APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-482/94-01

Operating License: NPF-42

Docket: 50-482

Licensee: Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, Kansas

Facility Name: Wolf Creek Generating Station

Inspection At: Coffey County, Burlington, Kansas

Inspection Conducted: January 16 through February 26, 1994

Inspectors: G. A. Pick, Senior Resident Inspector
J. F. Ringwald, Resident Inspector
L. E. Ellershaw, Inspector, Division of Reactor Safety

Accompanying Personnel: C. S. Mayberry, NRR Intern

Approved: LA Yandell — March 8, 1994
L. A. Yandell, Chief, Project Branch B — Date
Division of Reactor Projects

Inspection Summary

Areas Inspected: Routine, unannounced inspection including the areas of plant status, prompt onsite response to events at operating power reactors, operational safety verification, maintenance observations, surveillance observations, followup of corrective actions for violations, and onsite review of licensee event reports (LERs).

Results:

- A noncited violation was identified because engineering personnel failed to document troubleshooting steps while investigating the reactor coolant system isolation valve leakage (Section 3.2).

- A noncited violation was identified because engineering personnel failed to make required inservice test program pump and valve log entries. The inspector found the licensee's tools for managing inservice test results to be weak (Section 5.1).

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- The inspector found the licensee's response to an inoperable control rod to be appropriate. In addition, licensee management provided strong oversight of the rod control system troubleshooting activities (Section 2).

- The inspector found the operator's control of a plant heatup to be very good (Section 3.1).

- The inspector found the licensee's response to an emergency diesel generator (EDG) failure to stop to be appropriate (Section 3.3).

- Troubleshooting and maintenance activities were performed well, with good coordination with the control room (Section 4).

- Operators and technicians were found knowledgeable and proficient in the performance of surveillance tests. One engineering weakness in the incore/excore nuclear instrumentation calibration methodology was identified (Sections 5.2 - 5.7).

- The inspector found that the actions taken to resolve the issues raised in the essential service water self-assessment performed in October 1993 were well planned and comprehensive (Section 7.2).

Summary of Inspection Findings:

- Two noncited violations were identified (Sections 3.2 and 5.1).

- Violations 482/9327-01, 482/9327-02, and 482/9327-04 were closed (Section 6).

- LERs 482/93-006, 482/93-014, 482/93-015, and 482/93-016 were closed (Section 7).

Attachment:

- Attachment - Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS (71707)

On January 16, 1994, the licensee operated the plant in Mode 3 and had begun to heat up the reactor with criticality being obtained on January 17, 1994.

The licensee operated at 97 percent power from January 18-26 when the licensee initiated a controlled shutdown to troubleshoot and repair a control rod system problem. On January 27, 1994, the licensee manually tripped the reactor from 30 percent power. On January 28, 1994, while performing a reactor startup after repairing the control rod system, a control rod urgent failure alarm occurred at the moment the reactor reached the point of adding heat. This alarm was the result of a different problem than before, and the licensee manually tripped the reactor, reentered Mode 3, and repaired the rod control system. On January 29, 1994, the licensee commenced a reactor startup and achieved criticality, and reached 97 percent power the next day. The plant operated at 97 percent power through the end of the inspection period.

2 PROMPT ONSITE RESPONSE TO EVENTS AT OPERATING POWER REACTORS (93702)

2.1 Manual Reactor Trips

On January 24, 1994, while moving Shutdown Bank D control rods as part of the control rod exercise test, Control Rod M-12 dropped 10-12 steps. The licensee entered Technical Specification 3.1.3.1, Action 4, which allowed 72 hours to perform repairs and restore the rod to an operable status or initiate a controlled plant shutdown. The licensee developed a detailed troubleshooting plan to identify the problem in the rod control circuit. Troubleshooting plans were reviewed by the Manager, Maintenance and Modifications, who asked pertinent questions and provided recommendations, which were incorporated to aid in the problem identification. The licensee replaced five control circuit cards related to gripper coil timing operations and verified that the timing circuit actuated properly. During postmaintenance testing, Control Rod M-12 dropped a similar number of steps from the same position. The licensee hypothesized that a mechanical problem existed with Control Rod M-12. The licensee developed instructions to move Control Rod M-12 below the location where it had previously dropped. During the rod movement at the lower control rod height, Control Rod M-12 dropped and became misaligned from the other control rods within Shutdown Bank D by more than 12 steps.

Because of the misalignment, the licensee initiated the actions required by Technical Specification 3.1.3.1, Action 3, and reduced power below 75 percent. The licensee could not correct the problems prior to the expiration of the limiting conditions for operation of Technical Specification 3.1.3.1, Action 4, and initiated a Technical Specifications required shutdown on January 27, 1994. The licensee tripped the reactor from 30 percent power to troubleshoot and repair Control Rod M-12. The shift supervisor declared a notification of unusual event in accordance with Procedure EPP 01-2.1, "Emergency Classification," Revision 12, for a plant shutdown required by Technical Specifications, reported the event to the NRC Operations Center in accordance with 10 CFR 50.72, and reported the event as LER 482/94-002. The licensee identified that Control Rod M-12 had a loose connection in the power cable to the control rod gripper coils.

During the reactor startup on January 28, 1994, an urgent failure in the rod control circuit occurred preventing rod movement. The operators had just reached the point of adding heat and required control rods to easily control reactor coolant temperature (reactor power). The operators quickly compensated and stabilized the reactor at 0.5 percent power by adding boron. The licensee identified a failed thyristor that caused the urgent failure alarm and initiated a plant shutdown to repair the thyristor. The licensee manually tripped the reactor from 0.5 percent power on January 28, 1994. After replacing the thyristor, the licensee verified no other rod control problems existed, initiated a plant startup, and reached 97 percent power on January 30, 1994.

2.2 Posttrip Reviews

The inspector attended the posttrip review meeting conducted on January 27, 1994, in accordance with Procedure ADM 02-400, "Post-Trip Review," Revision 9. The licensee performed the posttrip review to evaluate whether plant components properly actuated. The review panel confirmed all equipment properly actuated as designed following the manual reactor trip. Licensee personnel reviewed the data and determined that nothing prevented restart of the reactor. The inspector verified that the appropriate licensee personnel attended the meeting as listed in Procedure ADM 02-400.

The licensee performed an adequate posttrip review following the manual trip from 0.5 percent power. All equipment actuated as required and no conditions prohibited startup. The inspector confirmed that the personnel properly characterized both manual reactor trips as Condition I. A Condition I trip indicates the cause of the trip was definitely known with no equipment failures or other abnormal conditions. The inspector concluded that the licensee decisions to restart were appropriate.

2.3 Assessment

On January 28, 1994, the inspector observed the reactor startup to the point of adding heat. The supervising operator, the shift supervisor, and the operations manager conducted pre-evolution briefings. During the operations manager's briefing, a reactor operator was taken away from the briefing while conducting licensed duties. No attempt was made by either the reactor operator to find out what details had been missed or by others attending the briefing to repeat the information for the operator's benefit. This observation was discussed with operations management.

Initially, the operators demonstrated formal communications during the startup with a superior level of repeat backs. As the startup progressed, this level of formality decreased. When the urgent failure in rod control system occurred at the point of adding heat, the supervising operator's direction to the reactor operator was, "You better get some boron in there." Despite the lack of consistent, formal communication, the operators worked well together even though two operators were from a different crew.

Initially, the supervising operator identified the tasks of these additional operators to the control room staff. The supervising operator did not, however, provide the staff with new information as their roles changed. This failure to keep all members of the control room staff fully informed did not appear to create any confusion. A reactor engineer worked well with the control room staff in monitoring the inverse count rate ratio plot and providing recommendations for subsequent rod withdrawals.

The inspector attended status meetings on January 28, 1994, at 1:30 and 6:30 p.m. related to the problems with Control Rod M-12. Operations management questioned the craft extensively about the work activities and problems encountered during troubleshooting. Specifically, the Vice President Plant Operations questioned the craft about the safety precautions, data analysis, coordination of clearance orders, and radiation work permits. After licensee management understood the scope of the activities, they allowed the craft personnel to perform the troubleshooting activities.

The Vice President Plant Operations decided to shut down the plant the second time because of personnel safety concerns while repairing the thyristor. Since this was not a Technical Specifications required shutdown, the shutdown did not require a declaration of a notification of unusual event. The licensee reported the manual reactor trip in accordance with 10 CFR 50.72 and described the manual reactor trip in LER 482/94-002.

The inspector reviewed the vendor manual, reviewed completed work requests, and interviewed the craft personnel involved with troubleshooting the rod control problem. The inspector determined that instrumentation and control (I&C) personnel and the system engineer performed thorough, detailed, and logical troubleshooting of the rod control system problems. The inspector found the qualified personnel to be knowledgeable of the system operation.

The inspector observed the reactor trip from 0.5 percent power. The inspector noted a superior level of repeat backs during recovery operations among the reactor operators and the supervising operator. The supervising operator provided instructions in a clear, easily understood manner. The supervising operator made the appropriate control room log entries, entered the emergency operating procedures, and properly transitioned into the affected plant operating procedure.

2.4 Conclusions

Licensee management provided strong oversight of the troubleshooting activities into the rod control problems. Operators provided clear, detailed communication responses during a reactor trip recovery. Licensed operators did not maintain consistent communication standards throughout the reactor startup. Operators did, however, perform a well controlled startup and responded appropriately to the rod control urgent failure alarm at the point of adding heat.

3 OPERATIONAL SAFETY VERIFICATION (71707)

The inspectors performed this inspection to ensure that the licensee operated the facility safely and in conformance with license and regulatory requirements and that the management control systems effectively discharged the licensee's responsibilities for safe operation.

The methods used to perform this inspection included direct observation of activities and equipment, observation of control room operations, tours of the facility, interviews and discussions with licensee personnel, independent verification of safety system status and Technical Specifications limiting conditions for operation, verification of corrective actions, and review of facility records.

3.1 Reactor Heatup

The licensee shut down the facility to implement a plant modification that would allow repair of Cavity Cooling Fan A at full power (refer to NRC Inspection Report 50-482/93-32, Section 2.5). On January 16, 1994, the inspector observed a portion of the reactor coolant system heatup from approximately 480°F to normal operating temperature. This was the temperature range where previous noise events occurred, as noted in NRC Inspection Report 50-482/93-14. No noise events and no seismic alarms occurred during this heatup. An engineer, stationed in the control room, evaluated indications of potential noise events by evaluating each loose parts monitor alarm. The inspector noted that the licensee could not record all occurrences because the loose parts monitor recording equipment recorded only 4 of the 12 channels monitored. The licensee was evaluating the merits of upgrading the recording equipment. The inspector considered control room communication to be formal and effective. The inspector verified that operators knew the heatup rate limits and noted that the heatup rate remained well below the limit.

3.2 Reactor Coolant System Isolation Valve Leakage

On January 17, 1994, operators promptly noted that they had been filling Accumulator C each shift (rather than once a week) and determined the leak to be 2 percent per hour. The licensee suspected that Valve EP HV8877C, Accumulator Tank C test line isolation, allowed the accumulator to drain since an outstanding work request (WR) documented valve leakage. The Vice President Plant Operations directed system engineering and operations to devise a method to stop the leakage to eliminate this operator distraction.

The licensee developed Temporary Modification 94-005-EM to place two freeze seals on the safety injection test line and install a manual isolation valve between the freeze seals. The licensee intended to use the manual valve to facilitate repair of Valve EP HV8877C. The licensee developed Plant Modification Request 04821, "Accumulator Test Line Isolation," to install the valve. Plant Modification Request 04821 specified the appropriate design requirements for the pipe class, provided the required postinstallation qualification test requirements, and contained a proper safety evaluation. The licensee initiated a single freeze seal to verify that the leakage was through Valve EP HV8877C. After verifying the integrity of the single freeze seal, the engineers determined that Accumulator C level continued to decrease, which indicated that the test line valve was not leaking. Consequently, engineers developed an additional troubleshooting strategy to isolate each line entering and leaving Accumulator C while monitoring the level.

On January 21, 1994, the inspector identified that the engineers had a well thought out troubleshooting plan but had not documented any completed troubleshooting steps. Upon questioning by the inspector, licensee personnel indicated that they planned to document their activities after completing the test. The inspector determined that the engineers' failure to document the troubleshooting steps was contrary to Procedure ADM 01-057, "Work Request," Revision 28, Attachment 3. Although the failure to satisfy the requirement to document the troubleshooting steps while performing them violated Technical Specification 6.8.1.a, this violation will not be cited because the licensee satisfied the criteria in paragraph VII.B.1 of Appendix C to 10 CFR Part 2 of the NRC's "Rules of Practice." The criteria was satisfied since the violation was of minor safety significance, the licensee took prompt corrective actions, and previous corrective actions would not have prevented this from occurring. The inspector verified that the completed WR detailed the troubleshooting steps performed. The licensee stated they would provide training to the system engineers on troubleshooting by March 31, 1994. For those system engineers presently in the system engineering training, the licensee will provide individual training prior to allowing them to perform troubleshooting activities.

From the troubleshooting activities, system engineers identified that Accumulator C leaked forward through Valve EP 8956C, Accumulator C discharge check valve, and leaked back through Valve EP 8818C, residual heat removal to accumulator injection line check valve. The engineers developed a Technical Specifications Interpretation which stated that any forward leakage through a valve of less than 1 gpm was not to be considered "flow." In addition, the engineers performed Procedure STS PE-019E, "RCS Isolation Check Valve Leak Test," Revision 9, at the direction of the Vice President Plant Operations to provide assurance that the leakage did not exceed Technical Specification 3.4.6.2.f limits for pressure isolation valve back leakage.

The leakage test fully seated Valve EP 8818C and reduced the leakage from Accumulator C. The licensee determined the new total pressure boundary leakage through all valves to be 1.95 gpm and verified that the leakage through each single valve was less than 1 gpm. From a review of the test data, the inspector noted that the postcalibration verification check of the pressure gauge found the instrument slightly out of calibration. The licensee performed the instrument calibration check following this test because of the importance of the test. The system engineer corrected the as-found values by performing a simple ratio. The inspector verified the engineer's assumption that the value changed linearly.

The inspector noted that senior management provided a high degree of oversight by monitoring the progress of the investigation and by requiring that the pressure isolation check valve back leakage test be performed. The engineers performed thorough evaluations when developing Temporary Modification 94-005-EM and when adjusting the measured values of a slightly out-of-calibration flow instrument.

3.3 EDG Test

On January 20, 1994, during the postmaintenance test of EDG A, the diesel engine failed to stop when the operator pressed the stop push button. The licensee initiated WR 00469-94 to troubleshoot and correct the failure to stop. The licensee suspected that a three-way control valve (a bleed-off valve) failed and prevented the rack boost, start, and shutdown fuel control cylinder from securing the fuel supply to the cylinders. After replacement of the three-way control valve, the EDG again failed to stop during the postmaintenance test. Further troubleshooting identified that an O-ring inside the rack boost, start, and shutdown fuel control cylinder failed, preventing proper operation. The licensee installed a replacement O-ring for the rack boost, start, and shutdown fuel control cylinder and successfully retested EDG A.

The licensee considered EDG A operable even though the engine failed to stop upon demand. The design of the EDG power system is to start and run to provide emergency power for safety-related components during design basis accidents. Previously, EDG A failed to stop in January 1993. The licensee, however, determined the root cause to be sticking excess flow check valves. The licensee corrected this problem during Refueling Outage VI (refer to NRC Inspection Reports 50-482/93-01 and 50-482/93-08). The licensee had no problems with EDG B and determined that the Callaway facility had no similar problems with their EDGs. The licensee initiated WR 00498-94 to have the EDG B rack boost, start, and shutdown fuel control cylinder inspected and cleaned during Refueling Outage VII.

3.4 Conclusions

The licensee performed appropriate troubleshooting of the EDG A failure to stop. The licensee demonstrated by engineering judgement that the problem affected only EDG A. Licensee management provided good oversight of and clear expectations for the excessive leakage from a safety injection accumulator.

4 MAINTENANCE OBSERVATIONS (62703)

During this inspection period, the inspectors observed and reviewed the selected maintenance and activities listed below to verify compliance with regulatory requirements and licensee procedures, required quality control department involvement, proper use of caution tags, appropriate radiation worker practices, calibrated test instruments, and proper postmaintenance testing. Specifically, the inspectors witnessed portions of the following maintenance activities:

- Environmentally Qualified Solenoid Valve Replacement
- Limitorque Motor-Operated Valve (MOV) Maintenance
- Main Control Board Handswitch Replacement
- Control Valve Troubleshooting

4.1 Environmentally Qualified Solenoid Valve Replacements

On January 26, 1994, the inspector observed portions of the replacements of environmentally qualified Solenoid Valves AB HY-0005, AB HV0005 Solenoid Valve-Main Steam Loop 2 to auxiliary feedwater pump turbine isolation, and AB HY-0006, AB HV0006 Solenoid Valve-Main Steam Loop 3 to auxiliary feedwater pump turbine isolation. The inspector identified that warehouse personnel properly marked the replacement valves and found the work instructions appropriate. The electricians removed the solenoid valve and its associated conduit and performed the qualified crimps and splices needed in the electrical shop.

A quality control inspector noted that the existing splice did not meet the environmentally qualified requirements of Procedure MGE EOOC-12, "Raychem WCFS Tubing (In-Line Splices), and NPKV Stub Connection Kit Select, Install and Inspect," Revision 2. The splice heat shrink material overlapped the wire braid (did not overlay a smooth surface) and did not have at least 2 inches of overlap. The electricians initiated Performance Improvement Request (PIR) 94-0203 for corrective action and Reportability Evaluation Request 94-04 to determine reportability. The inspector determined that the new splices conformed to environmental qualification requirements.

4.2 Limitorque MOV Maintenance

On February 10, 1994, the inspector observed electricians perform the 3-year preventive maintenance inspection implemented by WR 60644-92 on MOV EG HV0054, Component Cooling Water Train B to containment and radwaste building isolation valve. The WR specified several procedures to be used while performing the maintenance activity. The inspector found that the detailed procedures provided appropriate qualitative guidance for evaluating the MOV actuator conditions. The licensee properly documented tightening of several slightly loose terminal nuts, addition of grease to the gear case, and replacement of a cracked finger base. The licensee replaced the cracked finger base in accordance with WR 00908-94. There had been no problems with valve operation prior to performance of the maintenance activity. The inspector questioned the shift supervisor regarding operability of the MOV. The shift supervisor replied that after discussions with the electricians he concluded that the cracked finger base had not affected operability of the MOV. The finger base had cracked on the outer edge of a screw hole.

4.3 Main Control Board Handswitch Replacement

On February 10, 1994, the inspector observed electricians replace the main control board handswitch for Component Cooling Water Pump B in accordance with WR 06110-93 work instructions. WR 06110-93 documented that, when placed in pull-to-lock, the handswitch contacts that input to the engineered safety features status panel indicating light failed to continuously illuminate the light. The inspector found the electricians to be knowledgeable. The electricians revised the work package to remove a barrier that allowed easy access to the handswitch wire terminations. A quality control inspector provided continuous job coverage.

The licensee established an appropriate postmaintenance test. The test required that operators verify that the engineered safety features status panel window illuminated and that the operators started and stopped the pump. The supervising operator modified Procedure SYS EG-120, "Component Cooling Water System Startup," Revision 14, by marking "not applicable" all steps other than those required for starting/stopping the pump and verifying that the engineered features status light illuminated. No problems were identified.

4.4 Control Valve Troubleshooting

On February 18, 1994, the inspector observed portions of the troubleshooting following the surveillance testing failure described in paragraph 5.7. Following the initial test failure, I&C technicians realigned the control valve signal and re-biased the load limit circuit. Following the second failure, the inspector observed I&C technicians realign the control valve signal meters and troubleshoot a 60 MWe discrepancy between the load set and indicated load meters.

The technicians determined that the lower steam pressure following the unit rerate (refer to NRC Inspection Report 50-482/93-29, paragraph 2.10) caused the control valves to open further than normal, which required a longer close stroke time. The longer close stroke time allowed the first stage pressure feedback signal enough time to generate a load limit signal that removed the permissives for valve testing and redirected the control valve in the open direction. Operators reduced the unit load until control valves controlled at 40 percent open and repeated the test that again failed. Operators further reduced the unit load and repeated the test with the control valves at 35 percent open with satisfactory results.

4.5 Conclusions

Personnel performed maintenance in accordance with well written instructions and procedures. A supervising operator identified the appropriate steps needed to perform postmaintenance testing in accordance with management expectations and administrative procedures. I&C technicians performed troubleshooting on the main turbine control valves well and appropriately coordinated their activities with operators. The inspector concluded that the solenoid valve replacements were performed appropriately and noted that a quality control inspector identified the as-found nonconforming splices.

5 SURVEILLANCE OBSERVATIONS (61726)

The inspectors reviewed this area to ascertain whether the licensee conducts surveillances of safety-significant systems and components in accordance with Technical Specifications and approved procedures.

5.1 Inservice Test Review

The inspector reviewed the inservice test program used to meet the requirements of Technical Specification 4.0.5. The review included the pump and valve log and the weekly inservice test reports during the past year. The inspector identified two examples of failure to make pump and valve log entries required by Procedure ADM 05-200, "ASME Code Testing of Pumps and Valves," Step 4.1.6.1.4. The first example involved the failure to log removing Component Cooling Water Pump B from "alert" on September 13, 1993. The second example involved the failure to log the removal of motor-driven Auxiliary Feedwater Pump B from "alert" on November 22, 1993. These failures to make required entries are examples of a violation of Technical Specification 6.8.1.a. This violation of Technical Specification 6.8.1.a is not being cited because the licensee satisfied the criteria in paragraph VII.B.1 of Appendix C to 10 CFR Part 2 of the NRC's "Rules of Practice." The licensee promptly made pump and valve log entries and issued PIR 94-0379 as a result of these inspector identified concerns.

The inspector found numerous minor inconsistencies and errors in the weekly inservice test report. The inspector determined that the guidance provided in Procedure ADM 05-200 for making pump and valve log entries and for developing the weekly inservice test report was vague and confusing. Discussions with the inservice test engineer revealed that personnel issued the weekly inservice test report for information only and did not consider it important for the report to be completely accurate. Several personnel changes during the past year also contributed to inconsistent interpretation of the confusing guidance for making entries in these reports. Examples of discrepancies included not listing valves in the required action range, not describing the reason for removal of a component from the action range, and documenting incorrect dates for when components entered or exited the action or alert range. These types of errors made it difficult for management to manage and track component performance. The inspector concluded that this represented a weakness in the management of the inservice testing program.

The PIR corrective actions included revising Procedure ADM 05-200 to clarify requirements for the pump and valve log and the weekly inservice test report. Entries will be required for every pump and valve that enters or is removed from the alert or required action range. The inservice test engineer responsible for the pump and valve log and the weekly inservice test report was developing the procedure changes. The inspector concluded that these changes should address the identified concerns.

5.2 Component Cooling Water Pump B Inservice Test

On February 2, 1994, the inspector observed system engineers perform the quarterly inservice test of Component Cooling Water Pump D in accordance with Procedure STS EG-100B, "Component Cooling Water Pumps B/D Inservice Pump Test," Revision 10. The inspector verified that the engineers used the latest procedure revision and received permission to start from the supervising operator. The system engineer performed a detailed prejob brief in accordance with plant procedures. All personnel involved with the test performance attended the prejob brief.

The system engineer reviewed and followed the procedure steps during the test performance. The inspector verified that licensee personnel used calibrated gauges and test equipment. Personnel properly vented the test gauges and assured that the required flow had been maintained for at least 5 minutes prior to taking vibration, flow, and pressure data.

The inspector found the test procedure to be well written, clear, and easy to follow. The procedure specified appropriate acceptance criteria and verified that all data met acceptance criteria. Procedure STS EG-100B also verified that Valves EG V0016, Component Cooling Water Pump D discharge check valve, and EG V0012, Component Cooling Water Pump B discharge check valve, opened properly.

5.3 Incore-Excore Nuclear Instrumentation Calibration

On February 8, 1994, the inspector observed I&C technicians adjust setpoints for a power range instrument. The I&C technicians performed the surveillance in accordance with Procedure STS IC-447, "Channel Calibration Nuclear Instrumentation System Power Range Incore-Excore," Revision 10. The inspector noted that the I&C technicians obtained permission to start and explained the expected annunciators to the operator, as specified in the procedure. The I&C technicians performed the procedure in a careful, deliberate manner. From discussions with the I&C technicians, the inspector found them to be knowledgeable of the procedure purpose and familiar with the procedure content. The inspector found the procedure to be detailed, verified the instruments to be calibrated, and found the repeat back, during communications, to be extensive.

From discussions with licensee personnel and review of control room logs, the inspector identified that personnel did not complete Procedure STS IC-447 that began on February 7, 1994. The operators determined that axial flux difference (AFD) computer calculated validation values exceeded the acceptance criteria. The procedure ensured accuracy of the computer generated AFD values by verifying that expected reference values agreed with the actual reference values within ±1 percent tolerance. The licensee added this verification to Procedure STS IC-447 in January 1993 in response to a recommendation contained in a vendor technical bulletin.

The inspector determined that the licensee had added the computer generated AFD reference values in January 1993. Engineering personnel used the computer generated AFD reference values, which are normalized for power, rather than the vendor recommended axial offset values, which are independent of power, because AFD was an input to several other procedures and alarms and AFD values were considered more accurate. However, the personnel failed to recognize the effects of performing the verification at other than 100 percent power when using AFD. All values used in this procedure were normalized to 100 percent except AFD, thus introducing an error in the verification. The licensee initiated PIR 94-0290 for the failure to recognize the problem created by not using axial offset. Licensee personnel stated that no changes to the calibration of the incore-excore detectors have occurred due to this error. The computer calculated values were used only for verification purposes.

5.4 Local Leak Rate Test

On February 10, 1994, the inspector observed, in part, a system engineer and a test engineer perform local leak rate testing of the 18-inch containment minipurge valves in accordance with Procedure STS PE-015, "18-inch Purge Valve Leakage Test," Revision 6. The inspector found that the procedure provided appropriate guidance for the conduct of the test. From interviews, the inspector found the individuals to be very knowledgeable of the test methodology and test requirements. The inspector verified that engineers used calibrated test instruments and obtained the appropriate test conditions by stabilizing the pressure at 48.6 psig. Upon arrival at the test location, the inspector observed a quality assurance auditor observing the test.

From review of the completed test procedure, the inspector determined that all data met test requirements. A quality assurance auditor reviewed the test as part of the Technical Specifications and Surveillance Test audit. The auditor initiated PIR 94-0297 documenting that engineers connected the test equipment out of sequence, which could have overranged the test equipment. The missequencing of the test equipment occurred prior to the inspector's arrival. No problems occurred because the pressure regulator was adjusted at the appropriate setpoint.

5.5 EDG Air Start Tank Check Valve Test

On January 22, 1994, the inspector observed an operator test the EDG starting air tank check valves, in accordance with Procedure STS KJ 002B, "Air Start System B Starting Air Tank Check Valve Test," Revision 6, for EDG B. The inspector found that the operator carefully followed the procedure and that it provided adequate guidance. The operator obtained satisfactory test results, and the test satisfied Technical Specification Surveillance Requirement 4.0.5.

5.6 Valve Stroke-Time Test

On January 24, 1994, the inspector observed an operator perform a valve stroke-time surveillance test of Valve BL HV8047, reactor makeup water containment isolation valve, in accordance with Procedure STS BL-205, "Reactor Makeup Water System Inservice Valve Test," Revision 6. The licensee performed the surveillance as a postmaintenance test following replacement of the environmentally qualified valve position limit switches. The inspector found that the procedure provided adequate guidance for the conduct of the test. The operator performed the test correctly and obtained satisfactory test results. The test satisfied Technical Specification Surveillance Requirements 4.0.5 and 4.6.3.3.

5.7 Main Turbine Control Valve Test

On February 18, 1994, the inspector observed an operator perform Procedure STS AC-002, "Main Turbine Valve Cycle Test," Revision 8, that verified operability of the main turbine control valves. The inspector found the procedure to be adequate and noted that the operators followed it. The inspector observed unsatisfactory test results on two occasions while operators tested the first turbine control valve. The valve moved to approximately 5 percent closed then returned to full open. The inspector observed a portion of the troubleshooting as discussed in paragraph 4.4.

In preparation for the second test, one operator noted that either the "at set load," "load increasing," or "load decreasing," indicators should have been illuminated when all three were dark. The operators and an I&C technician determined that both lamps failed in the "at set load" indicator and immediately replaced them. The inspector concluded that this represented good familiarity with the control board indications and good attention to detail. Following the troubleshooting, operators repeated the test on all four control valves and obtained satisfactory results. The test satisfied Technical Specification Surveillance Requirements 4.3.4.2.b and c.

5.8 Conclusions

The inspector found the licensee's tools for managing inservice test results to be weak. The inspector identified that reactor engineers used AFD rather than axial offset while performing power range nuclear instrumentation calibrations The use of AFD only affected the reference values used to verify accuracy of the computer points. The inspectors found operators and technicians knowledgeable and proficient in performing surveillance testing.

6 FOLLOWUP ON CORRECTIVE ACTION FOR VIOLATIONS (92702)

6.1 (Closed) Violation 482/9327-01: Inadequate Clearance Order Implementation

This violation occurred on September 21, 1993, because licensee personnel failed to properly position and failed to verify that the actual valve positions for boric acid filter inlet and outlet valves reflected the required reach rod position.

The licensee attributed the root cause to personnel error for failing to use their self-checking program. As immediate corrective actions, licensee management conducted discussions with the shift supervisor involved to stress the importance of self-checking. Corrective actions to prevent recurrence included conducting a meeting with the shift supervisors and supervising operators to discuss the importance of reducing personnel errors by applying self-checking, a questioning attitude, and personnel accountability and teamwork. In addition, the licensee placed the PIR that described the event into required reading for operations personnel.

The inspector reviewed the meeting notes and interviewed the operations manager about the subjects discussed during the shift supervisor/supervising operator meeting. The inspector verified through review of required reading attendance sheets that operations personnel had read the PIR. The inspector found the licensee's corrective actions to be appropriate.

6.2 (Closed) Violation 482/9327-02: Inadequate Control of Temporary Equipment

The inspector identified this violation of Procedure ADM 01-201, "Control of Temporary Equipment," while touring the auxiliary building. The inspector found several carts with rollers that were not immobilized to restrict/limit movement. The licensee identified the root cause for the violation as personnel misinterpretation of the procedure requirements and a lack of familiarity with the procedure requirements.

Licensee management expressed their expectations about the requirements set forth in Procedure ADM 01-201 during quality time meetings. The licensee took this approach to eliminate any confusion regarding the procedure. Corrective actions to prevent recurrence involved modifying the procedures, developing designated storage areas, and training craft personnel on the modified procedure requirements.

Since the licensee implemented their immediate corrective actions, the inspector has identified no other failures to control temporary equipment. The inspector verified from discussions with the responsible individual that the scope and content of revised Procedure ADM 01-201 had been discussed with affected onsite personnel. The inspector determined that the licensee intends to establish designated storage areas by May 1994.

6.3 (Closed) Violation 482/9327-04: Failure to Obtain Permission to Start Work

Contrary to Procedure ADM 01-057, "Work Request," on October 12, 1993, the inspector observed an individual perform work in accordance with WR 02815-93 without first obtaining the shift supervisor's permission. The licensee attributed the root cause to personnel error/lack of attention to detail and lack of appropriate self-checking. Corrective actions included discussing this event and appropriate self-checking at a quality time meeting.

The inspector verified by review of the meeting summary that maintenance management discussed with craft personnel the appropriate self-checking techniques and the requirement for obtaining shift supervisor approval prior to commencing work.

7 ONSITE REVIEW OF LERs (92700)

7.1 (Closed) LER 482/93-006: Hot Particle Resulted in Exposure Exceeding 10 CFR 20.101 Requirements

On April 2, 1993, the licensee determined that a nonlicensed operator had a hot particle on his left thigh. The nonlicensed operator obtained the hot particle while draining the refueling cavity into a containment drain trench. The licensee determined the particle to be fission products from Cycle 6 fuel failures. The nonlicensed operator received an estimated skin exposure of 33.9 rem (8 microcurie-hours).

As documented in NRC Inspection Report 50-482/93-08, Section 2.5, the licensee performed a detailed, thorough investigation that identified the root cause and several contributing factors. The corrective actions addressed the contributing factors. The corrective actions included developing a preventive maintenance activity to decontaminate the drain trench grating, revising the lesson plans for watchstanders that have occasions to drain contaminated systems, training of licensed personnel on the revised venting/draining requirements and fabricating nozzles to eliminate backsplash.

The inspector verified that the licensee had included in essential reading a description of proper venting/draining techniques. Operations personnel must sign that they have read essential reading prior to standing watch. The inspector verified that the licensee altered the qualification requirements for both the auxiliary and radioactive waste building watchstanders. The inspector reviewed Lesson Plan 1637001, "Vent and Drain," and found that the information provided good instructions for venting and draining contaminated systems. From review of attendance sheets, the inspector verified that personnel received the training. The inspector verified that the licensee included the PIR on the contamination event in operations required reading. From discussions with personnel the inspector determined that they were familiar with the fabricated nozzles and proper actions for draining contaminated systems into the containment drain trenches. The inspector confirmed that the licensee scheduled WR 60128-93 for performing decontamination of the containment drain trench gratings during Refueling Outage VII.

7.2 (Closed) LER 482/93-014: Essential Service Water (ESW) System Low Flow Prior to Refueling Outage IV

As discussed in NRC Inspection Report 50-482/93-29, the licensee completed a service water system self-assessment in October 1993 and concluded that, in 1991, they had incorrectly determined the reportability of degraded ESW system flows. The licensee notified the NRC in accordance with 10 CFR 50.72 and reported the results of their evaluation in LER 482/93-014. The LER contained a comprehensive discussion regarding the basis for reportability, event description, root causes and contributing factors, corrective actions, preventive measures, and a safety analysis.

During the service water system self-assessment, the licensee reviewed Defect Deficiency Report (DDR) 90-021, initiated on May 1, 1990, to document that ESW flow rates may not have satisfied all required component design flows under accident conditions. In addition, DDR 90-021 addressed the need to reduce throttle valve positions to several components in order to ensure adequate flow to other components during postulated accident conditions. Subsequently, on February 5, 1991, the licensee determined that these conditions were not reportable in that no specific as-found data existed. NUREG-1022, "Licensee Event Report System," states that if there is no firm evidence to identify the time of initiation (of the deficient condition) other than the time of discovery then the event is not reportable. However, normal mode and postloss-of-coolant accident (LOCA) flow balancing performed on both trains of the ESW system during Refueling Outage IV (March-May 1990) identified problems that likely existed prior to Refueling Outage IV and indicated overall system fouling.

Inadequate flow to ESW components could have prevented the fulfillment of the safety functions that were needed to mitigate the consequences of an accident. The licensee's service water system self-assessment review resulted in a determination that this condition should have been reported; therefore, the licensee notified the NRC in accordance with 10 CFR 50.72 and initiated LER 482/93-014.

The licensee performed a root cause analyses to evaluate the previous reportability determination, operability, and system degradation. With respect to the incorrect reportability determination, the licensee established that there had been a lack of procedural controls regarding a formal evaluation process for DDRs and there existed no mechanism in the DDR procedure to initiate or track an engineering review. The licensee established that an operability determination had not been initiated at the time they discovered the throttle valve position changes because the plant was in Mode 6 and the ESW system was not required to be operable. System degradation resulted from general corrosion and organic fouling at a much faster rate than anticipated. This increased flow resistance in piping and components. The licensee determined that testing/monitoring programs in place at the time were inadequate for the detection of system degradation.

The events described in LER 482/93-014 occurred over 3 years ago, and the licensee corrected the low flow conditions prior to restart from Refueling Outage IV. Subsequently, the licensee implemented programmatic controls as described in NRC Inspection Report 50-482/93-29. The inspectors concluded that, had those controls been in place during the [illegible] of the events described in LER 482/93-014, there would have been little likelihood for those events to have occurred.

The licensee performed analyses of the affects on operability and functionality on the following safety-related equipment suspected of having low ESW flow prior to Refueling Outage IV: EDG coolers, auxiliary feedwater pump room Cooler B, penetration room coolers, spent fuel pool room coolers, containment air coolers, control room air conditioning Unit B, and centrifugal charging pump room Cooler B. While the containment air coolers were mentioned, the licensee previously addressed their analysis and corrective action (i.e., replacing annubars with ultrasonic flow measurement test equipment) in LER 482/91-002 and were excluded from this discussion. The licensee based the analyses on conservative estimates of the ESW system flow to the affected components. In addition, they used engineering judgement on the affects of lower-than-design flow on the ability of the components to perform their safety-related functions. With available data, including data from ESW system flow balancing during Refueling Outage IV, the licensee established approximate flows (including an allowance for conservatism) for each of the above components and evaluated this flow analysis against the worst case design basis accident. Using these conservative assumptions and sound calculations, each component was determined to be functional and considered operable. Initially, based strictly on engineering's conservatively established flow rates, it appeared that the EDG coolers may not have been operable. However, after taking into account other known factors and applying them to the conservative flow rates, the licensee determined that the EDGs remained functional and operable.

The inspectors reviewed each of the analyses (i.e., assumptions, calculations, and conclusions) and determined that the licensee utilized conservative assumptions and correct calculational methodology. The inspectors concluded that the licensee's conclusions were reasonable and proper.

The licensee addressed the formation of an assessment team to research the reportability and operability determination procedures in existence at the time of the event and those used currently. This effort was compiled in report form and titled "Self Assessment for LER 93-014," dated December 27, 1993. The assessment evaluated implementation and effectiveness of the corrective actions pertaining to LER 482/93-014. The inspectors considered the self-assessment to be a strong effort towards evaluating known conditions, identifying potential problems, and recommending suggestions to improve the program. The report addressed actions previously taken to ensure functionality and operability of the ESW system. In particular, the inspectors noted and verified that Procedure STN PE-037, "ESW Heat Exchanger Flow and Differential Pressure Trending," Revision 8, had required the monitoring of flow and differential pressure through all ESW heat exchangers every quarter since August 1990. Flushing of the heat exchangers is an option for the test engineer, however, it has become increasingly used during the course of any given test. The licensee analyzed the data for degraded flow and differential pressure conditions on all components. The inspectors reviewed selected quarterly test results to verify implementation of procedural requirements. In addition, the inspectors verified that the licensee performed post-LOCA flow balancing of both ESW A and B trains during Refueling Outage V by using Procedures TP-TS-41, "ESW Train A Post LOCA Flow Balance," Revision 0, and TP-TS-50, "ESW Train B Post LOCA Flow Balance," Revision 0, respectively.

The self-assessment report also addressed training of system engineering personnel. The inspectors were provided information pertaining to an 8-hour training course developed by Regulatory Compliance (4 hours each on operability and reportability). This training was provided on January 19, 1994, to the qualifying shift technical advisors by the individual responsible for making the reportability decision. The training was scheduled to be provided to the shift supervisors prior to the start of Refueling Outage VII. The licensee scheduled the training of the system engineers to commence during February 1994 and be completed during June 1994. The training material included a draft "Reportability Handbook." Regulatory compliance developed the handbook using training materials received from other licensees, regulatory requirements, and industry groups. The inspectors considered the handbook to be an excellent source of information for help in making reportability evaluations. The inspector determined that the expected revision to Procedure ADM 01-033, "Instructions For Evaluating, Reporting, and Documenting Potentially Reportable Events," Revision 29, should incorporate the reportability handbook by reference. The licensee expected to complete the procedure revision by mid-March 1994.

The inspectors considered the actions taken by the licensee in response to their identification of previous ESW system low flow conditions and the associated incorrect determination of reportability to be well planned and comprehensive.

7.3 <u>(Closed) LER 482/93-015: Breech of Containment During Refueling Operations Due to Sampling</u>

This event occurred during shutdown conditions, Modes 5 and 6, when containment integrity was required. Because of inadequate procedure guidance, chemistry personnel did not isolate the sample lines during Modes 5 and 6 because they did not understand containment integrity would be violated while changing the particulate filters.

The licensee implemented thorough corrective actions, as documented in NRC Inspection Report 50-482/93-29, Section 2.2. The inspector verified that the licensee reviewed 64 sampling procedures by December 24, 1993, and

23 administrative procedures by January 31, 1994. The inspector interviewed chemistry personnel and found that no other sampling procedures had deficiencies that would violate containment integrity. The licensee improved the affected procedures by ensuring each action step was an individual step.

7.4 (Closed) LER 482/93-016: Programmatic Deficiencies Result in Unqualified Fire Brigade Member

On November 12, 1993, the licensee identified that an individual who had not completed all required fire brigade training requirements in the third quarter 1993 stood watch as a fire brigade member on November 1, 1993. The licensee reviewed the other eligible fire brigade members' training to assure their qualifications had not lapsed. The licensee attributed the root cause to incomplete and inadequately implemented corrective actions for LER 482/90-022. The licensee determined that licensee personnel did not consistently review the computer database; delays had been experienced in updating the database; and periods that the computer was not available. Consequently, personnel relied on the hard copy fire brigade roster, Special Order 3. Even though the licensee updated Special Order 3 quarterly, changes occurred monthly.

The licensee's corrective actions included establishing Special Order 3 as the primary source of information to validate fire brigade member qualifications, using the computer program as a backup source of information, updating Special Order 3 prior to the end of each month, and updating the fire protection program guidance. The inspector verified the licensee completed each of the corrective actions.

ATTACHMENT 1

1 PERSONS CONTACTED

G. D. Boyer, Manager, Training
R. Q. Dunlap, Regulatory Services
C. W. Fowler, Manager, Maintenance and Modifications
R. B. Flannigan, Manager, Nuclear Safety Engineering
D. E. Gerrelts, Manager, Instrumentation and Control
D. Jacobs, Supervisor, Mechanical Maintenance
W. M. Lindsay, Manager, Quality Assurance
R. L. Logsdon, Manager, Chemistry
P. M. Martin, Assistant Manager, Operations
O. L. Maynard, Vice President Plant Operations
T. S. Morrill, Manager, Radiation and Protection
W. B. Norton, Manager, Nuclear Engineering
L. D. Ratzlaff, Supervisor, System Engineering
F. T. Rhodes, Vice President Engineering
C. E. Rich, Jr., Supervisor, Electrical Maintenance
T. L. Riley, Supervisor, Regulatory Compliance
E. W. Schmotzer, Manager, Purchasing and Material Services
R. L. Sims, Supervisor, Operations Support
B. B. Smith, Manager, Modifications
C. M. Sprout, Manager, System Engineering
S. M. Walgren, Shift Supervisor, Operations
J. D. Weeks, Assistant to Vice President Plant Operations
S. G. Wideman, Supervisor, Licensing
M. G. Williams, Manager, Plant Support

The above personnel attended the exit meeting. In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on February 28, 1994. During this meeting, the inspectors summarized the scope and findings of the report. The licensee acknowledged the inspection findings identified in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.