systems. Failure to properly implement procedures was the primary cause of these two events, but contributing causes included deficiencies in communications, poor self-checking techniques, and an outage schedule change which moved up some draining activities. Licensee investigations were thorough and corrective actions were prompt and appropriate (Section O1.6).

Maintenance

- Maintenance activities were performed methodically and in accordance with procedures. Unexpected testing results were properly reviewed, corrected, and retested. Refueling outage containment penetration control was adequate. Unannounced drills demonstrated prompt and adequate containment closure capability (Section M1.1).

- The licensee responded adequately to two emergent maintenance issues during the refueling outage. Leakage in nuclear services closed cycle cooling heat exchangers was appropriately dispositioned and a scope reduction to planned emergency diesel generator maintenance was adequately justified (Section M1.2).

- The control complex habitability envelope integrated leak test was conducted methodically and test results were satisfactory (Section M1.3).

- Inservice inspection activities were being performed in accordance with code and licensee requirements (Section M1.5).

- A detailed flow assisted corrosion monitoring program was in place and implemented in accordance with procedural requirements (Section M1.6).

Engineering

- Post-modification testing of major high pressure injection and low pressure injection system modifications was effective. Functional tests were detailed and reflected extensive preparatory work. Pre-job briefings were very thorough, management oversight was continuous, and test performance was methodical. Results were satisfactory and unexpected problems were appropriately dispositioned. Testing impact on critical shutdown plant safety functions was closely monitored and no problems occurred (Section E1.1).

- The licensee thoroughly analyzed large amounts of control rod assembly data to address problems identified by end-of-cycle rod drop testing, including several slow drop times. Fuel assembly bowing and thermal barrier induced hydraulic drag were attributed as causes. Corrective actions, including resetting hold-down springs and replacing thermal barriers, were appropriate (Section E1.2).

- Two failed yoke assemblies and a sheared radiator clutch drive shaft on the B emergency diesel radiator fan shaft were found by alert mechanics following overspeed testing. Fabrication problems with the yoke assembly were noted and addressed in corrective repair actions, but the initiating cause of the yoke failure was undetermined. Inspectors verified the physical evidence supported the licensee determination and noted the design ratings of the radiator drive train were adequate. Repair actions and long-term corrective actions were comprehensive and appropriate (Section E1.3).

- The engineering organization was effective in designing and implementing major emergency feedwater, high pressure injection, and low pressure injection modifications. The modification packages were generally complete, accurate, and of good quality. Installation and testing were satisfactory, and problems were being appropriately identified and resolved (Section E1.4).

Plant Support

- Overall, radiological controls were maintained and implemented in accordance with the Updated Final Safety Analysis Report, Technical Specifications, license conditions, and 10 CFR Part 20 requirements. Excluding workers' internal exposures, licensee dose assessments associated with unanticipated contamination events were adequate (Section R1.1).

- Several examples of poor radiological practices were identified regarding dosimetry use by personnel, contaminated area work practices, visibility of reactor building radiation postings, and limited worker communication with the health physics staff (Section R1.1).

- A non-cited violation was identified for failure to conduct accurate and timely evaluations of worker exposure from potential radioactive material intakes. Occupational worker doses were determined to be within administrative and regulatory limits (Section R1.1).

- As Low As Reasonably Achievable program activities and initiatives for the refueling cycle were conducted in accordance with approved procedures with outage cumulative dose expenditure revised upward from original estimates due to elevated dose rates, inexperienced workers, and emergent work activities (Section R1.2).

Report Details

Summary of Plant Status

The unit began the inspection period at full power and remained essentially at that level until September 30, when a power reduction was initiated to begin a scheduled refueling outage. On October 1, the reactor was tripped to initiate the outage. The unit remained shutdown as the outage continued for the rest of the report period.

I. Operations

O1 Conduct of Operations

O1.1 Routine Conduct of Operations Reviews (71707)

The resident inspectors conducted periodic reviews of plant operations, including shift turnovers, operator logs, and main control room board walkdowns. Compliance with Technical Specification (TS) requirements was verified as plant operating modes were changed. The inspectors routinely toured safety-related plant areas to verify the physical condition of selected plant equipment and structures and to monitor for acceptable system operation. The inspectors observed the performance of several significant evolutions and reviewed associated documentation including procedures for plant shutdown, plant cooldown, decay heat removal system operation, plant startup, plant heatup, and low temperature overpressure control procedural guidelines.

The inspectors observed consistent safety-conscious performance by operators. Operators were cognizant of plant conditions and methodically implementated procedures. Numerous off-shift senior reactor operators were available to prepare for plant changes in advance so most evolutions were very well controlled. Additionally, many evolutions were designated Infrequently Performed Tests or Evolutions (IPTE) and therefore received senior management oversight and more extensive preparation. Consequently, very few concerns were independently identified by inspectors. Although three non-cited violations are identified in subsequent Operations sections of this report, they were identified by the licensee. The inspectors considered them not reflective of overall Operations performance. Overall operational performance during the outage was good. Noteworthy observations and specific events are detailed in subsequent sections.

The inspectors also noted consistently strong focus on shutdown reactor safety. Administrative Instruction (AI)-504, Guidelines for Cold Shutdown and Refueling, was significantly revised from the previous outage and provided improved guidance. Operational information sheets used during the outage focused on key shutdown safety functions such as decay heat removal (DHR) and electrical power availability. Large status boards were created for the control room and work control center to track and display the status of each key safety function. Time to core boil and time to core uncovering estimates for a loss of DHR were updated several times per shift by the operators via a mainframe computer program. The licensee outage schedule was constructed to minimize shutdown safety risk and controls were implemented to ensure emergent work activities did not alter the pre-planned analysis. The licensee controls and safety focus were very effective at maintaining appropriate operational safety margins and defense in depth.

O1.2 Plant Shutdown to Initiate Refueling Outage (71707)

The inspectors observed that the plant shutdown sequence initiated September 30 was well controlled and uneventful until the unit auxiliary steam supply was transferred. The transfer to the steam supply line from adjacent coal Units 1 and 2 caused a slight steam pressure perturbation that resulted in a speed oscillation and overspeed trip of the "A" main feed pump (MFP). This initiated an emergency feedwater (EFW) actuation as expected for loss of both MFPs. The reactor was at 4% power and did not trip because the trip signal for loss of both MFPs is defeated below 10% power. The inspectors reviewed the details of the actuation and the response of the equipment. Operators responded appropriately, equipment responded as expected, and repairs were made to the MFP control valve seat to address pressure-induced speed changes. The inspectors did not identify any concerns with the plant response or actions of the licensee. Operators restored the B MFP and secured EFW within a few hours. The licensee reported the EFW actuation in Licensee Event Report (LER) 50-302/99-04. This LER is closed in Section O8.1.

O1.3 Rod Drop Testing

a. Inspection Scope (71707, 61726)

The inspectors observed portions of Surveillance Procedure (SP) 102, Control Rod Drop Time Tests, performed on October 1, 1999, for informational end-of-cycle (EOC) testing. Results were verified against TS limits.

b. Observations and Findings

The inspectors observed that reactor engineering coverage was thorough, with an engineer in the control room closely monitoring and discussing parameters with operators. Several activities were in progress in the control room which created a busy atmosphere. However, control room access was limited to essential personnel and senior management oversight was present. Operators were very focused on the tasks they were performing and extra personnel were staged to limit distractions by handling phone call, logs, and unrelated plant manipulations. Operators closely monitored appropriate plant parameters during each phase of the testing. Communications among the operators performing the test were complete and precise. The inspectors observed close adherence to SP-102 and related rod control operating procedures.

The performance of the test identified several slow rods (18 above 1.4 seconds and two above the TS upper limit of 1.66 seconds) and one rod in group 4 (4-3) that initially failed to fully insert. It stopped at approximately the 8% withdrawn position and slowly drifted inwards. The licensee dropped groups 4 and 5 again. Virtually all rod drop times improved. Rod 4-3 stopped again at 5% out, but drifted in on its own much quicker. The engineering analysis and resolution of the slow drop times are discussed in section E1.2 of this report.

c. Conclusions

Rod drop testing was effectively controlled. Senior management oversight was continual. Operators closely monitored plant instrumentation and distractions were limited during the testing. Communications were complete and precise.

O1.4 Reduced Inventory Controls

a. Inspection Scope (71707, TI 2515/142)

Inspectors reviewed preparations to monitor and control Reactor Coolant System (RCS) inventory during the outage. Particular focus was directed at control of reduced inventory conditions when RCS levels were below the reactor vessel flange. The inspectors reviewed RCS level indication issues associated with Inspector Follow-up Item (IFI) 50-302/97-11-01. Inspectors monitored and verified the alignment of RCS level indicators and observed controls during periods of reduced inventory. Also, per Temporary Inspection Instruction (TI) 2515/142, the inspectors reviewed the licensee's response to Generic Letter (GL) 98-02, Draindown During Shutdown and Common-Mode Failure.

b. Observations and Findings

The inspectors observed that the licensee had thoroughly prepared prior to the refueling outage for RCS draindown activities. Previous problems were addressed and industry experience was considered. A dedicated team of operators was assigned oversight for draindown activities. This team was sent on a benchmarking trip to another plant of similar design to review procedures and test a new initial RCS draindown methodology on the other plant simulator. The licensee implemented the new methodology which eliminated the use of a nitrogen over pressure on the steam generators and pressurizer. This over pressure had caused level oscillations in the past. A useful short-term instruction (STI) on expected level versus volume changes was developed with engineering input. The licensee also issued Operator Aid 99-011 which was a detailed and useful RCS component and elevation schematic.

The inspectors observed portions of the initial draindown and other RCS level changes throughout the outage and determined the licensee preparations were effective. Dedicated members of the aforementioned team were assigned for oversight and extra Operations personnel were available to plan and review ahead. The STI on expected level changes was used by operators who were very cognizant of expected plant response. During the previous outage, the licensee had experienced problems with alignment of tygon indicator valves and unexpected oscillations. The inspectors noted that the tygon valve alignment was well controlled and understood by all operators. Level indicators tracked well and were always within one or two inches of each other. Operators utilized several RCS level indication methods which included two tygon hose level indicators in the reactor building monitored via camera, permanently installed level transmitters indicating on the main control board, and spent fuel pool level which tracked RCS level when the two systems were connected. Operators also noted that reactor coolant inventory tracking system (RCITS) also tracked level change trends. Operators were knowledgeable of allowable deviations between each indicator. The inspectors determined that all of the deficiencies identified in IFI 50-302/97-11-01 had been adequately addressed. Section O8.2 of this report dispositions this item.

The inspectors did observe that operators' cognizance and tracking of RCS vent paths could be improved. Some operators were unaware of open vent path configurations or the status of local evolutions to vent portions of the RCS. Open vent paths were not tracked or included on shutdown safety function status boards. Vent paths were always adequate for plant conditions and Operations management was evaluating enhanced vent path tracking for future outages.

GL 98-02 described a problem with RCS inventory control at another plant and directed licensees evaluate their susceptibility to a similar problem and respond to the NRC in writing if susceptible. The licensee determined Crystal River 3 was not susceptible. Per TI 2515/142, the inspectors reviewed the licensee's associated analysis and evaluated the licensee's decay heat removal system configurations. The inspectors determined the licensee evaluation was appropriate. No concerns were noted with the analysis.

c. Conclusions

Preparations for refueling outage reactor coolant system inventory reductions were thorough. Dedicated oversight teams were established well in advance of the outage. Revised guidance, new operator aids, and a different initial draindown methodology were developed. The draindowns were closely supervised and operators were cognizant of all level indication instrument capability and readings. Level instrument performance was consistently accurate and stable, validating that previous concerns had been addressed. Temporary Instruction 2515/142 was completed to evaluate the licensee's analysis of Generic Letter 98-02 regarding reactor inventory control. The licensee's analysis was appropriate.

O1.5 Reactor Core Defuel and Reload Observations

c. Inspection Scope (71707)

The inspectors observed core defuel and reload activities, attended crew briefings, and interviewed several licensee and contractor personnel involved with these activities.

d. Observations and Findings

Observed communications between all personnel involved (contractors, Operations, Engineering, and Maintenance) were effective and consistently utilized good 3-way techniques. Minor problems were effectively resolved. Fuel movements were precisely controlled with specific move sheets and accurately tracked on a status tag board.

During control rod inspections, Reactor Engineering personnel conservatively revised move sheets to include placement of control rods into a test stand for eddy current testing. The inspectors considered this an example of good sensitivity to assuring proper placement of fuel assemblies and control rods.

However, during reinstallation of the reactor plenum assembly into the reactor vessel on November 1, 1999, difficulties were encountered. After indications of interference were noted, including fluctuating polar crane load cell readings, the licensee suspended the attempt and removed the plenum. Refueling personnel noted that the tripod lifting rig was attached incorrectly to the plenum. It was rotated 120 degrees so the prescribed attachment reference points on the tripod and plenum did not match. This caused the plenum to be slightly cocked.

The inspectors reviewed Refueling Procedure (FP) 501, Reactor Internals Removal and Replacement. FP-501, Steps 4.5.16 and 17 contained specific guidance for the attachment of the tripod to the plenum. The inspectors interviewed the fuel movement contract supervisor and foreman and licensee refueling senior reactor operators (RFSROs). The inspectors noted that the contract supervisor was unfamiliar with the cause or specific details of the event several days after it occurred. The lead contract floor worker responsible for the implementation of the pertinent FP-501 steps had been distracted by ongoing polar crane reliability problems and did not verify the correct attachment of the tripod to plenum. The steps also did not require a signature or check-off for completion so the foreman did not recognize his oversight. FP-501 states that RFSROs have ultimate responsibility for the overall evolution and RFSROs were described in pre-job briefing notes as in overall charge of the evolution. The inspectors determined that the RFSROs were focused primarily on reactor safety aspects of the evolution such as fuel canal levels and criticality control. They were not directly involved with the plenum rigging or the completion of specific FP-501 steps. Consequently, they were not an effective back-up to the contract foreman for FP-501 adherence. Additionally, the inspectors observed the licensee outage teams monitoring the contract performance were not functioning as direct oversight but as coordinators. Consequently, the error was not caught before the installation problems occurred.

The inspectors observed the licensee's subsequent video examination and verified the components were not damaged. TS 5.6.1.1 requires that written procedures be established, implemented, and maintained for the activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, such as refueling activities and core alterations. The failure to attach the tripod lifting rig to the plenum as directed by FP-501 was a violation of TS 5.6.1.1. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the NRC Enforcement Policy, and will be referenced as NCV 50-302/99-07-01, Reactor Plenum Rigged Improperly. This violation is in the licensee's corrective action system as Precursor Card (PC) 99-4142.

e. Conclusions

Fuel movement was precisely controlled. Communications were consistently effective and utilized 3-way techniques. A non-cited violation was identified for incorrect attachment of the reactor plenum to the tripod lifting device. Detailed procedural guidance for attaching the plenum to the tripod was not followed by contract refueling personnel and the error was not detected by licensee refueling senior operators.

O1.6 Improper Valve Manipulations Result in Spent Fuel Pool and Reactor Coolant System Level Decreases

a. Inspection Scope (71707)

The inspectors reviewed the circumstances and interviewed personnel involved with two instances of inadvertent water inventory displacement. One instance involved an operator manipulating the wrong valve and causing the spent fuel (SF) pool level to decrease. The second instance involved the performance of a valve lineup out of sequence which caused the RCS level to decrease.

b. Observations and Findings

On October 29, 1999, operators were restoring from draining the fuel transfer canal using SF pump 1B per operating procedure (OP) 406, SF Cooling System. An operator erroneously closed decay heat valve DHV-46 instead of spent fuel valve SFV-46, as required by OP-406, step 4.21.8. Both valves are connections between the SF system and the borated water storage tank (BWST), except SFV-46 connects the outlet of the SF pumps to the BWST and DHV-46 connects SF purification to the BWST. The low SF level alarm was received in the control room and the operators noticed BWST level increasing. SF pump 1A was secured. Another operator was dispatched and discovered SFV-46 five turns open from its closed position. The valve was closed, the pump restarted, and the BWST and SF pool level changes stopped. A total of 9980 gallons was transferred from the SF pool to the BWST and the SF pool temperature rose one degree Fahrenheit.

The licensee's investigation determined that the operator had a preconceived valve location in mind and had failed to reference the procedure or use proper self-checking tools. The inspectors independently observed the valve locations, reviewed the procedure, and reviewed personnel statements to verify the licensee conclusions. The inspectors determined the valves were clearly labeled and the procedure was clear. Some poor communication practices were noted in the personnel statement event descriptions. TS 5.6.1.1 requires that written procedures be established, implemented, and maintained for the activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, such as refueling and SF system operations. The failure to properly perform the steps directed in OP-406 was a violation of TS 5.6.1.1. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the NRC Enforcement Policy, and will be referenced as NCV 50-302/99-07-02, Failure to Properly Implement Procedure Results in Inadvertent Spent

Fuel Pool Level Decrease. This violation is in the licensee's corrective action system as PC 99-4073.

In the second instance, on November 1, 1999, approximately 1200 gallons of water was inadvertently transferred from the reactor vessel (RV) to the reactor building (RB) sump over a ten-minute period. Operators were performing a valve lineup to prepare for draining the RCS j-legs to remove steam generator (SG) nozzle dams in accordance with Enclosure 12, Step 2, of procedure OP-301, Operation of the RCS. The activity was moved ahead from the original scheduled sequence to prepare for draining the RCS to reduced inventory conditions when other critical path work was delayed. The lineup would align the RCS to be one valve away from draining the j-legs and was one of four tasks performed by a group of operators while in the RB. A pre-job brief was held between the two non-licensed operators performing the work and the senior reactor operator coordinating all RCS draindown evolutions. Control room personnel were not included in the brief. The non-licensed operators questioned why Step 1 of Enclosure 12, verifying other RCS valve positions, was not being performed. They were informed that some valves were SG "bowl drains" which did not impact this evolution and that the other valves were already known to be closed. This was an incorrect assumption because two main drain valves called out in Step 1 of Enclosure 12 were open due to minor SG nozzle dam leakage.

The control room was notified when the other three tasks were performed, but not for Enclosure 12 because no plant impact was expected. After the operators commenced the valve lineup, Health Physics personnel notified them that water was coming out of the floor drains. Control room operators detected the level changes by control room indication. Meanwhile, the operators in the RB had returned the manipulated valves to their previous position to stop the leak. RCS level dropped from 134.3 to 133.1 feet. Reduced inventory conditions, defined at 132 feet, were not entered.

The inspectors determined that Operations personnel had decided not to verify the valves in Step 1 of Enclosure 12 since they assumed they would not impact the valve alignment, even though a procedural note directed performing the valve lineup in the specified step sequence. Time pressure was a factor for the non-licensed operators since the valves to be manipulated were located in a high radiation area, thus limiting the operators' investigation of perceived flow indications when one valve was opened. Since the pre-job brief did not include control room personnel, they were unaware of the potential impacts on RCS level. TS 5.6.1.1 requires that written procedures be established, implemented, and maintained for the activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, including RCS operating procedures. The failure to properly perform the steps as directed in OP-301 was a violation of TS 5.6.1.1. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the NRC Enforcement Policy, and will be referenced as NCV 50-302/99-07-03, Failure to Follow Procedure Results in Inadvertent Draining of the Reactor Coolant System. This violation is in the licensee's corrective action system as PC 99-4143.

Corrective actions taken to address both of these events included an Operations stand-down and a lessons learned presentation by senior licensee management. Sensitivity to evolutions that have the potential to move water inventories was emphasized. Corrective actions implemented for the remainder of the outage included establishing an expectation that all valve positions would be verified by either of two methods: 1) hands on; or 2) confirmation that an administrative process is currently controlling the valve's position, and additional near-term training to discuss lessons learned from these events.

c. Conclusions

Two non-cited violations were identified for operator errors involving poor procedure adherence that resulted in inadvertent water level decreases in the spent fuel pool and reactor coolant system. Operators responded promptly to the events and terminated the draindowns prior to any impact on reactor coolant or spent fuel cooling systems. Failure to properly implement procedures was the primary cause of these two events, but contributing causes included deficiencies in communications, poor self-checking techniques, and an outage schedule change which moved up some draindown activities. Licensee investigations were thorough and corrective actions were prompt and appropriate.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) LER 50-302/99-04-00: Main Feedwater Pump Trip During Refueling Shutdown Results in Emergency Feedwater Actuation. As discussed in Section O1.2, the inspectors reviewed the details surrounding this actuation and did not identify any concerns or violations of regulatory requirements. The licensee addressed the actuation in their corrective action system under PC 99–3247. This LER is closed.

O8.2 (Closed) IFI 50-302/97-11-01: RCS Reduced Inventory Level Indication Problems. As discussed in Section O1.4, reliability of RCS level indicators was addressed by the licensee which was validated by successful tracking and indication of RCS level during refueling outage 11. As discussed in Section E8.2 of Inspection Report 50-302/99-01, this item also included the resolution of the RCS level indication system inconsistency with NRC Generic Letter 88-17 guidelines for two independent level instrument RCS fluid taps. The licensee only has a single fluid tap. The licensee was performing a design study to resolve the issue for the next refueling outage in the fall of 2001 which was being tracked in their corrective action system under PC 98-4167. The PC will adequately track resolution of the inconsistency. The lack of a second tap was previously determined not to be a violation of regulatory requirements. This item is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Observations

a. Inspection Scope (62707, 61726)

Using Inspection Procedures 62707 and 61726, the inspectors observed portions of several work requests and numerous surveillances and reviewed associated documentation, including the following significant activities:

- SP-406 Refueling Operations Daily Data Requirements
- SP-650 ASME Code Safety Valve Tests - Main Steam Safety Valve (MSSV)
- SP-456 Refueling Interval Equipment Response to an Engineered Safeguards Actuation System Test Signal
- SP-417 Refueling Interval Integrated Plant Response to an Engineered Safeguards Actuation
- SP-346 Containment Penetrations Weekly Check During Refueling Operations

b. Observations and Findings

All observed testing was performed methodically with good adherence to work instructions. Pre-job planning was notably thorough and pre-job briefings were consistently detailed. Personnel were knowledgeable of their assigned tasks. System engineers monitored job progress. Constant Inservice Testing engineering oversight was observed during MSSV testing. A minor problem with isolated MSSV test instrumentation was appropriately resolved during the testing and the inspectors verified MSSVs were adjusted and the test scope was appropriately expanded for out of tolerance results.

During the performance of SP-417, several A train components failed to properly respond. Building spray pump 1A failed to start, a reactor coolant pump seal controlled bleed off valve failed in mid-position, and a service water isolation valve position indicator linkage failed to develop a closed signal. The inspectors verified the licensee entered the discrepancies in the corrective action system (PC 99-4237), corrected the deficiencies, and properly retested the components. No further concerns were noted.

The inspectors reviewed the conduct of SP-346 for containment penetrations and identified a missing compensatory action sheet from the penetration logbook. However, detailed compensatory actions were not required for the noted penetration and the licensee quickly corrected the administrative deficiency. Overall, the inspectors considered the containment penetration controls adequate to assure proper and prompt containment closure based on independent walkdowns of open penetrations. The licensee conducted two unannounced drills which demonstrated prompt and adequate containment closure capability.

c. Conclusions

Maintenance activities were performed methodically and in accordance with procedures. Unexpected testing results were properly reviewed, corrected, and retested. Refueling outage containment penetration control was adequate. Unannounced drills demonstrated prompt and adequate containment closure capability.

M1.2 Emergent Maintenance Issues

a. Inspection Scope (62707)

The inspectors reviewed the circumstances and licensee response to two emergent issues in safety significant systems. The issues included tube leaks in nuclear services closed cycle cooling (SW) heat exchangers and a scope reduction in planned outage overhaul maintenance on the A emergency diesel generator (EDG).

b. Observations and Findings

The licensee discovered several leaks in three of the four SW heat exchangers (SWHE) following their return to service after draining the seawater side for maintenance. Recent SWHE operating experience has been very good, with no leaks, so the licensee issued PC 99-3946 to assess the potential for a common mode failure. The inspectors reviewed the licensee leak response plan and discussed the extent of condition and postulated causes with system engineers. The licensee found that the leaks were occurring due to localized pitting in the first 12 inches of the tube in the first pass of the dual pass SWHEs most likely from disruption of the protective passive layer on the tube by poor lay-up control and use of low oxygen content water to fill standby SWHEs. The leaks were repaired or plugged, with the resultant loss of SWHE capacity on ultimate heat sink capability accounted for in routine tube sheet blockage inspections. The licensee implemented close monitoring of the SW system for 60 days and appropriate long term repairs and corrective actions to the lay-up process were being considered. The licensee also verified that the small experienced leakage amounts remain bounded by a previous analysis limit of 0.4 gpm. The licensee response to the problems was considered appropriate.

Following extensive repairs due to problems encountered with the B EDG (discussed in Section E1.3), the licensee modified the planned scope of overhaul maintenance for the A EDG to prevent severely perturbing the outage schedule. The inspectors reviewed the original outage plan against the plan implemented by the licensee. Items that were deferred or not done were documented in a detailed Engineering Evaluation, EEM-99-059. Justifications included the ability to monitor parameters and work certain tasks online and included crediting previous performance of similar tasks. The inspectors noted that a few of the deferred items could not be worked online and could not be monitored for degradation before a failure would occur. However, the licensee's justification for these items noted that licensee and industry operating experience indicated the items were extremely reliable, which the inspectors determined was acceptable. The licensee also researched commitments associated with EDG maintenance and did not identify any that conflicted with their course of action. The inspectors found the licensee decision methodology to be sound and did not identify any violations of regulatory requirements.

c. Conclusions

The licensee responded adequately to two emergent maintenance issues during the refueling outage. Leakage in SW heat exchangers was appropriately dispositioned and a scope reduction to planned A EDG maintenance was adequately justified.

M1.3 Control Complex Habitability Envelope (CCHE) Integrated Leak Rate Testing

a. Inspection Scope (61726)

Technical Specifications Surveillance Requirement 3.7.12.4 requires that a CCHE integrated air leakage test be performed every 24 months under simulated worst case conditions to ensure operator protection following an accident. The inspectors observed the test setup and initial gathering of air samples and data, reviewed the procedure and final analysis results, and interviewed involved personnel.

b. Observations and Findings

The leakage test, conducted on September 21 through 24, 1999, consisted of injecting a known initial concentration of sulfur hexafluoride into the closed CCHE, waiting several hours, then measuring the remaining concentration. To obtain the worst case post-accident ventilation system induced differential pressure (dP) across the CCHE, the control room emergency ventilation system was placed into the emergency recirculation mode of operation, all fans that normally supply air into the auxiliary building (AB) were secured, and two AB exhaust fans were in operation. Uniform CCHE mixing was established by portable fans and opening of all interior doors. The inspectors verified a roving fire watch was established and security personnel were posted at keycarded doors as appropriate compensatory measures.

The inspectors attended the pre-job brief and noted a few minor problems during the test setup, which were adequately resolved prior to test commencement. Contractor personnel conducting the test were very knowledgeable and experienced in conducting similar tests at other facilities. Analysis of the test data revealed that the inleakage results were bounded by the previous (1997) test results and therefore, calculation of an increased CCHE breach margin was not necessary. The inspectors reviewed the final results and concluded that the current breach margin of 35.5 square inches remains valid.

c. Conclusions

The CCHE integrated leak test was conducted methodically and test results were satisfactory.

M1.4 Reactor Building (RB) Tour Observations (62707, 71707)

Throughout the refuel outage the inspectors toured the RB to observe outage work and assess building and equipment condition. No significant equipment problems were noted beyond those the licensee had already identified. Although the inspectors questioned the appropriateness of placing outage test equipment near some reactor vessel level indication components, the licensee ensured the equipment was moved and had not impacted the instruments prior to the next level change. The inspectors noted throughout that Health Physics personnel were readily recognizable and accessible. The inspectors initially observed several deficiencies with implementation of the licensee safety-at-heights program. Licensee Safety Evaluators had also observed similar problems and initiated corrective actions Safety-at-heights program implementation improved by outage end. Overall, the inspectors concluded that RB work activities were adequately coordinated and conducted.

M1.5 Inservice Inspection

a. Inspection Scope (73753)

The inspectors observed in-process work activities and reviewed selected procedures and records to evaluate implementation of the licensee's inservice inspection (ISI) program. The observations, procedures and records were compared to the Technical Specifications (TS), the Final Safety Analysis Report (FSAR), and the applicable code (ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, with no Addenda). Portions of the following in-process ISI nondestructive examinations (NDE) and records were observed:

- Liquid Penetrant (PT) of Weld MU-273A on Isometric Drawing (ISO) Sk-5.1
- Magnetic Particle (MT) of Welds FW-89FF and FW-173B ISO Sk-107.2
- Visual (VT) examination of the following Pipe Supports:
  - Support SWH-33 on ISO SKH-220.1
  - Support SWR-393 on ISO SKH-222.1
  - Support EFH-84 on ISO SKH-233.1
  - Support EFH-543 on ISO SKH-108.2
  - Support RWH-75 on ISO SKH-215.1
- Ultrasonic (UT) examination of the following welds:
  - Welds FW-89FF and FW-173B ISO Sk-107.2
  - Weld MK8 to 6 (Surge Nozzle) on ISO Sk-1AC.8

- Eddy Current (ET) examination of Steam Generator (SG) Tubes

  The inspectors discussed the inspection program for the current outage with licensee personnel, observed ET data acquisition for a sample of tubes for SGs A and B, and observed resolution of inspection results for Calibration Group SG3BCCAL00027 for SGB.

The inspectors also reviewed the ASME Section XI repair and replacement (R&R) documentation for the following replacements associated with modifications of the High Pressure Injection (HPI) and Emergency Feedwater systems and compared the documentation with the requirements of ASME Section XI:

Work Request (WR) NU 0358586 - Fabrication of Emergency Feedwater Pump-3 Diesel Fuel Tank DFT-4 Vent

WR NU 0360474 - Installation of new MUV-27 Valve and subassembly

The documentation reviewed included: ASME Section XI Repair/Replacement Program Evaluations, Process Travelers, Weld Travelers and/or Weld Control Records. In addition, completed radiographic (RT) inspection film were review for HPI welds Mu-85-809, 815, 817, 830, 836, 895, 848, 870, and 879.

b. Observations and Findings

The inspectors found that, in general: detailed instructions and procedures were in place and were being followed by knowledgeable and qualified inspection personnel; approved and calibrated inspection equipment was being used; inspections were being performed in accordance with applicable Code requirements; program changes, including appropriate approval of code relief requests, were being controlled; and examination results were being properly evaluated and corrective actions taken as required. Plans and schedules for the current inspection period were in accordance with the approved ISI program. R&R activities complied with ASME Section XI requirements.

During observation of NDE examinations, the inspectors noted that control procedures did not provide details for determination and documentation of extent of coverage for limited examinations. ASME Section XI, as clarified by Code Case 460, requires that each weld be inspected “essentially 100%”, where “essentially 100%” means greater than 90% of the required surface or volume. This interpretation is further clarified in 10 CFR 50.55a(g)(6)(ii)(A)(2) and NRC Information Notice 98-42. Although not proceduralized, the extent of coverage was being documented by the licensee where examination coverage was limited. After questions by the inspectors, the licensee agreed that details of documenting limited examinations should be proceduralized and stated that procedures would be revised to provide details.

The ET program included a 100% BOBBIN Probe inspection of all tubes in both SGs. In addition, the program included inspection of a large sample of tubes, tube sleeves, and tube plugs using the Mechanized Rotating Pancake Coil (MRPC) and Plus Point inspection techniques. These samples were selected based on known problem areas. As of October 15, 1999, the BOBBIN inspection had been completed for approximately 20% of SG A tubes and 10% of SG B tubes. In addition, a significant portion of the MRPC and Plus Point inspections had been completed. In SG A, using the Plus Point Probe, 10 of the 528 lower tube sheet secondary face ding/dent population sample were found to contain defect indications. This condition was reported to NRC on October 15, 1999, in accordance with Technical Specification requirements. The sample was expanded to include 100% of the tubes in the ding/dent region for SG A and 6% of the tubes for SG B.

c. Conclusions

ISI activities were being performed in accordance with ASME Code and licensee requirements.

M1.6 Flow Accelerated Corrosion (FAC) (49001)

The inspectors reviewed the scope of FAC inspections for this outage and observed portions of UT thickness measurements and/or grid layout for several components including 122-02E and 107-51E.

Compliance with program procedure requirements for these examinations was verified. The inspectors found that a detailed FAC program was in place and implemented in accordance with procedural requirements.

III. Engineering

E1 Conduct of Engineering

E1.1 High Pressure Injection/Low Pressure Injection Post-modification Flow Testing

a. Inspection Scope (37551)

The inspectors observed significant portions of post-modification flow testing on the high pressure injection portion (HPI) of the makeup and purification system (MU) and the low pressure injection (LPI) portion of the decay heat removal system (DH). HPI modifications included new throttle valves and crossover lines and a new automatic closure valve for the normal makeup and seal injection line. LPI modifications included new main flow control valves. The inspectors attended pre-job briefings, reviewed IPTE briefing materials, observed testing in progress, validated test results, and monitored the resolution of emergent problems.

b. Observations and Findings

HPI flow testing was done per Modification Approval Record (MAR) 97-02-12-01 Functional Test Procedure (TP) 1 with the plant in Mode 6 from October 27 to 29, 1999. HPI TP-1 was done in two phases: drawing suction from the DH pump discharge in piggyback mode; and drawing suction from the borated water storage tank (BWST). The test was written to also fulfill normal refueling outage MU surveillance tests which the inspectors verified were appropriately met. The inspectors verified the scope of the test fully encompassed the modifications and tested appropriate combinations of equipment and monitored appropriate parameters. Test procedures and briefing materials were fully developed, reflecting extensive preparation prior to the outage. Numerous complex test prerequisites were tracked via a detailed flow chart and extensive management and test personnel support was present at all times. Pre-job briefings were very thorough. Conduct of the complex testing was methodical and coordinated effectively. The inspectors reviewed MU and DH system configurations in detail to verify critical outage safety functions such as decay heat removal were not adversely impacted. No problems were identified.

Detailed hydraulic modeling of the modifications was done to develop expected system response. The detailed expectations enabled prompt detection of two test problems (position of normal makeup control valve and discharge stop check valves) which were subsequently adequately addressed. The inspectors reviewed the licensee's analysis and verified that these problems did not impact the test results. LPI Testing was done per MAR 98-12-04-01 Functional Test Procedure (TP) 1 separately for each DH system train. B DH was tested on October 22, 1999, and A DH on October 29, 1999. The testing required throttling flow on the running DH train, rendering it inoperable per Technical Specifications (TS). The inspectors verified the licensee complied with appropriate TS requirements and that adequate DH system cooling of the reactor was maintained. Both phases of the LPI testing were successful. Problem with tolerances on the throttle valve stem position indicators were addressed in the corrective action program. No other concerns were noted.

The inspectors reviewed completed test procedures, test director logs, and test exception reports (TER). One discrepancy was noted with a TER on HPI TP-2 that added a test step to fix a previous revision error. Overall, changes to the tests were well controlled.

HPI electrical testing was done per MAR 97-02-12-02 TP-1 (Engineered Safeguards Functional Test) and TP-2 (Control Logic Functional Test). The inspectors also observed some portions of this testing and noted the testing identified several discrepancies with incorrectly wired control functions. These were appropriately corrected and retested.

c. <u>Conclusions</u>

Post-modification testing of major high pressure injection and low pressure injection system modifications was effective. Functional tests were accurately detailed and reflected extensive preparatory work. Pre-job briefings were very thorough, management oversight was continuous, and test performance was methodical. Results were satisfactory and unexpected problems were appropriately dispositioned. Testing impact on critical shutdown plant safety functions was closely monitored and no problems occurred.

E1.2 Resolution of Slow Rod Drop Times and Fuel Assembly Bowing Issues

a. Inspection Scope (37551)

As discussed in Section O1.2, the licensee performed end-of-cycle (EOC) drop testing on all control rods on October 1, 1999. The inspectors reviewed the licensee analysis and corrective actions for the rod that stopped and several rods that exhibited slow drop times.

b. Observations and Findings

The licensee performed a detailed analysis of the rod drop time data, analyzing velocity profiles, historical results, and system configuration. The licensee identified that two conditions, old thermal barrier (TB) design and high fuel assembly burn up, were associated with slow control rod drive (CRD) drop times. Ten CRD assemblies were identified for further action based on the analysis. However, after evaluating dose rates on the reactor head and results of removed fuel assembly (FA) video inspection that showed bowing in older FAs, the licensee reduced the repair scope to five CRD assemblies. The licensee decision process was logical and justifiable. The group of five included the one slow CRD with a new TB design, the CRD that stopped, the two CRDs that exceeded the TS drop time criteria, and the CRD with the remaining worst rod drop time. In 1996, the licensee had upgraded 22 of the 60 CRD TBs to a new design to address generic industry problems with slow rod drops due to crud deposition causing stuck TB ball check valves.

The licensee inspected the five CRD TBs and found several stuck ball checks. The four old design TBs were upgraded to the new design. The fifth (new design TB) CRD was replaced with an entire new CRD assembly due to corrosion noted on the guide tube assembly. The inspectors also observed the licensee perform 100% CRD eddy current testing and drag force measurements when the CRDs were removed in the spent fuel pool. No problems were noted on the eddy current testing while several assemblies were observed to have slightly higher than expected drag forces.

The licensee integrated the results of these inspections back into their original detailed analysis and concluded that the slow drop times were due to a combination of binding from FA bowing and TB induced hydraulic drag. The inspectors independently verified the licensee data and root cause analysis. Large amounts of data had been very effectively integrated to form the basis for the licensee conclusion. No problems were noted with the analysis or corrective actions. FA bowing had been observed at other similar plants so corrective actions were developed in conjunction with the fuel and reactor design vendor. Actions included shuffling FAs to opposite core quadrants to evenly distribute flux exposures and resetting FA cruciform hold-down springs. The reset involved plastically deforming the spring to lower the resultant force exerted and was done on the 60 FAs with control rods, the 48 FAs with burnable poison assemblies, and on the 24 new FAs. The new cruciform spring design exerted higher hold down forces than the previous helical spring design and was postulated to exacerbate FA bowing. The inspectors observed the spring sets in the spent fuel pool and reviewed the associated safety assessment (SA). Although the SA did not specifically discuss the consequences of a spring failure, the licencee had thoroughly evaluated the spring set change and had insisted on documented data from the vendor to support their conclusion. The licensee also had the SA approved by the onsite review committee and reviewed by an independent third party consultant. No other problems were noted with the SA or spring set performance.

The inspectors determined the licensee's corrective actions were balanced and appropriate. The inspectors noted that even with the slow drop times, the licensee retained large margins in the rod drop safety analysis which was based on overall group average drop times. Consequently, the slow rods had minimal safety significance. No further issues were noted pending successful beginning-of-cycle drop tests during plant startup.

c. Conclusions

The licensee thoroughly analyzed large amounts of control rod assembly data to address problems identified by end-of-cycle rod drop testing, including several slow drop times. Fuel assembly bowing and thermal barrier induced hydraulic drag were attributed as causes. Corrective actions, including resetting hold-down springs and replacing thermal barriers, were appropriate.

E1.3 Failure of Emergency Diesel Generator 1B Radiator Fan Drive shaft

a. Inspection Scope (37551)

Following successful overspeed testing on October 16, 1999, the drive shaft for the 1B emergency diesel generator (EDG) radiator fan was found completely failed in one location and damaged in two others. The licensee initiated a root cause investigation. The inspectors observed the failed components and independently verified the licensee investigation results.

b. Observations and Findings

Mechanics inspecting the EDG radiator housing after the test discovered a broken cotter pin piece on the floor. Pursuing this discovery, the licensee found the EDG radiator drive shaft completely sheared between the engine power take-off shaft and the radiator clutch. Additionally, two yoke friction-fit joints on the shaft from the right angle gear box up to the fan were found to have slipped, resulting in broken keyway assemblies. The pursuit of the broken cotter pin was noteworthy because it identified an inoperable diesel that would not otherwise have been detected for several days until the next routine surveillance test.

The overspeed (OS) set point for the 1B EDG had been adjusted upwards from an acceptable range of 990 - 1053 rpm to 1035 - 1053 rpm after a reevaluation of a 1991 service information letter (SIL) from the EDG vendor. This required an adjustment of the EDG OS mechanism which previously had been set at approximately 1028 rpm.

Consequently, the October 16 OS testing was done at slightly higher speeds than previously. Following the failure, the licensee assembled a dedicated root cause team and obtained vendor and expert failure analysis assistance. The team investigation concluded the cause of the failure was a slippage in the upper yoke assembly that allowed the fan assembly to slow. When this assembly seized shortly thereafter, the rapid acceleration of the fan assembly transmitted a torsional force along the fan drive train adequate to break the other yoke assembly and fracture the clutch shaft. Although the licensee was unable to identify the specific cause for the upper yoke failure, they did identify several fabrication deficiencies for the yoke friction fitting which were addressed in their corrective actions.

The inspectors inspected the failed components, reviewed the EDG design ratings and documentation, and reviewed the licensee root cause report. The physical evidence supported the licensee root cause determination. Review of the design documentation validated the licensee determination that portions of the fan drive assembly would operate above design horsepower ratings and in design safety factor ranges during the higher overspeed testing. However, the inspectors confirmed that ultimate ratings of the components were more than adequate for the transmitted forces and were an appropriate design. Although the licensee could not determine the cause of the initial yoke slippage, it was apparent that the yoke design would have been adequate for OS loads if assembled properly. The licensee corrective actions implemented to repair the EDG thoroughly addressed each plausible cause. The inspectors determined the licensee investigation was very thorough. Appropriate long-term corrective actions were identified to analyze the metallurgy of the failed parts, periodically inspect the EDG radiator drivetrain connections, and to further evaluate the capability and design of the drivetrain. The licensee also appropriately considered the A EDG and implemented corrective actions on that fan assembly at the first available opportunity.

The inspectors closely reviewed the vendor SIL and the overspeed setpoint change documentation and safety assessment. In 1991, a licensee engineering evaluation had determined that the SIL did not apply to their EDG governor configuration because the licensee had not experienced the problems described in the SIL. During planning for the current refueling outage, system engineers decided the SIL was applicable and implemented it. The inspectors determined that the licensee action to raise the EDG OS setpoint per the SIL was fully evaluated and per the vendor recommendation. The inspectors did not identify any concerns with the licensee decision. However, since the subsequent licensee investigation determined that the fan drive system operates in the design safety factor range above 990 rpm, a corrective action conservatively lowered the EDG OS set point range to 1005 to 1025 to limit challenges to the system. The inspectors reviewed the safety assessment for lowering the set point and verified it was adequate and margin existed to satisfy requirements for largest load rejection capability. All of the various set points implemented by the licensee have fallen within the historical range of 990 - 1053 rpm established by the EDG vendor at initial construction.

The inspectors did not identify any concerns with licensee corrective actions that had been taken or planned. Although the specific cause of the yoke connection failure has yet to be determined, the licensee investigation was thorough and corrective actions were comprehensive and appropriate.

c. Conclusions

Two failed yoke assemblies and a sheared radiator clutch drive shaft on the B emergency diesel radiator fan shaft were found by alert mechanics following overspeed testing. Fabrication problems with the yoke assembly were noted and addressed in corrective repair actions, but the initiating cause of the yoke failure was undetermined. Inspectors verified the physical evidence supported the licensee determination and noted the design ratings of the radiator drive train were adequate. Repair actions and long-term corrective actions were comprehensive and appropriate.

E1.4 Review of Design Modifications

a. Inspection Scope (37550 and 37001)

The inspectors evaluated the engineering activities associated with design modifications being implemented on the emergency feedwater (EFW), high pressure injection (HPI), and low pressure injection (LPI) systems. The activities were evaluated by directly assessing the activities and/or their products (documents, functional tests, and installed equipment). The evaluation was a continuation of a previous review documented in Inspection Report (IR) 50-302/99-06.

b. Observations and Findings

Description of Modifications

The EFW modification was a significant addition to the plant capabilities. It involved installation of a safety related diesel-driven EFW pump; controls and instrumentation; batteries, fuel tank, and starting air for the diesel; a building to house the pump and support equipment; and additional piping and valves. The LPI modification primarily involved installation of two new globe flow control valves to provide improved throttling capabilities for the piggyback mode of operation (LPI to HPI). The modification also added five vent valves for improved system venting.The HPI modification involved installation of additional piping and valves to reduce operator burden and enhance the operator's capabilities to manage a small break loss of coolant accident.

Mechanical

The inspectors reviewed the LPI, HPI, and portions of the EFW modification packages. Consistent with their previous review (IR 50-302/99-06), the inspectors found that the modifications and testing specified in the packages would fulfill the stated design objectives when satisfactorily implemented. The packages were complete and included satisfactory 10 CFR 50.59 evaluations. Sampled design information was accurate. The packages were generally of good quality, although some typographical errors were noted.

The inspectors reviewed the following design calculations for the modifications: thrust calculations for motor operated valves EFV-33 (EFW isolation) and MUV-596 (HPI pump discharge isolation), diesel fuel oil tank sizing calculation, and EFW system design pressure calculation. The inspectors verified that appropriate calculation methods and inputs were used and verified the accuracy of selected computations.

The inspectors reviewed the procurement records for a sample of the HPI and LPI modification hardware and equipment. The sample included motor-, air-, and manually-operated valves; piping; and pipe fittings. The inspectors found that appropriate codes and engineering requirements were specified in the procurement, deviations were satisfactorily dispositioned, and vendor reports supported compliance with the procurement requirements except where deviations were approved.

The inspectors walked down sections of the modifications. While some work was still in-progress, most of the major component weldments had been completed. Observed weld quality and fit-up were satisfactory. Piping installation met plan configuration and design requirements. The inspectors noted that the controls features were being properly installed in accordance with the installation plan. Quality Control personnel were observed to be satisfactorily marking several HPI piping welds for repair. The inspectors selected a large HPI component piping installation with a heavily loaded support configuration and verified that the stress analysis demonstrated that the loading was acceptable.

The inspectors reviewed the results of static diagnostic tests performed on motor-operated valves EFV-33 and MUV-596. The testing demonstrated satisfactory performance of the installed valves under static conditions. The inspectors verified that dynamic testing had been completed on MUV-596 and that dynamic testing was scheduled for EFV-33 to confirm their capabilities under design conditions.

Fire protection was particularly significant for the EFW modification because of the diesel fuel used, the possible presence of hydrogen from batteries, various potential sources of sparks, and the operating temperatures of the equipment. The inspectors observed that ventilation and sprinkler systems were being appropriately installed and exhibited good workmanship. At the time of the inspection, the licensee had not completed installation of the fire protection equipment, calculations, flow tests, and model comparisons. These were due to be completed after the EFW system was in placed into service. Planned and reviewed compensatory measures were to be instituted until their completion.

The inspectors discussed the HPI changes with the operator at the controls and the senior reactor operator on shift and found that they had received training on the new system changes and understood the effects on operation. The inspectors also reviewed the training records and verified that they documented acceptable training of the operations staff. The inspectors observed evidence in field changes that the operations staff had interacted with the modifications personnel on details of the installations.The inspectors verified that the operating procedures identified for the HPI and LPI modification implementation were appropriately tracked and scheduled for issue prior to the operational modes for their use. A sample of the procedures was reviewed to ensure that the appropriate components of the modifications were included.

The inspectors observed licensee personnel from different departments and the primary construction vendor evaluate the HPI flow test procedure using the control room simulator. The personnel involved contributed to clarification of the complex procedure. During subsequent test performance, as discussed in Section E1.1, only minor issues were identified that required procedure changes. These changes did not alter the test intent or impact the test results. The inspectors reviewed the completed HPI full flow test procedure and verified that the test results were correctly captured, evaluated and dispositioned. The test results supported satisfactory functional performance of the modification.

The inspectors observed portions of the recirculation flow test performed on the EFW modification and reviewed the recorded results. Moderate vibration was observed on the speed increaser, discharge piping, and cooling fan components. Otherwise the new pump and diesel driver operated well. Vibration had been noted by the licensee in previous tests and documented for disposition in Nonconformance Reports 99-00-528 (pillow block bearing and right angle fan drive) and 99-00-529 (speed increaser). The inspectors found that the test procedure was thorough, well-executed, and that appropriate data was recorded and evaluated. Exclusive of the vibration problems, the test results supported satisfactory functional performance of the modification. The vibration problems were still under evaluation at the conclusion of the inspection.

<u>Electrical</u>

The inspectors found that the battery sizing calculation for the new EFW pump (EFW-3) battery used a methodology recommended in IEEE standards, and incorporated conservative design inputs. The inspectors confirmed that the correct discharge characteristics were used in the calculation by referring to the manufacturer's data sheets. The inspectors reviewed the factory and on-site battery capacity test results for the EFW-3 battery. The results indicated that the battery had 104 percent of rated capacity and performed well when subjected to the design basis load profile. The battery float voltage and equalize voltage were within the manufacturer's recommended range and suitable for the loads.

Three valves in the HPI system (MUV-18, -27, and -596) were reviewed by the inspectors in relation to NRC Information Notice 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire" and were found to be designed against the problem described in the notice.

The inspectors' review of Generic Letter 96-01, "Testing of Safety-Related Logic Circuits" considerations for selected valves added to the HPI system indicated that the valves were added to the appropriate technical specification surveillance procedures and would be appropriately tested.

The control circuits for the new LPI flow control valves and heat exchanger outlet valves were also reviewed, and the circuitry was found to correctly implement the intended logic.

c. Conclusions

The engineering organization was effective in designing and implementing EFW, HPI, and LPI modifications. The licensee's modification packages were generally complete, accurate, and of good quality. Installation and testing were satisfactory, and problems were being appropriately identified and resolved.

E2 Engineering Support of Facilities and Equipment

E2.1 Once-Through Steam Generator (OTSG) Outage Inspections (37551)

The inspectors closely monitored the results of the licensee 100% OTSG Inspections against historical inspection results. To address historical problems with minor leakage at tubesheet boundaries, the licensee performed a bubble test on both OTSG upper tubesheets. The test identified several minor tube leaks on both OTSGs and three notable leaking tubes in the B OTSG which they factored into repair and inspection pans. The licensee also performed detailed inspections to evaluate tube end cracks (TEC) found in a previous outage and previously considered as outside-the-pressure-boundary indications. TECs were also the predominate cause of the bubble test indications. Recent license changes allowed leaving specific TECs in-service and re-rolling repairs of tubes with TECs. The licensee utilized these changes extensively and appropriately to resolve the TECs. Although several targeted inspection scope population results were categorized as Category C per Technical Specification 5.6.2.10, these were generally expected results for known historical problems. Overall inspection results were very good and the licensee OTSGs were in very good condition with only 52 tubes in A OTSG and 69 in B OTSG plugged during this outage. The inspectors had no concerns with the OTSG inspection findings and observed good oversight by the Engineering Programs group.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Inspection Followup Item (IFI) 97-17-03: Review of Cable Ampacity Issue. The IFI was closed based on review of design details of the identified problem cable tray sections by the inspector and a physical examination of potential problem cables by the license using the scientifically developed "indentor" methodology. For nearly all the cable tray sections having potential ampacity problems, the inspector evaluated the number and size of cables in the section and the load currents. The inspector independently determined ampacity using Insulated Cable Engineers Association standard P-46-426, "Power-Cable Ampacities," plus the appropriate derate factor for fire barrier material. Other factors considered were total watts dissipated, diversity among the installed cables and ambient temperature. The derate factors for fire barriers had been determined through testing conducted by Underwriters Laboratories. The test results were reviewed by NRC Headquarters and Sandia National Laboratories. The insitu cable inspections found that the cables were in good condition. The inspectors walked down the diesel generator leads in the control complex where they were installed in trays enclosed by fire barrier material to ascertain the as-built configuration. The overall condition of the questioned cables was good (i.e. no evidence of heat induced degradation). In nearly all the cases reviewed, traditional methods of determining ampacity plus prudent engineering judgement indicated that the ampacities were adequate for the actual configurations. The diesel generator leads had adequate ampacity. In a very limited number of cases the analysis may need to be supplemented by periodic examinations of the type discussed above to show continued operability. The inspectors concluded that the licensee's programs would ensure this action if necessary.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Conduct of Refueling Outage Radiological Protection Controls (83750)

a. Inspection Scope

During Radiological Control Area (RCA) tours, the inspectors observed work activities in progress, discussed procedural and Radiation Work Permit (RWP) requirements with workers, and verified selected radiation survey results. Radiological controls and housekeeping practices for the auxiliary building, reactor building, and turbine building, and for outside RCA locations used for radioactive material control/storage and for solid radioactive waste processing and storage were observed. Dosimetry use, air-sampling, area postings, container labels, housekeeping, and controls for high radiation areas (HRA), locked-HRAs, and very-HRAs were reviewed and evaluated.

The implementation and results of radiation protection activities were compared against applicable sections of the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), and 10 CFR Part 20.

b. Observations and Findings

Excluding several observations of poor health physics practices, workers and Health Physics (HP) technicians were knowledgeable of radiological conditions and RWP requirements. The poor radiological practices were identified for dosimetry use by personnel in turbine building RCAs, contaminated area work practices, visibility of reactor building high and locked high-radiation area postings, and limited communication with the HP staff regarding repositioning of spent fuel pool tele-dosimetry and removing lead shielding from boxes housing control rod drive mechanisms. The observed poor practices did not resulted in any significant unanticipated increase in dose to workers.

Excluding internal exposure assessments, radiological controls for outage activities, including HP coverage, required protective clothing, personnel dosimetry use, and air sampling were established and implemented in accordance with established procedures.

High radiation areas and locked-HRAs were controlled appropriately. As of October 21, 1999, occupational doses resulting from worker exposure to external and internal radioactive sources were below regulatory limits with the documented maximum 1999 year-to-date (99YTD) worker total effective dose equivalent (TEDE) slightly exceeding 2000 millirem (mrem). The licensee reported an unexpected increase in personnel contamination events (PCE) involving both potential external and internal exposure. For the current outage, more than six PCE events per 5000 RCA-hours worked exceeded the licensee's established 99YTD target of one event per 5000 RCA-hours. Inspector review of data associated with approximately 15 skin contamination PCEs during the current outage verified the dose evaluations were conducted in accordance with approved procedures, were technically correct based on conservative exposure times and source term determinations, and the calculated shallow doses were below regulatory limits.

Concerns regarding the adequacy of licensee evaluations for potential radionuclide intakes by workers were identified. From review of selected whole-body counting (WBC) analysis records, the inspectors noted that the licensee failed to input accurate dates and times, and complete timely follow-up analysis for several unanticipated internal exposure events. For example, an individual was initially analyzed as having a committed effective dose equivalent (CEDE) of approximately 24 mrem due to a radionuclide intake of 2.035 E+3 nanocuries of mixed radionuclides on October 2, 1999. The initial WBC data did not include documentation that an evaluation was conducted to determine if the material was internal or external contamination, or if an evaluation of potential intake of alpha-emitting radionuclides was considered. A follow-up WBC analysis as recommended in Health Physics Procedure (HPP) - 320, Whole Body Counting System Operation, Revision 12, was not conducted until October 11, 1999. Further, the October 11, 1999 follow-up WBC analysis used an incorrect intake date of October 11, rather than the actual October 2, 1999 event date. A re-analysis of the follow-up WBC data using the correct intake event date and time resulted in a calculated CEDE dose of 0.4 mrem. Subsequent licensee review of approximately 200 WBC analyses conducted during Refueling Outage 11 identified that approximately 20 percent of the intake evaluations conducted had used the incorrect event dates and times for internal dose calculations. The inspectors noted that 10 CFR 20.1501(a) requires the licensee to make surveys necessary to comply with the regulations in this part. The surveys are to be reasonable to evaluate the concentrations or quantities and the potential radiological hazards that could be present. The failure to conduct accurate and timely analyses (surveys) of potential radionuclide intakes by workers to evaluate the hazards present was identified as a violation of 10 CFR Part 20.1501(a) requirements. Based on re-analysis of the worst-case conditions, input of the incorrect event dates were found to not significantly increase the individuals CEDEs and did not result in any of the affected workers exceeding administrative or regulatory dose limits. No concerns were identified for the licensee's immediate corrective actions to improve evaluation of intake events. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, and will be referenced as NCV 50-302/ 99-07-04: Failure to Conduct Timely and Accurate Analysis of Potential Radionuclide Intakes by Workers. This issue was entered into the licensee's corrective action process as PC 99-3844.

c. Conclusions

Overall, radiological controls were maintained and implemented in accordance with UFSAR, TS, license conditions, and 10 CFR Part 20 requirements. Excluding workers' internal exposures, licensee dose assessments associated with unanticipated contamination events were adequate. Several examples of poor radiological practices were identified regarding dosimetry use by personnel, contaminated area work practices, visibility of reactor building radiation postings, and limited worker communication with the HP staff. A non-cited violation was identified for failure to conduct accurate and timely evaluations of worker exposure from potential radioactive material intakes. Occupational worker doses were determined to be within administrative and regulatory limits.

R1.2 As Low As Reasonably Achievable (ALARA) Program Implementation (83750)

a. Inspection Scope

The licensee's As Low As Reasonably Achievable (ALARA) program implementation for the Cycle 11 refueling outage (RFO 11) was reviewed. The review included shutdown cooling and reactor coolant system cleanup initiatives for radioactive source-term reduction. In addition, exposure reduction initiatives, work planning, dose estimates, and resultant cumulative doses were evaluated for the following high dose rate or high cumulative exposure tasks: removal and installation of primary manways; nozzle dam installation/removal; eddy current examinations; reactor head disassembly; and scaffolding construction.

The ALARA program implementation and results were evaluated against applicable sections of 10 CFR Part 20 and approved procedures.

b. Observations and Findings

Licensee management had implemented dose reduction initiatives in accordance with their ALARA program guidance. For calender year 1999, the licensee budgeted approximately 195 person-rem for all site operations, with 152 person-rem allocated to RFO 11 outage activities. As of October 21, 1999, RFO 11 dose expenditure was approximately 117 person-rem and above the target value of approximately 96 person-rem.

Based on limited documentation, the inspectors verified that specific dose budgets and exposure reduction initiatives were implemented for selected high dose and dose rate tasks. Licensee representatives calculated that the initial shutdown cooling and chemically induced crud burst evolution followed by extended cleanup removed approximately 1005 curies of cobalt-58 and reduced reactor coolant system activity from 2.85 to 0.147 microcuries per cubic centimeter. Excluding eddy current testing and scaffolding evolutions, dose budgets for the reviewed evolutions had been revised upwards. Only eddy current testing was on target with the actual cumulative dose for scaffolding evolutions exceeding the budget estimate. Documented factors contributing to the increased cumulative dose associated with selected tasks included elevated dose rates associated with steam generator work evolutions, increased numbers of new personnel on scaffolding crews, and emergent work activities associated with control rod drive work activities. The inspectors verified that unexpected elevated dose rates, RWP-hours, and unexpectedly elevated person-rem were identified and documented for review, evaluation and development of lessons learned by responsible ALARA program personnel and management.

c. Conclusions

ALARA program activities and initiatives for the refueling outage were conducted in accordance with approved procedures with outage cumulative dose expenditure revised upward from original estimates due to elevated dose rates, inexperienced workers, and emergent work activities.

R8. Miscellaneous Radiation Protection and Chemistry Issues (83750)

R8.1 (Closed) Unresolved Item (URI) 50-302/99-05-01: Review Adequacy of Calibration Activities and Implementation of Verification Determinations for General Area Radiation Monitors. The licensee's actions regarding the adequacy of general area radiation monitor calibrations and performance of verification tests were reviewed and discussed. The inspectors verified completion of licensee review and actions regarding this issue. No violations were identified and this item is closed.

V. Management Meeting

X1 Exit Meeting Summary

The inspection scope and findings were summarized on November 8, 1999. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

PARTIAL LIST OF PERSONS CONTACTED

Licensees

S. Bernhoft, Director, Nuclear Regulatory Affairs
J. Cowan, Vice President, Nuclear Operations
R. Davis, Assistant Plant Director, Operations
R. Grazio, Director, Nuclear Site and Business Support
G. Halnon, Director, Nuclear Quality Programs
J. Holden, Vice President and Director, Site Nuclear Operations
C. Pardee, Director, Nuclear Plant Operations
D. Roderick, Director, Nuclear Engineering & Projects
M. Schiavoni, Assistant Plant Director, Maintenance
T. Taylor, Director, Nuclear Operations Training

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B. Crowley, Reactor Inspector, Region II
P. Fillion, Reactor Inspector, Region II
E. Girard, Reactor Inspector, Region II
G. Kuzo, Senior Radiation Specialist, Region II
M. Scott, Reactor Inspector, Region II

INSPECTION PROCEDURES USED

IP 37001:	10 CFR 50.59 Safety Evaluation Program
IP 37550:	Engineering
IP 37551:	Onsite Engineering
IP 49001:	Inspection of Erosion/Corrosion Programs
IP 61726:	Surveillance Observations
IP 62707:	Conduct of Maintenance
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 73753:	Inservice Inspection
IP 83750:	Occupational Radiation Exposure
IP 92901:	Followup - Operations
IP 92903:	Followup - Engineering
TI 2515/142:	Draindown During Shutdown and Common-Mode Failure

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-302/99-07-01	NCV	Reactor Plenum Rigged Improperly (Section O1.5)
50-302/99-07-02	NCV	Failure to Properly Implement Procedure Results in Inadvertent Spent Fuel Pool Level Decrease. (Section O1.6)
50-302/99-07-03	NCV	Failure to Follow Procedure Results in Inadvertent Draining of the Reactor Coolant System. (Section O1.6)
50-302/99-07-04	NCV	Failure to Conduct Timely and Accurate Analysis of Potential Radionuclide Intakes by Workers. (Section R1.1)

Closed

50-302/99-07-01	NCV	Reactor Plenum Rigged Improperly (Section O1.5)
50-302/99-07-02	NCV	Failure to Properly Implement Procedure Results in Inadvertent Spent Fuel Pool Level Decrease. (Section O1.6)

50-302/99-07-03	NCV	Failure to Follow Procedure Results in Inadvertent Draining of the Reactor Coolant System. (Section O1.6)
50-302/99-07-04	NCV	Failure to Conduct Timely and Accurate Analysis of Potential Radionuclide Intakes by Workers. (Section R1.1)
50-302/99-04-00	LER	Main Feedwater Pump Trip During Refueling Shutdown Results in Emergency Feedwater Actuation. (Section O8.1)
50-302/97-11-01	IFI	RCS Reduced Inventory Level Indication Problems. (Section O8.2)
50-302/97-17-03	IFI	Review of Cable Ampacity Issue. (Section E8.1)
50-302/99-05-01	URI	Review Adequacy of Calibration Activities and Implementation of Verification Determinations for General Area Radiation Monitors. (Section R8.1)