APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-482/92-12 Operating License No.: NPF-42

Docket No.: 50-482

Licensee: Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, Kansas 66839

Facility Name: Wolf Creek Generating Station

Inspection At: Coffey County, Burlington, Kansas

Inspection Conducted: May 31 through July 11, 1992

Inspectors: G. A. Pick, Senior Resident Inspector
L. E. Myers, Resident Inspector

Approved: Mark A. Satorius 8/4/92
for A. T. Howell, Chief, Project Section D Date
Division of Reactor Projects

Inspection Summary

Inspection Conducted May 31 through July 11, 1992 (Report 50-482/92-12)

Areas Inspected: Routine, unannounced inspection including plant status, inoffice review of written reports of nonroutine events at power reactor facilities, operational safety verification, surveillance observations, and maintenance observations.

Results: In the area of plant operations, performance was good. Licensed personnel promptly entered Technical Specification (TS) limiting conditions for operation (LCOs) when notified of nonconforming conditions (Sections 4.3, and 4.5).

Strengths and weaknesses were identified in the area of corrective actions. After an individual received low levels of contamination, the licensee created a root cause evaluation team. The creation of the team indicated increased management oversight (Section 4.4). The licensee's initial response to an NRC Bulletin that pertains to Thermo-Lag fire barrier issues was prompt and conservative (Section 4.2). The licensee identified numerous Westinghouse Technical Bulletins and NRC Information Notices (INs) that were not evaluated or were closed without being adequately evaluated. NRC previously identified an example of an inadequately reviewed NRC IN. An inspection followup item will be used to track the licensee's ongoing evaluations of the subject INs

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(Section 4.8). Additional examples of not completing surveillance requirements within the specified surveillance interval were indicative of less than fully effective corrective actions associated with previous surveillance scheduling problems (Sections 4.3 and 4.7).

Surveillance performance was mixed. All observed surveillances were performed well. On-the-job instruction of hot-license candidates was excellent. Communications during observed surveillances was exemplary (Section 5). The licensed operators properly diagnosed excessive seat leakage through a check valve even though the acceptance criteria was subjective (Section 4.4). However, an additional failure to meet TS surveillance requirements resulted from a loss of information in the scheduling database (Section 4.7). An emergency diesel generator (EDG) surveillance was not performed within the specified surveillance interval (Section 4.3). Finally, a spray additive tank was inadvertently diluted because of failure to follow a test procedure (Section 4.1).

Maintenance activities, observed by the inspectors, were preformed well. Review of completed work activities determined that the work instructions were well-written and detailed (Section 6). However, recurring problems with battery charger voltage fluctuations and an emergency diesel generator lube oil thermostatic control valve were noted (Sections 6.1 and 6.2).

A list of acronyms and initialisms is provided in the attachment to this report.

DETAILS

1. Persons Contacted

B. D. Withers, President and Chief Executive Officer
J. A. Bailey, Vice President, Operations
F. T. Rhodes, Vice President, Engineering and Technical Services
T. M. Anselmz, Licensing Engineer
R. S. Benedict, Manager, Quality Control
A. B. Clason, Supervisor, Maintenance Engineering
T. F. Deddens, Jr., Manager, Outage
M. E. Dingler. Manager, Nuclear Plant Engineering Support
D. L. Fehr, Manager, Operations Training
R. B. Flannigan, Manager, Nuclear Safety Engineering
C. W. Fowler, Manager, Instrumentation and Control
R. A. Hammond, Health Physics
N. W. Hoadley, Manager, Equipment Engineering, Nuclear Plant Engin ring
R. W. Holloway, Manager, Maintenance and Modifications
T. P. Hood, Manager, System Engineering
D. Jacobs, Supervisor, Mechanical Maintenance
R. K. Lewis, Supervisor, Results Engineering
R. L. Logsdon, Manager, Chemistry
D. G. Naylor, Supervisor, Operations Support
C. E. Parry, Director, Quality and Safety
A. L. Payne, Manager, Supplier/Material & Quality
E. M. Peterson, Supervisor, Quality Assurance Audits
B. B. Smith, Manager, Modifications
S. G. Wideman, Supervisor, Licensing
M. G. Williams, Manager, Plant Support

The above licensee personnel attended the exit interview conducted on July 15, 1992. The inspectors held discussions with the Director, Plant Operations and various other licensee and contractor personnel during this inspection.

2. PLANT STATUS

The plant remained at or near full power throughout the inspection period.

3. INOFFICE REVIEW OF WRITTEN REPORTS OF NONROUTINE EVENTS AT POWER REACTOR FACILITIES (90712)

The inspectors verified that the following Licensee Event Reports (LERs) properly described the subject events, that corrective actions taken and planned addressed the root cause of the event, and that the corrective actions should prevent further occurrences.

3.1 (Closed) LER 90-022: Technical Specification (TS) Violation - Unqualified Individual Assuming Fire Brigade Member Duties Because of Lack of Administrative Controls

3.2 (Closed) LER 90-025-01: Both Safety Injection Pumps Inoperable Because of Frozen Minimum Recirculation Line to the Refueling Water Storage Tank (RWST)

3.3 (Closed) LER 90-006: Engineered Safety Features Equipment Actuations and Technical Specification Violation Caused by Inadequate Procedural Guidance

4. OPERATIONAL SAFETY VERIFICATION (71707)

The objectives of this inspection were to ensure that the facility was being operated safely and in conformance with license and regulatory requirements, and that the licensee's management control systems were effectively discharging the licensee's responsibilities for continued safe operation. The inspectors monitored licensee activities related to: inadvertent dilution of spray additive tank (SAT), licensee response to failures in Thermo-Lag 330 fire barrier systems, a valid EDG failure, chemical and volume control system (CVCS) check valve leakage, failed containment instrument tunnel sump level instrument, power operated relief valve (PORV) block valve seat leakage, missed TS surveillance tests, and industry experience reviews.

The methods used to perform this inspection included direct observation of activities and equipment, control room observations, tours of the facility, interviews and discussions with licensee personnel, independent verification of safety-system status and TS LCOs, corrective actions, and review of facility records.

4.1 Inadvertent Dilution of SAT

On June 3, 1992, during the performance of a surveillance test, an inadvertent dilution of the SAT occurred when a misalignment of valves resulted in the transfer of approximately 100 gallons of water from the RWST to the SAT. The level of the sodium hydroxide (NaOH) solution in the tank and the concentration of the NaOH solution became indeterminate as a result of the dilution.

Upon the direction of a licensed operator, a nuclear station operator (NSO) performed Procedure STS EN-205, Revision 6, "Containment Spray System Inservice Valve Test," Step 5.2.3, that filled and vented the system in preparation for stroking Valve EN HV-16, SAT to containment spray B isolation valve. The NSO closed EN-111, SAT outlet header valve, and had the closure of the valve independently verified by another licensed operator. The NSO then proceeded to put away the equipment used to vent the system without informing the licensed operator of the completion of the step. After the equipment was secured, the NSO assumed, without communicating with the licensed operator, that the licensed operator had completed the closure stroke-time test of Valve EN HV-16 in accordance with Steps 5.2.4 and 5.2.5. The NSO proceeded to restore the system in accordance with Section 6.0 and, as the NSO opened Valve EN V097, spray additive tank outlet isolation valve, he heard a pressure surge. The licensed operator closed Valve EN HV-16 to prevent further dilution of the SAT with RWST water and directed that the NSO close Valve EN V097. Approximately 100 gallons of RWST borated water injected into the SAT. SAT level increased by 2 percent from 98 percent. One hundred percent level corresponds to 4534 gallons.

The licensee declared the containment spray additive system inoperable at 6:06 a.m, on June 3, 1992, until the concentration of the NaOH solution and the level of the SAT could be determined. TS 3.6.2.2 requires that the SAT contain a volume between 4340 and 4540 gallons at a concentration from 28 to 31 percent by weight of NaOH solution. The licensee determined that the normal method for measuring the NaOH concentration may not be accurate because the added RWST water may stratify. Also, the system design had no provisions for mixing the tank. Therefore, samples were obtained at several levels at 2-foot intervals from top to bottom, which indicated a range of NaOH concentration from 27.74 to 28.94 percent by weight in the tank. The licensee restored the SAT level and concentration by lowering the tank level to 90 percent and adding approximately 400 gallons of 50 percent by weight NaOH solution to the tank. The level was restored to 98 percent. The average concentration of NaOH solution from four samples obtained at different levels was 29.9 percent by weight. The containment spray additive system was declared operable at 2:40 p.m., on June 5, 1992.

The inspector discussed the event with various licensed personnel and reviewed Performance Improvement Request (PIR) OP92-0430. Also, the inspector reviewed Procedure STS EN-205 and Procedure TP-OP-258, "TEN01 Drain/Fill," that were implemented to restore NaOH concentration to within TS limits. The inspector noted that communications between the licensed operator and the NSO were not effective because there was confusion about who was directing the test. The PIR indicated that the NSO may have been under pressure to complete the test because he was nearing the end of his shift. Failure to perform step-by-step performance of Procedure STS EN-205 is a violation of TS 6.8.1.a for failure to properly implement a surveillance procedure (482/9212-01). The licensee's immediate corrective actions to prevent further dilution of the SAT and to restore the SAT level and NaOH concentration were effective.

4.2 Licensee Response to Failures in Thermo-Lag 330 Fire Barrier Systems

On June 24, 1992, the inspector provided the licensee a copy of NRC Bulletin 92-01, "Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage." After a brief review of the contents of the Bulletin and the location of Thermo-Lag fire barriers in the plant, the licensee instituted 1-hour fire watches in all affected areas of the plant including the turbine, auxiliary, spent fuel, and radwaste buildings as required by the fire protection manual for degraded fire barriers. The licensee's review of Thermo-Lag fire barriers located in the containment building identified one 1 1/2-inch conduit containing cabling for pressurizer level instrumentation with a 3-hour Thermo-Lag fire barrier. The licensee's preliminary engineering evaluation determined that the fire barrier was changed and evaluated as a noncombustible radiant energy shield but was left constructed as a 3-hour fire barrier. In accordance with 10 CFR Part 50, Appendix R, Section III, G.2.L and Generic Letter 86-10, "Implementation of Fire Protection Requirements," a radiant energy shield may be used to separate redundant train components in containment with a 1/2-hour fire rating.

The licensee's preliminary evaluation determined that there was reasonable assurance that a 3-hour Thermo-Lag fire barrier installation would survive a 1/2-hour endurance test; therefore, the licensee concluded that no compensatory measures were necessary. The licensee initiated approximately 200 work requests (WRs) to allow for individual evaluation and tracking of all identified Thermo-Lag fire barriers. At the end of the inspection period, the licensee was reviewing Thermo-Lag fire barrier applications to determine which were Appendix R and non-Appendix R.

4.3 Valid EDG Failure

On June 8, 1992, while performing the TS operability test for EDG B, the licensed operators noted that the lube oil and bearing temperatures were high. The licensed operators stopped the EDG and exited Procedure STS KJ-005B, Revision 14, "Manual/Auto Start, Synchronization, and Loading of Emergency Diesel Generator NE02." The licensed operators promptly entered TS 3.8.1.1, and declared the EDG inoperable. The inspectors monitored the repair activities and root cause evaluation (refer to Section 6.1).

The licensee returned the EDG to service on June 9, 1992. The necessary information was provided to the licensing department so the NRC-required special report could be written and to the plant trending department so that the number of failures could be tracked. An individual in the trending group counted the number of valid test failures on June 26, 1992. Since EDG B had five valid failures in the last 100 tests, the EDG should have been operated at least once per 7 days in accordance with TS 4.8.1.1.2.a, Table 4.8-1, "Diesel Generator Test Schedule." The surveillance requirement specified in TS Table 4.8-1 was missed on two occasions (June 16 and 23, 1992). The shift supervisor declared the EDG inoperable in accordance with TS 4.0.3, which specifies that the failure to meet a surveillance requirement constitutes a failure to meet the operability requirements, and entered TS 3.8.1.1, Action b, which allows 72 hours to return the inoperable EDG to operable status. Following successful completion of a 1-hour operability test in accordance with Procedure STS KJ-005B, EDG B was declared operable. This is Example 1 of a violation for failure to comply with TS surveillance requirements (482/9212-02). The licensee issued PIR PS92-0499 to assure the root cause would be identified and appropriate corrective actions taken. The licensee will document this failure to follow TS in LER 92-011.

The inspector reviewed the previous special reports that documented valid test failures for EDG B and determined that a previous valid test failure described in Special Report 89-003 was caused by power pill failures (refer to Section 6.1 for a description of the power pills). The licensee determined the most likely scenario for the power pill failures was overheating of the power pills; however, there were no EDG trips or high jacket water temperatures recorded that would be indicative of such a failure. The licensee increased the frequency for replacing the EDG jacket water thermostatic control valve (TCV) power pills to once every 18 months from the vendor-recommended 60 months since a root cause was not positively identified.

Following the 1989 failure of the EDG jacket water TCV, the licensee failed to maintain the EDG on an increased testing frequency until the number of valid test failures in the last 20 EDG starts was one as required by TS 4.8.1.1.2.a, Table 4.8-1. The licensee described this event in LER 89-001. As corrective action, the licensee counselled the individuals on being more diligent and paying closer attention to detail. However, the licensee failed to develop any programmatic controls that specified the time frame required for counting the number of test failures, and the licensee failed to develop a procedure to implement the requirements for meeting TS 4.8.1.1.2.a, Table 4.8-1. The inspector considered the failure to develop programmatic controls, following the first occurrence in 1989, to be indicative of corrective action program weaknesses that previously have been identified in recent NRC inspection reports.

4.4 CVCS Check Valve Leakage

On June 12, 1992, during performance of Procedure STS BG-210, Revision 10, "CVCS Inservice Check Valve Test," licensed operators determined that Check Valve BG V589, Centrifugal Charging Pump (CCP) A to seal injection filters, leaked excessively. The operators could not establish the conditions for completing Section 5.1 of the procedure since the Train B pressure could not be decreased. The operators determined that Check Valve BG V589 was leaking after they noticed that Valve BG HV8357B, which isolated CCP B discharge to the seal injection filter, lifted from its seat while they tried to decrease pressure downstream of Check Valve BG V589. Valve BG V589 and its companion check valve limit the leakage from the seal injection line of the inservice train to the other train in the event of a failure. The flow through Valve BG HV8357B was not a problem because the valve was not designed to stop flow in the reverse direction. The licensee danger-tagged out CCP B, isolated flow from the volume control tank, and determined the leakage through Check Valve BG V589 to be approximately 1 gallon per minute (gpm).

Valve BG V589 is a Category C check valve with no specific acceptance criteria; however, the licensee included this valve as part of their Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," evaluation which is scheduled to be completed by December 31, 1992. The specified acceptance criteria for Valve BG V589 was a pressure greater than 50 pounds per square inch differential with no rapid pressure increase. The onshift shift supervisor determined that the seat leakage for Valve BG V589 was excessive in relation to the Train A valve. The Train A piping repressurized in approximately 7 minutes while Train B repressurized in approximately 1 1/2 minutes.

The licensee pursued two concurrent courses of action. The first course of action required developing a work package for repair or replacement of Check Valve BG V589. In order to establish an isolation boundary for the repair of BG V589, a freeze seal would be required. The licensee discussed the need for preparing an engineering justification to support a temporary waiver of compliance since the maintenance could take longer than the 72 hours allowed by the TS. The second course of action required evaluating the safety significance of the "as found" leak rate to determine whether the repairs were necessary or whether the valve could be used "as is."

The engineering disposition determined that the licensee does not take credit for the 80 gpm seal injection flow in any design-basis accident; consequently, this degraded condition did not represent an operability concern. Since the leakage into the seal injection line could come from several sources, the licensee developed a plan that evaluated possible sources. The inspectors observed the performance of this surveillance activity (refer to Section 5.3). The licensee danger-tagged out of service CCP B, isolated the volume control tank and danger-tagged closed the cross-connect valve to Train A. The licensee quantified the leakage through BG V589 at 0.767 gpm, with leakage from other sources determined to be 0.013 gpm. The licensee determined the leakage through the check valve was acceptable for use "as is" based on the engineering disposition. The valve will be replaced during the next outage of sufficient duration.

On June 18, 1992, while trying to reperform the valve surveillance, a licensed operator was contaminated. The operator opened the bleed valve in accordance with the procedure while watching for pressure to decrease and water to stop flowing to the drain. After noting that the water stopped flowing, the operator rotated the pressure gage so that he could read the pressure. At that moment, a blockage in the line cleared. This resulted in the drain hose lifting from the drain and spraying reactor coolant into the boron injection tank room. The operator grabbed the hose and directed the flow into the drain. Since a trash screen covered the drain, the operator became soaked from the backsplash. Subsequently, the operator redirected the flow to a corner of the room until he closed the bleed valve. The operator was slightly contaminated on both hands and behind the right ear, but no internal contamination was identified from ear and nasal smears.

The licensee initiated a radiological occurrence report and a PIR to assure that a root cause was identified and that corrective actions would be taken for the personnel contamination event. The licensee formed an evaluation team consisting of an operator, an engineer, and a health physicist. The licensee researched previous radiological occurrence reports and identified three personnel contamination events that were caused by drain hoses coming loose from floor drains. The licensee determined that human factors considerations contributed to the event. The gage should face the operator, and a method should be identified to better secure the drain hoses. Several other actions were recommended; however, the final corrective actions were not completed by the end of the inspection period.

4.5 Failed Containment Instrument Tunnel Sump Level Instrument

On June 12, 1992, after determining that the computer point for calculating the containment leak rate was unavailable, the licensee researched the actions necessary to perform a manual calculation of the total containment leak rate. Control panel indication of the instrument tunnel sump level was also unavailable. Licensed operators determined that a conflict existed between Procedure STS LF-001, Revision 3, "Containment Normal Sump Inventory and Discharge Determination," and Procedure SYS LF-120, Revision 2, "Containment Leak Detection System Operation." The shift supervisor entered the action statement for TS 3.4.6.1 because of the indeterminate condition of the "Containment Normal Sump Level Measurement System."

Procedure SYS LF-120 includes both the normal and instrument tunnel sump levels as part of the leak rate while Procedure STS LF-001 uses the normal sump in the leak rate determination. The operators researched the TS and the Updated Safety Analysis Report (USAR) and determined that the instrument tunnel leak rate was not to be included in the TS leak rate determination. However, USAR Section 9.3.3.2.1.1 specified that the containment and instrument tunnel sumps measure unidentified leakage. The licensee initiated PIR OP92-0451 to ensure the leakage was determined in accordance with TS, the USAR, and RG 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," requirements. The licensee initiated a reportability evaluation request to evaluate reportability of the failure to determine leakage into the instrument tunnel sump.

The licensee determined that a conflict existed among the TS, the USAR, and RG 1.45 because the TS Bases specify that the monitoring requirements of TS 3.4.6.1 meet the intent of RG 1.45. However, the licensee concluded that the failure to include the instrument tunnel sump in the TS was intentional and not inconsistent with the TS Bases. The instrument tunnel sump can be used to locate and quantify the source of any unidentified leakage; however, detection would occur more quickly by other methods. Since the TS did not require surveillance of the instrument tunnel sump, there was no violation of TS and no requirement to report the unavailability of the instrument tunnel sump level indication. The inspectors concurred with the licensee's determination. The licensee determined that the USAR requirement for containment sump level and flow monitoring systems meets the full intent of RG 1.45. The resolution of any discrepancies and corrective actions taken to clarify differences among the TS, the USAR, and RG 1.45 will be documented under PIR OP92-0451.

4.6 PORV Block Valve Seat Leakage

In March 1992, following startup from the forced shutdown, the licensee closed PORV Block Valve BB HV8000B after determining that PORV BB PCV456A had a small amount of seat leakage. The licensee closed the block valve in an attempt to limit steam cutting induced wear to the seat of the PORV and comply with

TS 3.4.4.a. TS 3.4.4.a requires that one or more inoperable PORV(s) (due to seat leakage) must be restored to an operable status or its associated block valve must be closed.

On June 25, 1992, a licensed operator noticed an increase in the rate of pressure rise in the pressurizer relief tank (PRT). The licensee began investigating, by the process of elimination, to determine the cause of increased rate of rise in PRT pressure. The operators closed the nitrogen and waste gas manual isolation valves to determine whether these gas sources caused the increased rise in pressure. Pressure increased even though the isolation valves were closed. An operator entered containment and utilized a pyrometer to check the downstream side of several relief valves that relieve to the PRT. No leakage was detected. The licensee routed the seal leakoff that normally goes to the PRT to the reactor coolant drain tank, which eliminated the seal leakoffs as a source of increased leakage. On July 3, 1992, the licensee took temperatures on the upstream side of the PORV block valves, between the PORV block valves and their respective PORVs, and downstream of the PORVs. The temperature readings indicated that there was zero leakage through PORV BB PCV455A and that there was slow leakage through both PORV BB PCV456A and its block valve, BB HV8000B. The licensee determined that the pressure in the PRT rises approximately 3.0 pounds per square inch gage every 48 hours, which does not represent an operational concern.

4.7 <u>Missed TS Surveillance Tests</u>

On July 7, 1992, while entering a completed surveillance test into a manual tracking log, an individual noticed that the surveillance interval was longer than previously completed surveillance tests. The individual researched the affected surveillances and determined that the surveillances were performed past the due date and that the scheduling program issued incorrect intervals. The affected surveillance tests were STS PE-14A, Revision 8, "Containment Air Locks Test (Personnel Hatch)," and STS PE-14B, Revision 8, "Containment Air Locks Test (Equipment Hatch)."

The surveillances implement the overall air lock leakage tests as required by TS 4.6.1.3.b. The air locks are required to be tested every 6 months at a pressure of 48 pounds per square inch gage and verified to meet leakage rate limits. A foot note associated with TS 4.6.1.3.b states that the provisions of TS 4.0.2 do not apply. TS 4.0.2 allows a specified surveillance interval to be extended by up to 25 percent of the interval. Surveillance STS PE-14A became late on May 26, 1992, but was performed on June 25, 1992, 31 days overdue. Surveillance STS PE-14B became late on March 21, 1992, but was performed on April 15, 1992, 26 days overdue. The failure to conduct TS surveillances within the specified interval is Example 2 of Violation 482/9212-02.

The licensee postulated that the computer program surveillance scheduling problems were created during the first half of 1991. Computer support group personnel modified the scheduling software to delete the TS 4.0.2 requirement that prevented scheduling a surveillance 3.25 times past the specified interval. The change was implemented after this requirement was deleted from TS 4.0.2 by a TS amendment. The licensee determined that the error was not in the computer software but in the database. The flags that ensured the requirement to multiply the surveillance interval by 1.25 or to subtract a fixed period of time from the due date were deleted from the database inadvertently. The licensee had not determined how the data field was eliminated; however, the licensee postulated that the most likely cause was related to the change in the database to implement the change to the TS 4.0.2 requirements. Other surveillances that were not subject to the TS 4.0.2 provisions were misscheduled but were not overdue. The licensee initiated PIR CS92-0528 to evaluate methods and controls used by the computer support group to make modifications to the database and to determine appropriate corrective actions. The corrective actions were not completed by the end of the inspection period.

The inspectors noted that several other TS violations had occurred because of various surveillance scheduling problems. The inspectors concluded that corrective actions to prevent recurrence have not been fully effective.

4.8 Industry Experience Reviews

During a review of PIRs, the inspector noted that PIR SE92-0208 documented that several Westinghouse Technical Bulletins, issued from 1973 to 1985, had not received evaluations under the licensee's industry technical information program (ITIP). The PIR stated that 31 technical bulletins transferred from licensing to the support group were not evaluated. The ITIP coordinator subsequently verified that 6 of the 31 were previously evaluated. On July 8, 1992, during a meeting with the ITIP coordinator, the inspector determined that the licensee had reviewed 14 of the remaining 25 technical bulletins. The ITIP coordinator closed 8 of the 14 technical bulletins during initial review because the subject matter was not applicable to Wolf Creek Generating Station or the action was already implemented. Another six were being evaluated by the licensee as of the end of the inspection period.

The licensee also initiated a review of the ITIP database to identify all NRC Information Notices (INs) that were closed during initial review. The inspector determined from discussions with the licensee that of 339 INs reviewed, 43 INs appeared to have been inappropriately evaluated and closed during the initial screening process. NRC previously identified a similar problem pertaining to inadequate review of an IN (refer to NRC Inspection Report 50-482/92-02).

By the end of the inspection period, three of the 43 INs were reviewed by the licensee. One was adequately addressed under another ITIP on the same subject. The other two INs were being reviewed by the applicable work groups. Followup of the licensee's evaluations and corrective actions related to these ITIPs will be tracked by Inspection Followup Item 482/9212-03.

Conclusions

A violation of TS 6.8.1.a occurred as a result of an a NSO failing to follow a test procedure. This resulted in diluting the SAT. Licensee actions taken to prevent further dilution and to restore the SAT within limits were effective. The licensee's initial response to NRC Bulletin 92-01 was prompt and conservative. The licensee failed to meet TS surveillance requirements that resulted in an inoperable EDG. The failure to properly implement the surveillance requirement within the specified surveillance interval is a violation of TS 4.8.1.1.2.a, Table 4.8-1. A violation of TS 4.8.1.1.2.a, Table 4.8-1 previously occurred in 1989. The inspectors considered the failure to implement programmatic controls following the 1989 event to be another example of previously identified corrective action program weaknesses. A second example of failure to meet TS surveillance requirements occurred because of scheduling database errors. The inspectors noted that several previous TS violations had occurred because of surveillance scheduling weaknesses. Corrective actions to prevent recurrence have not been fully effective. The review and evaluation of the cause and effect of the excessive check valve leakage, the instrument tunnel sump level detector malfunction, and the increased rate of pressure rise in the PRT were extensive. The licensee performed excellent investigations into these issues. The results of the licensee's reevaluation of NRC INs that were closed during initial screening will be tracked by an inspection followup item.

5. SURVEILLANCE OBSERVATIONS (61726)

The purpose of this inspection was to ascertain whether surveillance of safety-significant systems and components was being conducted in accordance with TS.

5.1 Steam Calorimetric Verification

On June 16, 1992, the inspector observed a licensed operator and a hot license candidate perform a calorimetric in accordance with Procedure STS SE-002, Revision 6, "Manual Calculation of Reactor Thermal Power." The procedure was performed to determine whether the power range nuclear instruments required adjustment. The operators requested the data at 5-minute intervals, with six sets of data taken. The data included steam flow, feedwater temperature, steam pressure, and indicated power. The procedure provided instructions for performing the necessary calculations. The operators transferred the data to a spreadsheet that performed the calculations.

The inspector determined from discussions with the hot license candidate that he was familiar with the procedure purpose and content. Both individuals ensured that data was properly transferred. The procedure was well written and easy to follow.

5.2 <u>Position Indication Test</u>

On July 6, 1992, the inspector observed NSO activities at Valve AL HV009, Steam Generator B motor-driven auxiliary feedwater (MDAFW) pump regulating valve, during performance of a full-stroke and a position indication test. The valve was tested in accordance with Procedure STS AL-201, Revision 10, "Auxiliary Feedwater System Inservice Valve Test." The stroke time during the observed testing was approximately 17 seconds, which was the nominal, expected value. The valve was tested on an increased frequency because of the results of previous stroke-time tests. In March 1992, the open-stroke time was approximately 14 seconds, which was 3 seconds faster than nominal. However, in June 1992, the valve stroked open in 18 seconds, which represented an approximately 25 percent increase in the stroke time. Consequently, the inservice test engineer placed the valve on an increased frequency even though the stroke time was close to the average. The valve will be tested one additional time on an increased frequency; however, the inservice test engineer believed that the quicker open-stroke time was an aberration because the other stroke times were more consistent.

The NSO established communications as required by the procedure. During performance of the test, the communications demonstrated by the NSO with the control room were specific, detailed, and informative. From discussions with the NSO, the inspector determined that he was knowledgeable about the test purpose and test steps and had reviewed the procedure in the control room prior to performing the test.

5.3 <u>CVCS Check Valve Testing</u>

On June 18, 1992, the inspectors observed a partial test performance of Procedure STS BG-210, Revision 10, "CVCS In-Service Check Valve Test." The licensee performed the test to verify that the test line repressurized because of leakage through Check Valve BG V589 (refer to Section 4.4).

A licensed operator provided direction, as needed, to the hot license candidate who was conducting the test. From discussion with the hot license candidate, the inspector determined he had read the procedure and was familiar with the test purpose and scope. The supervising operator conducted a test prebriefing with the control room personnel and personnel performing local valve manipulation. When the test was performed, communication was established among the NSO stationed outside the boron injection tank room, the NSO inside the room manipulating the valves, and the control room. A health physics technician was present to provide oversight of radiological protection practices.

No problems were identified.

5.4 <u>Inservice Test of MDAFW Pump</u>

On June 23, 1992, the inspector observed licensed operators perform portions of the operability test for the MDAFW Pump B as required by TS 4.0.5 and

TS 4.7.1.2.1.a.1. The test was implemented in accordance with Procedure STS AL-104, Revision 14, "MDAFW Pump B In-Service Pump Test." The inspector noted that the procedure contained adequate precautions and limitations. The communications between the licensed operator and the NSO were good. The inspector identified no problems.

Conclusions

On-the-job training of hot license candidates was well performed during this inspection period. Excellent communications occurred among licensee personnel. Test personnel followed good radiation protection practices.

6. MAINTENANCE OBSERVATION (62703)

The purpose of inspections in this area was to ascertain that maintenance activities on safety-related systems and components were conducted in accordance with approved procedures and TS. Methods used in this inspection included direct observations of maintenance activities and review of records.

6.1 Maintenance on EDG B Lube Oil Cooler TCV

On June 8, 1992, during performance of the EDG operability test in accordance with Procedure STS KJ-005B, the licensed operators received high lube oil temperature and high bearing oil temperature alarms, exited the procedure, entered TS 3.1.1, and initiated WR 2903-92 (refer to Section 4.3). Troubleshooting indicated that the lube oil cooler TCV failed to control lube oil flow through the heat exchanger which, in turn, maintains proper lube oil temperature. Mechanics disassembled the valve, replaced all three of the power pills, and reassembled the valve. The power pills are hydraulic units containing wax inside a metal housing which expands against a rubber diaphragm and metal plug assembly. Three power pills in series provide the force required to position the valve. A postmaintenance test determined that two of the three power pills failed to respond to the maximum opening temperature of 175°F. The observed maintenance was performed well, and the work instructions were well written and detailed.

The licensee, in accordance with vendor recommendations, had instituted a 5-year service life for the power pills used in the lube oil and intercooler TCVs. Because of a commitment made in Special Report 89-003 concerning a failure of a jacket water TCV with the same type power pills, the licensee replaced the power pills in the jacket water TCVs every refueling outage. The power pills replaced in lube oil TCV had a short service life. The licensee determined that the vendor had recommended a 3-year shelf life for the power pills. The installed power pills had a greater than 3-year shelf life. As a result, on July 7, 1992, they were replaced with newly obtained power pills. The licensee also changed the procurement documents so that there will be a 3-year shelf life on all power pills. The inspectors noted that this TCV previously failed in April 1992 because of a power pill failure.

6.2 Failures of the NK 23 Battery Charger

On June 7, 1992, fluctuating voltage on the NK 23 battery charger to the NK 13 125 volt direct current bus caused the battery monitor alarm to annunciate. Troubleshooting determined voltage fluctuated from 135 to 138 volts direct current, and adjustment of the float voltage did not correct the voltage fluctuation. The amplifier and firing boards were replaced in the charger and a defective lug to the firing board was replaced by WR 2394-92. Individual battery cell voltages were checked and found to be within specification. The voltage and current of the battery charger were monitored for 24 hours with no further fluctuations noted.

On June 15, 1992, fluctuating voltages were again observed on Charger NK 23. The licensee discussed the voltage fluctuation with the vendor. The vendor suggested that loose rectifiers could cause the failures, but may be eliminated as the cause if the charger operated properly after the board replacements. On June 20, 1992, the voltage fluctuations increased to the point where amplifier and firing board replacement was again necessary. Voltage and current of the battery were monitored for 24 hours and found to be steady and within specifications. Further discussions were held with the vendor to determine the root cause of the failures, and the boards were returned to the vendor for failure analysis. Although the observed activities were performed well, the inspectors noted that battery charger voltage fluctuations have been a recurring problem and the root cause has not been determined.

Conclusions

Although the observed maintenance activities were performed well, problems with battery charger voltage fluctuations and EDG lube oil TCVs are recurring.

7. EXIT MEETING

The inspectors met with licensee personnel (denoted in paragraph 1) on July 15, 1992. The inspectors summarized the scope and findings of the inspection. The licensee did not identify as proprietary any of the information provided to, or reviewed by, the inspectors.

ATTACHMENT

Acronym List

ADM	administrative procedure
CCP	centrifugal charging pump
CVCS	chemical and volume control system
EDG	emergency diesel generator
gpm	gallons per minute
ITIP	Industry Technical Information Program
IN	information notice
LCO	limiting conditions for operation
LER	licensee event report
MDAFW	motor driven auxiliary feedwater
NaOH	sodium hydroxide
NRC	U.S. Nuclear Regulatory Commission
NSO	nuclear station operator
PIR	performance improvement request
PORV	power operated relief valve
PRT	pressurizer relief tank
RG	regulatory guide
RHR	residual heat removal
RWST	refueling water storage tank
SAT	spray additive tank
STS	surveillance technical specification
TCV	thermostatic control valve
TS	Technical Specification
USAR	Updated Safety Analysis Report
WR	work request